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W. K. Hughey
Director
Nuclear Safety & Regulatory
Affairs

April 9, 1997

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
1996 Grand Gulf Nuclear Station (GGNS) Annual
Environmental Operating Report (AEOR)

GNRO-97/00028

Gentlemen:

Attached is the Grand Gulf Nuclear Station (GGNS) Annual Environmental Operating Report (AEOR) for the period January 1, 1996 through December 31, 1996. 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 Gary C. Coker at (601) 437-2242, or this office.

Yours truly,

G A Daughtery FOR

C001/,

WKH/MJL
attachment: 1996 Annual Environmental Operating Report
cc: (See Next Page)

cc: GGNS NRC Senior Resident (w/a)
Mr. R. B. McGehee (w/a)
Mr. N. S. Reynolds (w/a)
Mr. H. L. Thomas (w/o)
Mr. J. W. Yelverton (w/o)

Mr. E. W. Merschoff (w/a)
Regional Administrator
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GRAND GULF NUCLEAR STATION

1936 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, 1996.

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

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, 1996. 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

GGNS activities did not include any unreviewed environmental questions during 1996. Review of environmental evaluations indicated routine matters within the scope of expected activities. GGNS did not observe any environmental consequences as a result of conduct of the activities evaluated. Table 4-1, which has the evaluations attached, summarizes environmental evaluations performed in 1996.

4.0 ADMINISTRATIVE REQUIREMENTS

4.1 EPP Changes

GGNS made no changes to the EPP in 1996.

4.2 EPP Noncompliances

GGNS activities contained no EPP noncompliances during 1996.

4.3 Nonroutine Reports

GGNS submitted no nonroutine reports in 1996.

4.4 Potentially Significant Unreviewed Environmental Issues

GGNS encountered no potentially significant unreviewed environmental issues in 1996. GGNS personnel made changes in station design and operation, tests and experiments, of which none resulted in an unreviewed environmental question, in accordance with the EPP, paragraph 3.1, Plant Design and Operation.

Table 4-1

1996 Environmental Evaluation Summary *

Safety and Environmental Evaluation Number	**Description**
96-0026-R00	Revises SAR wording which requires the Radiation Control Manager to review each plant design change or modification, after the change has been through the development review and approval process, during the implementation authorization review process for ALARA concerns as part of the ALARA program. The change is to require the review of change documents as directed by the GGNS General Manager.
96-0069-R00	Incorporates the use of electronic alarming dosimeters into the UFSAR
96-0069-R001	Evaluates the use of electronic alarming dosimeters and modifies Table 12.5-1 of the UFSAR.
96-0071-R00	Storage of radioactive materials in areas outside the controlled access area, protected area and plant structures.
96-0085-R00	Reduce the surveillance test frequency of interlock functions.
96-0087-R00	Documents troubleshooting efforts for recirculation flow control valve B (1B33F060B).
96-0098-R00	Revises Technical Requirements Manual surveillance requirement 7.6.3.3.g.2 to allow the Standby Liquid Control pump relief valves to be tested at least once per 18 months during plant or system shutdowns to verify that they open within the specified 3% tolerance.

*** See attached for completed evaluations.**

CA RECD. B14.33
NC. QA RECORD
INITIALS
NUMBER of PAGES 5
DATE 6/19/96
RELATED DOCUMENT NUMBER N/A

96-021-PSE

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

I. Safety Evaluation Overview

A. Reference Data

ORIGINATOR: (1) DENNIS FRANKLIN   DEPT/SECT: (2) P&SE   EVAL. #: (3) 96-0026-R-00

DOCUMENT EVALUATED: (4) UFSAR CR # 96-035

SYSTEM NO:(S) (5) N/A   (INSERT N/A IF NOT APPLICABLE)

REFERENCES: (6) REG. GUIDE 8.8, NPEAP 311

FSAR CHANGE REQUIRED? (7) ☑ Yes ☐ No CR# 96-035

FSAR SECTIONS TO BE REVISED: 12.1.1.2

TRM CHANGE REQUIRED? (8) ☐ Yes ☑ No

TECH. SPEC. CHANGE REQUIRED? (9) ☐ Yes ☑ No CR# CR or (n/a) N/A

IS THE VALIDITY OF THIS SAFETY EVALUATION DEPENDENT ON ANY CHANGES OTHER THAN THE CHANGE BEING EVALUATED (E.G. PROCEDURAL, OPERATIONAL CONDITIONS)? (10) ☐ Yes ☑ No

EXPLAIN: NO OTHER CHANGES ARE REQUIRED TO SUPPORT THIS SAR CHANGE REQUEST.

IF YES TO THE LAST QUESTION, HAVE THE ORGANIZATIONS RESPONSIBLE FOR THOSE CHANGES BEEN NOTIFIED? (11) ☐ Yes

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

Signatures and Approvals of Attached Safety and Environmental Evaluation

Evaluated: (12) DENNIS FRANKLIN (PRINT NAME) / ORIGINATOR Dennis Franklin (SIGNATURE) 4-1-96 DATE

Reviewed: (13) DE JOHNSON (PRINT NAME) / INDEPENDENT REVIEWER DE Johnson (SIGNATURE) 4-1-96 DATE

Reviewed: (14) Ricky Fili (PRINT NAME) / OTHER REVIEWER (IF REQUIRED) R Fili (SIGNATURE) 4-1-96 DATE

Plant Safety Review Committee Review

(15) Charles M Dugger
CHAIRMAN, PSRC   4-4-96 DATE

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

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT [16]

THE PROPOSED CHANGE REVISES SAR WORDING WHICH REQUIRES THE RADIATION CONTROL MANAGER TO REVIEW EACH PLANT DESIGN CHANGE OR MODIFICATION, AFTER THE CHANGE HAS BEEN THROUGH THE DEVELOPMENT REVIEW AND APPROVAL PROCESS, DURING THE IMPLEMENTATION AUTHORIZATION REVIEW PROCESS FOR ALARA CONCERNS. AS A PART OF THE ALARA PROGRAM. THE CHANGE IS TO REQUIRE THE REVIEW OF CHANGE DOCUMENTS AS DIRECTED BY THE GGNS GENERAL MANAGER.

REASON FOR CHANGE, TEST OR EXPERIMENT [17]

THE PROPOSED CHANGE IS INTENDED TO HELP STREAMLINE THE PROCESS FOR AUTHORIZATION OF CHANGES FOR IMPLEMENTATION, ONCE ISSUED BY ENGINEERING AS AN APPROVED DESIGN PACKAGE. THE RADIATION PROTECTION MANAGER REVIEW IS REDUNDANT TO OTHER SIMILAR REQUIREMENTS EFFECTIVE DURING DESIGN DEVELOPMENT AND WORK IMPLEMENTATION PROCESSES AND IS NOT NEEDED.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS [18]

THE PROPOSED CHANGE TO THE SAR IS TO ELIMINATE A REDUNDANT REVIEW BY THE RADIATION PROTECTION MANAGER OF APPROVED (ENGINEERING AND PLANT STAFF) DESIGN CHANGES, DURING THE CHANGE IMPLEMENTATION AUTHORIZATION PROCESS. THE PROPOSED CHANGE TO THE SAR IS TO AN ADMINISTRATIVE PROCESS THAT HAS NO DIRECT OR INDIRECT IMPACT ON THE OPERATION, CONTROL OR FUNCTION OF ANY GGNS PLANT EQUIPMENT, AND THEREFORE WILL NOT AFFECT THE SAFETY OF OPERATIONS. DESIGN CHANGES UNDERGO ALARA DESIGN REVIEW DURING THE DEVELOPMENT PROCESS IN ACCORDANCE WITH DESIGN ENGINEERING ADMINISTRATIVE PROCEDURES, AND EACH WORK ORDER THAT IMPLEMENTS A PLANT CHANGE OR MODIFICATION IS REVIEWED BY APPROPRIATE RADIATION CONTROL/HEALTH PHYSICS PERSONNEL FOR ALARA CONCERNS. THE PROPOSED CHANGE CANNOT INCREASE THE PROBABILITY OF OCCURRENCE OR CONSEQUENCES OF AN ACCIDENT PREVIOUSLY EVALUATED IN THE SAR, INCREASE THE PROBABILITY OF OCCURRENCE OR CONSEQUENCES OF A MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY PREVIOUSLY EVALUATED IN THE SAR, CREATE THE POSSIBILITY OF AN ACCIDENT OR MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY OF A DIFFERENT TYPE THAN ANY PREVIOUSLY EVALUATED IN THE SAR, AND WILL NOT REDUCE THE MARGIN OF SAFETY AS DEFINED IN THE BASIS FOR ANY TECHNICAL SPECIFICATION.

II. Safety Evaluation

[19] ☐ Not applicable per Safety Evaluation /Pre-screening/Applicability Review

A. Technical Specifications [20]

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 THE SAR IS TO ELIMINATE A REDUNDANT REVIEW OF APPROVED DESIGN CHANGES DURING THE IMPLEMENTATION AUTHORIZATION PROCESS BY THE RADIATION PROTECTION MANAGER. THE TECHNICAL SPECIFICATIONS DO NOT ADDRESS THE ALARA PROGRAM OR ITS IMPLEMENTATION PROCESSES AND REQUIREMENTS. THE TECHNICAL SPECIFICATIONS DO REQUIRE GGNS GENERAL MANAGER AUTHORIZATION FOR IMPLEMENTATION OF NUCLEAR SAFETY RELATED DESIGN CHANGES, BUT DO NOT DISCUSS OTHER REVIEW OR APPROVAL REQUIREMENTS FOR CHANGES. THEREFORE, A CHANGE TO THE TECHNICAL SPECIFICATIONS IS NOT REQUIRED.*

B. Unreviewed Safety Question [21]

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 TO THE SAR IS TO ELIMINATE A REDUNDANT REVIEW BY THE RADIATION PROTECTION MANAGER OF APPROVED (ENGINEERING AND PLANT STAFF) DESIGN CHANGES DURING THE CHANGE IMPLEMENTATION AUTHORIZATION PROCESS. THE PROPOSED CHANGE TO THE SAR IS TO AN ADMINISTRATIVE PROCESS THAT HAS NO DIRECT OR INDIRECT IMPACT ON THE ACCIDENT ANALYSES IN THE SAR, AND THEREFORE DOES NOT AFFECT IN ANY WAY THE PROBABILITY OF OCCURRENCE OF ANY ACCIDENT PREVIOUSLY EVALUATED IN THE SAR.

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

☐ Yes
☑ No

BASIS: *THE PROPOSED CHANGE TO THE SAR IS TO ELIMINATE A REDUNDANT REVIEW BY THE RADIATION PROTECTION MANAGER OF APPROVED (ENGINEERING AND PLANT STAFF) DESIGN CHANGES DURING THE CHANGE IMPLEMENTATION AUTHORIZATION PROCESS. THE PROPOSED CHANGE TO THE SAR IS TO AN ADMINISTRATIVE PROCESS. THE ALARA PROGRAM REVIEW IS INTENDED TO MAINTAIN RADIATION EXPOSURES DURING NORMAL PLANT OPERATION AS LOW AS REASONABLY ACHIEVABLE, AND HAS NO IMPACT ON DOSE CONSEQUENCES DURING ACCIDENTS 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: *THE PROPOSED CHANGE TO THE SAR IS TO ELIMINATE A REDUNDANT REVIEW BY THE RADIATION PROTECTION MANAGER OF APPROVED (ENGINEERING AND PLANT STAFF) DESIGN CHANGES, DURING THE CHANGE IMPLEMENTATION AUTHORIZATION PROCESS. THE PROPOSED CHANGE TO THE SAR IS TO AN ADMINISTRATIVE PROCESS THAT HAS NO DIRECT OR INDIRECT IMPACT ON THE OPERATION, CONTROL OR FUNCTION OF EQUIPMENT IMPORTANT TO SAFETY. THEREFORE, THE PROPOSED CHANGE CANNOT INCREASE THE PROBABILITY OF OCCURRENCE OF A MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY PREVIOUSLY EVALUATED IN THE SAR.*

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

☐ Yes
☑ No

BASIS: *THE PROPOSED CHANGE TO THE SAR IS TO ELIMINATE A REDUNDANT REVIEW BY THE RADIATION PROTECTION MANAGER OF APPROVED (ENGINEERING AND PLANT STAFF) DESIGN CHANGES, DURING THE CHANGE IMPLEMENTATION AUTHORIZATION PROCESS. THE PROPOSED CHANGE TO THE SAR IS TO AN ADMINISTRATIVE PROCESS THAT HAS NO DIRECT OR INDIRECT IMPACT ON THE OPERATION, CONTROL OR FUNCTION OF EQUIPMENT IMPORTANT TO SAFETY. THEREFORE, THE PROPOSED CHANGE CANNOT INCREASE THE CONSEQUENCES OF A MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY PREVIOUSLY EVALUATED IN THE SAR.*

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

☐ Yes
☑ No

BASIS: *THE PROPOSED CHANGE TO THE SAR IS TO ELIMINATE A REDUNDANT REVIEW BY THE RADIATION PROTECTION MANAGER OF APPROVED (ENGINEERING AND PLANT STAFF) DESIGN CHANGES, DURING THE CHANGE IMPLEMENTATION AUTHORIZATION PROCESS. THE PROPOSED CHANGE TO THE SAR IS TO AN ADMINISTRATIVE PROCESS THAT HAS NO DIRECT OR INDIRECT IMPACT ON THE OPERATION, CONTROL OR FUNCTION OF ANY GGNS PLANT EQUIPMENT. THEREFORE, THE PROPOSED CHANGE CANNOT CREATE 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: *THE PROPOSED CHANGE TO THE SAR IS TO ELIMINATE A REDUNDANT REVIEW BY THE RADIATION PROTECTION MANAGER OF APPROVED (ENGINEERING AND PLANT STAFF) DESIGN CHANGES, DURING THE CHANGE IMPLEMENTATION AUTHORIZATION PROCESS. THE PROPOSED CHANGE TO THE SAR IS TO AN ADMINISTRATIVE PROCESS THAT HAS NO DIRECT OR INDIRECT IMPACT ON THE OPERATION, CONTROL OR FUNCTION OF ANY GGNS PLANT EQUIPMENT. THEREFORE, THE PROPOSED CHANGE CANNOT CREATE THE POSSIBILITY FOR A MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY 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: *THE PROPOSED CHANGE TO THE SAR IS TO ELIMINATE A REDUNDANT REVIEW BY THE RADIATION PROTECTION MANAGER OF APPROVED (ENGINEERING AND PLANT STAFF) DESIGN CHANGES, DURING THE CHANGE IMPLEMENTATION AUTHORIZATION PROCESS. THE PROPOSED CHANGE TO THE SAR IS TO AN ADMINISTRATIVE PROCESS THAT HAS NO DIRECT OR INDIRECT IMPACT ON THE OPERATION, CONTROL OR FUNCTION OF ANY GGNS PLANT EQUIPMENT. THEREFORE, THE PROPOSED CHANGE CANNOT REDUCE THE MARGIN OF SAFETY AS DEFINED IN THE BASIS FOR ANY TECHNICAL SPECIFICATION.*

III. Environmental Evaluation[22]

☐ Not applicable per Prescreening/ Environmental Evaluation Applicability Review

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

A. Environmental Protection Plan [23]

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

BASIS: *THE PROPOSED CHANGE TO THE SAR DEALS WITH THE REVIEW OF DESIGN CHANGES AND MODIFICATIONS BY THE RADIATION PROTECTION MANAGER FOR ALARA CONSIDERATIONS, AND IS UNRELATED TO THE ENVIRONMENTAL PROTECTION PLAN OR THE FINAL ENVIRONMENTAL STATEMENT. THE PROPOSED CHANGE IN NO WAY AFFECTS OR HAS THE POTENTIAL TO AFFECT THE ENVIRONMENT.*

B. Unreviewed Environmental Question [24]

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 TO THE SAR DEALS WITH THE REVIEW OF DESIGN CHANGES AND MODIFICATIONS BY THE RADIATION PROTECTION MANAGER FOR ALARA CONSIDERATIONS, AND IS UNRELATED TO THE ENVIRONMENTAL PROTECTION PLAN OR THE FINAL ENVIRONMENTAL STATEMENT. THE PROPOSED CHANGE IN NO WAY AFFECTS OR HAS THE POTENTIAL TO AFFECT THE ENVIRONMENT.*

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

BASIS: *THE PROPOSED CHANGE TO THE SAR DEALS WITH THE REVIEW OF DESIGN CHANGES AND MODIFICATIONS BY THE RADIATION PROTECTION MANAGER FOR ALARA CONSIDERATIONS, AND IS UNRELATED TO THE REACTOR OPERATIONS OR CONTROLS AND CANNOT HAVE ANY AFFECT ON EFFLUENT LEVELS FROM THE PLANT.*

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

BASIS: *THE PROPOSED CHANGE TO THE SAR DEALS WITH THE REVIEW OF DESIGN CHANGES AND MODIFICATIONS BY THE RADIATION PROTECTION MANAGER FOR ALARA CONSIDERATIONS, AND IS UNRELATED TO THE ENVIRONMENTAL PROTECTION PLAN OR THE FINAL ENVIRONMENTAL STATEMENT. THE PROPOSED CHANGE IN NO WAY AFFECTS OR HAS THE POTENTIAL TO AFFECT THE ENVIRONMENT.*

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

I. Safety Evaluation Overview

Page __ of __

A. Reference Data

ORIGINATOR: BRIAN D. PATRICK DEPT/SECT: HP / DOSIMETRY EVAL #: 96-0069-R00

DOCUMENT EVALUATED: UFSAR SECT. 12 (SEE ATTACHED)

SYSTEM NO:(S) N/A

REFERENCES:

FSAR CHANGE REQUIRED? X Yes ☐ No CR # 96-091

FSAR SECTIONS TO BE REVISED: 12.1.3.1.L, 12.5.2.1.1, 12.5.2.2.3, TABLE 12.5-1

TRM CHANGE REQUIRED? (8) ☐ Yes X No

TECH. SPEC. CHANGE REQUIRED? (9) ☐ Yes X 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 X No

EXPLAIN:

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.)

Signatures and Approvals of Attached Safety and Environmental Evaluation

Evaluated: (12) Brian D. Patrick / ORIGINATOR / [signature] / 8-22-96
(PRINT NAME) ORIGINATOR (SIGNATURE) DATE

Reviewed: (13) Brett D Roberson / INDEPENDENT REVIEWER / [signature] / 8/22/96
(PRINT NAME) INDEPENDENT REVIEWER (SIGNATURE) DATE

Reviewed: (14) N/A /
(PRINT NAME) OTHER REVIEWER (IF REQUIRED) (SIGNATURE) DATE

Plant Safety Review Committee Review

(15)

[signature] 9/10/96
CHAIRMAN, PSRC DATE

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

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT

INCORPORATES THE USE OF ELECTRONIC ALARMING DOSIMETERS (EADS) INTO THE UFSAR.

REASON FOR CHANGE, TEST OR EXPERIMENT

ALLOWS THE USE OF EADS AS MONITORING DEVICES FOR PERSONNEL

SAFETY EVALUATION SUMMARY AND CONCLUSIONS SEE ATTACHED ACCEPTANCE TESTING REPORT AND ELECTRONIC DOSIMETER / TLD COMPARISON REPORTS

II. Safety Evaluation

(19) ☐ Safety Evaluation not required per Completed Safety and Environmental Pre-Screening or Applicability Review. Proceed to Section III for Environmental Evaluation.

A. Technical Specifications (20)

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

BASIS: TECH SPECIFICATION NUMBER 5.7.1 (SEE ATTACHED) REQUIRES THAT ANY INDIVIDUAL OR GROUP PERMITTED TO ENTER A HIGH RADIATION AREA SHALL BE PROVIDED WITH ONE OR MORE OF THE FOLLOWING:

• A RADIATION MONITORING DEVICE THAT CONTINUOUSLY INDICATES THE RADIATION DOSE RATE IN THE AREA

• A RADIATION MONITORING DEVICE THAT CONTINUOUSLY INTEGRATES THE RADIATION DOSE RATE IN THE AREA AND ALARMS WHEN A PRESET INTEGRATED DOSE IS RECEIVED...

ISSUING ALL WORKERS THAT ENTER RADIOLOGICALLY POSTED AREAS EADS WILL SATISFY THIS REQUIREMENT BECAUSE IT INTEGRATES DOSE AND ALARMS WHEN A PRESET LIMIT IS RECEIVED.

B. Unreviewed Safety Question (21)

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 X No

BASIS: THE USE OF EADS TO MONITOR PERSONNEL WORKING INSIDE RADIOLOGICALLY POSTED AREAS WILL NOT EFFECT ANY SAFETY RELATED EQUIPMENT OR OTHERWISE INCREASE THE PROBABILITY OF AN ACCIDENT THAT HAS BEEN EVALUATED IN THE SAR. THE EAD SYSTEM WORKS INDEPENDENTLY OF ANY OTHER EXISTING SYSTEM. IT WILL NOT AFFECT THE OPERATION OR FUNCTION OF ANY OTHER SYSTEM, SAFETY OR OTHERWISE.

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

   BASIS: THE USE OF EADS WILL NOT INCREASE ANY CONSEQUENCES. THEY WILL SIMPLY BE AN ENHANCEMENT, SHOULD AN ACCIDENT OCCUR, OVER THE USE OF POCKET ION CHAMBERS BECAUSE OF THEIR ABILITY TO GIVE DOSE AND RATES ALTERNATELY AND ALARM IF EITHER REACHES A PRESET LIMIT.

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

   BASIS: THE USE OF EADS WILL NOT EFFECT ANY SAFETY RELATED EQUIPMENT. THE EAD SYSTEM WILL OPERATE INDEPENDENTLY OF ANY EXISTING SYSTEM, SAFETY OR OTHERWISE.

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

   BASIS : THE USE OF EADS WILL HAVE NO EFFECT ON THE CONSEQUENCES OF A MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY. IT WILL BE AN ENHANCEMENT OVER THE USE OF POCKET ION CHAMBERS DUE TO THEIR ABILITY TO DISPLAY DOSE AND RATE AND ALARM WHEN A PREDETERMINED LIMIT IS EXCEEDED.

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

   BASIS: THE USE OF EADS AS PERSONNEL MONITORING DEVICES INSTEAD OF POCKET ION CHAMBERS WILL NOT CREATE ANY POSSIBILITY FOR AN ACCIDENT OF A DIFFERENT TYPE THAN ANY PREVIOUSLY EVALUATED IN THE SAR DUE TO THE FACT THAT IT OPERATES INDEPENDENTLY OF ALL OTHER SYSTEMS.

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 X No

   BASIS: EADS WILL NOT EFFECT EQUIPMENT IMPORTANT TO SAFETY OF A DIFFERENT TYPE THAN ANY PREVIOUSLY EVALUATED IN THE SAR DUE ITS INDEPENDENT OPERATION.

7. Will reduce the margin of safety as defined in the BASIS: for any Technical Specification. ☐ Yes X No

   BASIS: THE USE OF EADS WILL NOT REDUCE THE MARGIN OF SAFETY DEFINED IN THE BASIS FOR ANY TECHNICAL SPECIFICATIONS. IT WILL BE AN ENHANCEMENT FOR TECH SPEC 5.7.1 (SEE ATTACHED) DUE TO ITS ABILITY TO DISPLAY DOSE AND RATE WITH A SINGLE INSTRUMENT AND THEREFORE WILL INCREASE THE MARGIN OF SAFETY.

III. Environmental Evaluation[22]

☐ Environmental Evaluation not required per Completed Safety and Environmental Pre-Screening or Applicability Review.

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

A. Environmental Protection Plan [23]

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

   BASIS: **THE USE OF EADS TO MONITOR PERSONNEL RADIATION DOSE WILL HAVE NO EFFECT ON THE ENVIRONMENTAL PROTECTION PLAN.**

B. Unreviewed Environmental Question [24]

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 X No

   BASIS: **THE USE OF EADS TO MONITOR PERSONNEL FOR RADIATION DOSE WILL NOT CONCERN A MATTER WHICH WILL HAVE AN ADVERSE ENVIRONMENTAL IMPACT PREVIOUSLY EVALUATED. EADS ARE USED EXCLUSIVELY FOR MONITORING PERSONNEL FOR EXTERNAL RADIATION.**

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

   BASIS: **THE USE OF EADS FOR PERSONNEL MONITORING WILL HAVE NO EFFECT ON EFFLUENT DISCHARGES OR POWER LEVEL. IT IS AN INDEPENDENT SYSTEM.**

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

   BASIS: **THE USE OF EADS DOES NOT CONCERN ANY MATTERS OF ENVIRONMENTAL SIGNIFICANCE WHICH WERE NOT ADDRESSED ABOVE.**

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

I. Safety Evaluation Overview

Page __ of __

A. Reference Data

ORIGINATOR: BRIAN D. PATRICK DEPT/SECT: HP / DOSIMETRY EVAL. #: 96-0069-R~~00~~ 01 9-11-96

DOCUMENT EVALUATED: UFSAR SECT. 12 (SEE ATTACHED)

SYSTEM NO:(S) N/A

REFERENCES:

FSAR CHANGE REQUIRED? X Yes ☐ No CR # 96-091

FSAR SECTIONS TO BE REVISED: TABLE 12.5-1

TRM CHANGE REQUIRED? (8) ☐ Yes X No

TECH. SPEC. CHANGE REQUIRED? (9) ☐ Yes X 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 X No

EXPLAIN:

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.)

Signatures and Approvals of Attached Safety and Environmental Evaluation

Evaluated: (12) Brian D. Patrick / ORIGINATOR [signature] / 8-22-96

(PRINT NAME) ORIGINATOR (SIGNATURE) DATE

Reviewed: (13) S. JOHNSON / INDEPENDENT REVIEWER SJohnson / 9-10-96

(PRINT NAME) INDEPENDENT REVIEWER (SIGNATURE) DATE

Reviewed: (14) N/A /

(PRINT NAME) OTHER REVIEWER (IF REQUIRED) (SIGNATURE) DATE

Plant Safety Review Committee Review

(15) [signature] 9-12-96

CHAIRMAN, PSRC DATE

SE No. 96-0069-R01 9-11-96
Page ____ of ____

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

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT

EVALUATES THE USE OF ELECTRONIC ALARMING DOSIMETERS (EADS). MODIFYS TABLE 12.5-1 SEE ATTACHED

REASON FOR CHANGE, TEST OR EXPERIMENT

ALLOWS THE USE OF EADS AS MONITORING DEVICES FOR PERSONNEL

SAFETY EVALUATION SUMMARY AND CONCLUSIONS SEE ATTACHED ACCEPTANCE TESTING REPORT AND ELECTRONIC DOSIMETER / TLD COMPARISON REPORTS

II. Safety Evaluation

(19) ☐ Safety Evaluation not required per Completed Safety and Environmental Pre-Screening or Applicability Review. Proceed to Section III for Environmental Evaluation.

A. Technical Specifications (20)

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

BASIS: **TECH SPECIFICATION NUMBER 5.7.1 (SEE ATTACHED) REQUIRES THAT ANY INDIVIDUAL OR GROUP PERMITTED TO ENTER A HIGH RADIATION AREA SHALL BE PROVIDED WITH ONE OR MORE OF THE FOLLOWING:**

- **A RADIATION MONITORING DEVICE THAT CONTINUOUSLY INDICATES THE RADIATION DOSE RATE IN THE AREA**
- **A RADIATION MONITORING DEVICE THAT CONTINUOUSLY INTEGRATES THE RADIATION DOSE RATE IN THE AREA AND ALARMS WHEN A PRESET INTEGRATED DOSE IS RECEIVED...**

ISSUING ALL WORKERS THAT ENTER RADIOLOGICALLY POSTED AREAS EADS WILL SATISFY THIS REQUIREMENT BECAUSE IT INTEGRATES DOSE AND ALARMS WHEN A PRESET LIMIT IS RECEIVED.

B. Unreviewed Safety Question (21)

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 X No

BASIS: **THE USE OF EADS TO MONITOR PERSONNEL WORKING INSIDE RADIOLOGICALLY POSTED AREAS WILL NOT EFFECT ANY SAFETY RELATED EQUIPMENT OR OTHERWISE INCREASE THE PROBABILITY OF AN ACCIDENT THAT HAS BEEN EVALUATED IN THE SAR. THE EAD SYSTEM WORKS INDEPENDENTLY OF ANY OTHER EXISTING SYSTEM. IT WILL NOT AFFECT THE OPERATION OR FUNCTION OF ANY OTHER SYSTEM, SAFETY OR OTHERWISE.**

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

   BASIS: THE USE OF EADS WILL NOT INCREASE ANY CONSEQUENCES. THEY WILL SIMPLY BE AN ENHANCEMENT, SHOULD AN ACCIDENT OCCUR, OVER THE USE OF POCKET ION CHAMBERS BECAUSE OF THEIR ABILITY TO GIVE DOSE AND RATES ALTERNATELY AND ALARM IF EITHER REACHES A PRESET LIMIT.

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

   BASIS: THE USE OF EADS WILL NOT EFFECT ANY SAFETY RELATED EQUIPMENT. THE EAD SYSTEM WILL OPERATE INDEPENDENTLY OF ANY EXISTING SYSTEM, SAFETY OR OTHERWISE.

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

   BASIS : THE USE OF EADS WILL HAVE NO EFFECT ON THE CONSEQUENCES OF A MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY. IT WILL BE AN ENHANCEMENT OVER THE USE OF POCKET ION CHAMBERS DUE TO THEIR ABILITY TO DISPLAY DOSE AND RATE AND ALARM WHEN A PREDETERMINED LIMIT IS EXCEEDED.

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

   BASIS: THE USE OF EADS AS PERSONNEL MONITORING DEVICES INSTEAD OF POCKET ION CHAMBERS WILL NOT CREATE ANY POSSIBILITY FOR AN ACCIDENT OF A DIFFERENT TYPE THAN ANY PREVIOUSLY EVALUATED IN THE SAR DUE TO THE FACT THAT IT OPERATES INDEPENDENTLY OF ALL OTHER SYSTEMS.

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 X No

   BASIS: EADS WILL NOT EFFECT EQUIPMENT IMPORTANT TO SAFETY OF A DIFFERENT TYPE THAN ANY PREVIOUSLY EVALUATED IN THE SAR DUE ITS INDEPENDENT OPERATION.

7. Will reduce the margin of safety as defined in the BASIS: for any Technical Specification. ☐ Yes X No

   BASIS: THE USE OF EADS WILL NOT REDUCE THE MARGIN OF SAFETY DEFINED IN THE BASIS FOR ANY TECHNICAL SPECIFICATIONS. IT WILL BE AN ENHANCEMENT FOR TECH SPEC 5.7.1 (SEE ATTACHED) DUE TO ITS ABILITY TO DISPLAY DOSE AND RATE WITH A SINGLE INSTRUMENT AND THEREFORE WILL INCREASE THE MARGIN OF SAFETY.

III. Environmental Evaluation[22]

☐ Environmental Evaluation not required per Completed Safety and Environmental Pre-Screening or Applicability Review.

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

A. Environmental Protection Plan [23]

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

   BASIS: **THE USE OF EADS TO MONITOR PERSONNEL RADIATION DOSE WILL HAVE NO EFFECT ON THE ENVIRONMENTAL PROTECTION PLAN.**

B. Unreviewed Environmental Question [24]

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 X No

   BASIS: **THE USE OF EADS TO MONITOR PERSONNEL FOR RADIATION DOSE WILL NOT CONCERN A MATTER WHICH WILL HAVE AN ADVERSE ENVIRONMENTAL IMPACT PREVIOUSLY EVALUATED. EADS ARE USED EXCLUSIVELY FOR MONITORING PERSONNEL FOR EXTERNAL RADIATION.**

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

   BASIS: **THE USE OF EADS FOR PERSONNEL MONITORING WILL HAVE NO EFFECT ON EFFLUENT DISCHARGES OR POWER LEVEL. IT IS AN INDEPENDENT SYSTEM.**

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

   BASIS: **THE USE OF EADS DOES NOT CONCERN ANY MATTERS OF ENVIRONMENTAL SIGNIFICANCE WHICH WERE NOT ADDRESSED ABOVE.**

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RELATED DOCUMENT N/A

96-076-HP

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

I. Safety Evaluation Overview

A. Reference Data

ORIGINATOR: *MARK S. MILLER* DEPT/SECT: *HEALTH PHYSICS* EVAL. #: *96-0071-R00*

DOCUMENT EVALUATED: *QDR 0144-96*

SYSTEM NO:(S) *N/A* (INSERT N/A IF NOT APPLICABLE)

REFERENCES: *N/A*

FSAR CHANGE REQUIRED? ☒ Yes ☐ No *CR# 96-100*

FSAR SECTIONS TO BE REVISED: *12.2.1.6*

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

EXPLAIN: *THERE ARE NO PER-REQUIRED CHANGES TO PROCEDURES, TECH SPECS, OPERATING DOCUMENTS, OR ANY OPERATING CONDITIONS.*

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

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

Signatures and Approvals of Attached Safety and Environmental Evaluation

Evaluated: M. S. Miller, (signature), 9-18-96
(PRINT NAME) ORIGINATOR (SIGNATURE) DATE

Reviewed: Michael J. Larson, (signature), 9-18-96
(PRINT NAME) INDEPENDENT REVIEWER (SIGNATURE) DATE

Reviewed: NA / /
(PRINT NAME) OTHER REVIEWER (IF REQUIRED) (SIGNATURE) DATE

Plant Safety Review Committee Review

(signature) 9/19/96
CHAIRMAN, PSRC DATE

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

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT

Storage of radioactive material in areas outside the controlled access area, protected area, and plant structures. These areas include the warehouses, onsite working area buildings, storage areas outside of plant structures but inside the protected area, Unit II, and storage buildings for radioactive components such as the High Pressure and Low Pressure Turbines. This evaluation also includes a short duration holding area for radioactive waste in transition from processing to transporting offsite.

REASON FOR CHANGE, TEST OR EXPERIMENT

This material may still have useful life and is not being discarded, or is awaiting final disposition. In the case of radioactive waste, it is in a transitory period between processing and shipping, and is normally packaged in its final shipping containers. The normal duration the waste spends in the holding area outside of the protected area is less than one week, with the longest expected duration of about two weeks. The exception to this is the sewage sludge which is being held for decay and subsequent environmental disposition, and is below any radiological posting limits within the site boundary. The normal duration the waste spends inside the protected area holding area is less than one cycle.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS

Section 12.2.1.6 of the SAR describes the storage of material outside of plant structures, and states that adequate protective measures are to be taken by health physics if material is to be stored outside of plant structures. SAR Section 11.4.7 describes the use of shipping containers and vehicles outside plant structures but inside the restricted area (protected area). The restricted area at Grand Gulf is defined in several license basis documents, including Technical Specifications and the SAR. For the purposes of radiation protection, the restricted area is the same as the protected area, as clearly outlined in Section 2.1.1.3 of the SAR. For effluents and other purposes it is clearly outlined as the site boundary in Technical Specification's Offsite Dose Calculations Manual (ODCM) Appendix A 1.6. Furthermore, 10 CFR 20.1003 defines a controlled area as being outside of a restricted area but inside the site boundary, with controlled access by the licensee. The storage areas outside the protected area are maintained as controlled areas, enclosed as necessary, locked, and under control of Health Physics. All radioactive material storage areas outside of plant structures have been assigned a survey frequency, and area TLDs are used for each storage area to ensure the requirements of 10 CFR 20.1301 are met. The same administrative, radiation protection, and section procedures that are used in the controlled access area will also be used in the protected area and controlled areas outside of plant structures.

GL80051 and GL81038 gives definitions and guidelines for radioactive waste storage facilities. The high pressure turbine rotors and other radioactive materials in storage are not waste since they are not being discarded, but may be given the same considerations as such for the purposes of radioactive material control and regulatory compliance.

10 CFR 20.1301(a) describes the dose and dose rate limits to members of the public. SAR section 2.1.2.1.1 paragraph 3 states that Entergy Operations will allow access to parts of the site property outside of the exclusion area. 10 CFR 20.1301(b) states that if members of the public are permitted access to controlled areas, the limits of 10 CFR 1301(a) still apply. Since these controlled areas are enclosed and locked under the control of Health Physics, public access is controlled. The dose rates at the controlled area fence boundary are below the limits of 10 CFR 20.1301(a), thereby ensuring the dose limits at the site boundary.

10 CFR 50.75 gives requirements for records of information important to decommissioning of the plant. Surveys of these areas are currently performed on a monthly basis, and retained for use for plant decommissioning. This is currently covered in section procedure 08-S-01-11, Health Physics Document Handling and Control.

It is concluded that the storage of radioactive materials is acceptable outside plant structures and inside Grand Gulf's site boundary provided that it is in a controlled area or inside the protected area.

II. Safety Evaluation

☐ Safety Evaluation not required per Completed Safety and Environmental Pre-Screening or Applicability Review. Proceed to Section III for Environmental Evaluation.

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 storage of radioactive material outside of the controlled access area or protected area is not discussed in the Grand Gulf Tech Specs section. However, the requirements of Tech Spec sections 5.5.4 and 5.7 still apply. No change in the Tech Spec program would be required as a result of storage of radioactive material outside the controlled access area but within the site boundary.*

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: *Section 12.2.1.6 of the SAR discusses stored radioactive materials outside the plant. Section 11.4.7 discusses the use of shipping containers within the restricted area (protected area) but outside plant structures. This would include the area outside of the Radioactive Waste Building truck bay, and other areas inside the protected area.*

Radioactive waste being held inside the protected area but outside plant structures is normally in shipping or suitable storage containers that meet the guidelines of GL80051 and 81038.

Radioactive waste in the radioactive waste holding area outside of the protected area is held temporarily pending shipment, usually within two weeks. It is in final shipping containers and readied for shipment before leaving the protected area, conforms to DOT and NRC specifications, thereby preventing any unmonitored release, the release of radioactivity above the limits of Tech Spec 5.5.4, and ensuring the requirements of 10 CFR 20.1301 are met.

The high pressure turbine rotors and other radioactive materials in storage are not waste since they are not being discarded, but may be given the same considerations as discussed in GL80051 and 81038 for the purposes of radioactive material control and regulatory compliance. The radioactive materials are packaged, housed, or enclosed to reduce the effects of weathering and provide containment of any contaminants, provide material controls, and prevent premature decay of protective measures. For areas outside the protected area, a controlled area will be established to situate the proper boundary and ensure the limits of 10 CFR 20.1301 are met. Consequently, the storage of radioactive material does not affect or increase the probability of occurrence of any accident previously evaluated in the SAR.

10 CFR 50.75 gives requirements for records of information important to decommissioning of the plant. Surveys of these areas are currently performed on a monthly basis, and retained for use for plant decommissioning. This is currently covered in section procedure 08-S-01-11, Health Physics Document Handling and Control.

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

☐ Yes
☒ No

BASIS: *The radioactive material for storage consists of material in solid form and in low levels of radioactivity mostly from fixed contamination. The radioactive materials are packaged, housed, or enclosed to reduce the effects of weathering and provide containment of any contaminants, material controls, and prevent premature decay of protective measures outlined in GL80051 and 81038. For the radioactive material storage areas outside the protected area but within the site boundary, a controlled area will be established, such as a building, room, or fencing with locked gates. This allows access to be controlled by Health Physics, proper situation of the boundary, and ensures the limits of 10 CFR 20.1301 are met. As a result, no adverse affects from its storage will occur, and will have no effect or 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: *There are no system components or equipment important to safety used in the storage of radioactive material outside of controlled access areas and protected areas, nor are any affected by this action. Therefore, there is no increase in the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.*

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

☐ Yes
☒ No

BASIS: *The storage of radioactive materials outside of controlled access areas and protected areas is outside of plant structures, equipment, and systems, and does not affect their operation in any way. Therefore, there is no increase in the consequences of a malfunction of a component or equipment important to safety.*

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

☐ Yes
☒ No

BASIS: *Section 2.4.13.3 of the SAR discusses the accident effects in the event of a spill or release of liquid radioactive material. The radioactive materials are packaged, housed, or enclosed to reduce the effects of weathering and provide containment of any contaminants, provide material controls, and prevent premature decay of protective measures. For areas outside the protected area, a controlled area will be established to situate the proper boundary and ensure the limits of 10 CFR 20.1301 are met. The radioactive waste holding area is not a storage area, and the waste is appropriately packaged for shipment conforming to DOT and NRC specifications, thereby preventing any unmonitored release, the release of radioactivity above the limits of Tech Spec 5.5.4, and ensuring the requirements of 10 CFR 20.1301 are met. Hence, the storage of radioactive material does not create 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: *For the storage of radioactive material or the holding of radioactive waste, no equipment important to safety will be used or affected in any way. As a result of storing radioactive material or holding radioactive waste, no equipment important to safety is used or affected. Accordingly, there is no creation of a possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in SAR.*

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

   BASIS: *Tech Spec 5.5.4 details the limitations and requirements of radioactive effluents and maintaining dose to the public ALARA. The stored radioactive material was decontaminated before leaving the controlled access area to remove as much activity as possible and is packaged, housed, or enclosed to provide containment of any contaminants. Therefore there is no reduction in the margin of safety as defined in the BASIS for any Technical Specification*

III. Environmental Evaluation

☐ Environmental Evaluation not required per Completed Safety and Environmental Pre-Screening or 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: *Section 2.4.13.3 of the SAR addresses accidental releases of liquid radioactive material. Liquid radioactive material is not present in significant amounts in any of the components or items being held or stored. The radioactive materials are further protected from rain and incidental water or liquids by being suitably packaged, housed, or enclosed. This also reduces the effects of weathering and provides containment of any contaminants, provides material controls, and prevents premature decay of protective measures.*

   Technical Specification's Offsite Dose Calculations Manual (ODCM) Appendix A 1.6 establishes the restricted area for effluents as the site boundary. The radioactive material storage locations outside of the protected area are within the site boundary and enclosed in a controlled area (eg, building, room, or fencing with locked gates) with access controlled by Health Physics. The dose rates at the controlled area border are below the limits listed in 10 CFR 20.1301 for dose to the public, and are further ensured by periodic survey and area TLD readings. Therefore, storage of radioactive material outside of the controlled access area or protected area 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: *Section 5.9.1.1.2(2) of the FES states that calculations show the cumulative dose to the exposed population from radiation fields from the reactor and its associated components would be insignificant when compared to natural background dose, less than 1 person-rem/year. It also states that low-level radioactive material storage containers will contribute an estimated 0.1% of the exposure from nitrogen-16 at the site boundary. Section 2.6.1 of the ODCM requires determination of cumulative dose to members of the public from direct radiation, and requires this be made by our environmental TLDs which are located near the property line in 8 of 16 meteorological sectors. The placement of these containers is such that the radiological contribution at the site boundary is negligible and will not significantly increase any radiation levels already in existence, and will be further ensured by the readings taken from the environmental TLDs.*

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

   ☐ Yes
   ☒ No

   BASIS: *The radioactive materials being stored or held do not contain any liquid radioactive material above trace amounts which would have a negligible effect in effluents. These materials are stored within the site boundary. Therefore, there are no effluents or power level changes associated with the storage of these materials.*

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

   ☐ Yes
   ☒ No

   BASIS: *The storage of radioactive material within the site boundary is addressed in FES Section 5.9.1.1.2(2), and does not concern a matter not previously evaluated in III.B.1 of this form.*

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NUMBER of PAGES 7
DATE 10/7/96
RELATED DOCUMENT NUMBER N/A

96-075-NSRH

BSF 10/2/96

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

I. Safety Evaluation Overview

A. Reference Data

ORIGINATOR: Bryan Ford   dept/sect: NS&RA   EVAL. #: 96-00085-R00

DOCUMENT EVALUATED: TRM/UFSAR Appendix 16B Change - Reduce surveillance frequencies for refuel platform, auxiliary platform and fuel handling platform for LCOs 6.9.3 & 6.9.5.

SYSTEM NO: F11

REFERENCES: GGNS - Technical Specifications updated through amendment #122, UFSAR Rev. 9, Ch. 7, 9, 15 NUREG-0612, NEDO 31466.

FSAR CHANGE REQUIRED? ■ Yes ☐ No CR# 96-111

FSAR SECTIONS TO BE REVISED: CHAPTER 16, APPENDIX 16B

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

EXPLAIN: N/A

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.)

Signatures and Approvals of Attached Safety and Environmental Evaluation

Evaluated: Bryan Ford (PRINT NAME) ORIGINATOR Bryan Ford (SIGNATURE) 9/24/96 DATE

Reviewed: BILLY PARMAN (PRINT NAME) INDEPENDENT REVIEWER Billy Parman (SIGNATURE) 9/24/96 DATE

Reviewed: N/A (PRINT NAME) OTHER REVIEWER (IF REQUIRED) (SIGNATURE) DATE

Plant Safety Review Committee Review

[signature] CHAIRMAN, PSRC 10/3/96 DATE

B. Executive Summary

Brief Description of Change, Test or Experiment:

The surveillance test frequency of interlock functions is being reduced to eliminate unnecessary testing. These interlocks have a reliable operating history. The proposed change will continue to provide the intended equipment protection currently provided by the interlocks. The proposed change in surveillance frequencies is listed below:

Refueling Platform:

For TRM/UFSAR Appendix 16B surveillances SR 6.9.3.3 through 6.9.3.5, the frequency will be changed from "within 7 days prior to handling fuel assemblies or control rods" to "31 days".

Fuel Handling Platform:

For TRM/UFSAR Appendix 16B surveillance SR 6.9.5.1 the frequency will be changed from "Within 2 hours of starting hoist operation AND 12 hours" to "12 hours".

For TRM/UFSAR Appendix 16B surveillances SR 6.9.5.4 through 6.9.5.6 the frequency will be changed from "within 7 days prior to handling fuel assemblies or control rods" to "31 days."

There is no documented technical basis for the current surveillance frequencies of these interlocks.

Reason for Change, Test or Experiment

Reduce surveillance test frequency of the refueling interlocks listed above to eliminate the unnecessary testing.

Safety Evaluation Summary and Conclusion

The interlocks affected by this change primarily protect the refueling equipment, reactor vessel internals, fuel storage racks or fuel assemblies during refueling operations. Some interlock functions apply to both the Refueling Platform (TRM/UFSAR Appendix 16B 6.9.3) and the Fuel Handling Platform (TRM/UFSAR Appendix 16B 6.9.5). The interlock functions with a brief description and the associated Surveillance Requirement (SR) references are listed below:

Jam Cutoff Interlock: Protects reactor vessel internals and fuel storage racks from excessive lifting force should they become inadvertently engaged during lifting operation and fuel assemblies from excessive force should they become stuck. (SR 6.9.3.3 and SR 6.9.5.4)

Primary & Redundant Overload Interlock: Protects the hoists and limits the loads carried by the hoists by assuring that loads in excess of their of the expected ranges in weight are not moved by the hoists. (SR 6.9.3.4, SR 6.9.5.5, and SR 6.9.5.6)

Downtravel Cutoff Interlock: Protects the main hoist and the reactor internals by assuring that the hoist does not reach the top guide and that the hoist cable ends are not pulled loose from the drum. (SR 6.9.3.5)

Monorail Auxiliary Hoist Load Override: Protects the monorail auxiliary hoist by preventing lifting of loads above its design capacity. (SR 6.9.5.1)

Accident Analyses

The Fuel Handling Accidents (FHA) in the containment and auxiliary building were evaluated for the effect of proposed changes in the surveillance frequencies of refueling equipment. The failure of any of these interlock functions is not an event initiator or mitigator in the postulated FHAs discussed in UFSAR Sections 15.7.4 and 15.7.6 or in any of the reactivity events discussed in UFSAR Section 15.4.1.1.

These interlocks, except for the Jam Cutoff Interlock, are credited in the analyses presented in UFSAR Appendix 9D, "GGNS Compliance with NUREG-0612, Control of Heavy Loads at Nuclear Power Plants. This analyses does credit the existence of these interlocks to prevent these hoists from being of concern for drops of heavy loads. But this analyses does not credit any specific surveillance intervals. The Jam Cutoff Interlock functions to protect the reactor internals and fuel storage racks from damage and prevents separating the handle from a stuck fuel bundle. The change only reduces the surveillance frequency of the refueling equipment which is shown to operate reliably. Reliability of these interlocks has been established through their operating history over the last three refueling outages during which no surveillance test failures were experienced. Engineering judgment was used in conjunction with this operating history of the interlocks to determine the proposed surveillance test frequencies. Due to the demonstrated high reliability of the subject interlocks no discernible increase in the frequency of the failure of this equipment is expected at the revised intervals.

The proposed changes only reduces the surveillance frequencies of these interlocks. The interlocks will continue to provide the intended equipment protection after the change since no change is made in the design or operation of these interlocks or the associated refueling equipment and due to the established reliability of the interlocks.

The surveillances affected by this change were removed from the Technical Specifications via amendment #120. Therefore, this change does not require a change in the Technical Specifications.

The proposed change does not result in an unreviewed safety question.

II Safety Evaluation

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 surveillance requirements being changed are contained in the TRM and UFSAR Appendix 16B and not in the Technical Specifications. This change has no impact on any current Technical Specification.

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 interlock functions affected by this change protect the refueling platform, the fuel handling platform, the reactor vessel, the reactor vessel internals, the fuel storage racks, and the fuel assemblies from overloading or overstressing. However, failure of any of these interlock functions is not an event initiator in the postulated FHAs discussed in UFSAR Sections 15.7.4 and 15.7.6 or in any of the reactivity events discussed in Section 15.4.1.1.

The proposed change only reduces the surveillance frequencies of these interlocks. No change is made in the design or operation of these interlocks or the associated refueling equipment. Reliability of these interlocks has been established through their operating history over the last three refueling outages during which no surveillance test failures were experienced. Engineering judgment was used in conjunction with this operating history of the interlocks to determine the proposed surveillance test frequencies. Due to the demonstrated high reliability of the subject interlocks no discernible increase in the frequency of the failure of this equipment is expected at the revised intervals. The interlocks will, therefore, continue to provide the intended equipment protection described in UFSAR Chapter 9 after the change.

Therefore, the proposed change will not increase the probability of the accident previously evaluated in the SAR.

2. May increase consequences of accidents previously evaluated in the SAR.

Yes ☐ No ■

BASIS:

The refueling interlocks affected by this change and their associated instrumentation are not assumed in the mitigation of the consequences of FHA analyses discussed in UFSAR Sections 15.7.4 and 15.7.6 or the reactivity events discussed in Section 15.4.1.1. The proposed change, therefore, does not increase the consequences of the accident previously evaluated in the SAR.

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

BASIS:

The interlock functions affected by this change protect the refueling platform, the fuel handling platform, the reactor vessel, the reactor vessel internals, the upper containment fuel racks, and the fuel assemblies from overloading or overstressing. However, failure of any of these interlock functions is not an event initiator in the postulated FHAs discussed in UFSAR Sections 15.7.4 and 15.7.6 or in any of the reactivity events discussed in Section 15.4.1.1.

These interlocks, except for the Jam Cutoff Interlock, are credited in the analyses presented in UFSAR Appendix 9D, "GGNS Compliance with NUREG-0612, Control of Heavy Loads at Nuclear Power Plants. This analyses does credit the existence of these interlocks to prevent these hoists from being of concern for drops of heavy loads. But this analyses does not credit any specific surveillance intervals. The Jam Cutoff Interlock functions to protect the reactor internals and upper containment fuel racks from damage and prevents separating the handle from a stuck fuel bundle.

The proposed change only reduces the surveillance frequencies of these interlocks. No change is made in the design or operation of these interlocks or the associated refueling equipment. Reliability of these interlocks has been established through their operating history over the last three refueling outages during which no surveillance test failures were experienced. Engineering judgment was used in conjunction with this operating history of the interlocks to determine the proposed surveillance test frequencies. Due to the demonstrated high reliability of the subject interlocks no discernible increase in the frequency of the failure of this equipment is expected at the revised intervals. The interlocks will, therefore, continue to provide the intended equipment protection discussed of UFSAR Chapter 9 after the change.

The proposed change, therefore, does not increase the probability of malfunction of the equipment important to safety.

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

BASIS:

The proposed change only reduces the surveillance frequencies of these interlocks. No change is made in the design or operation of these interlocks or the associated refueling equipment. Reliability of these interlocks has been established through their operating history over the last three refueling outages during which no surveillance test failures were experienced. Engineering judgment was used in conjunction with this operating history of the interlocks to determine the proposed surveillance test frequencies. Due to the demonstrated high reliability of the subject interlocks no discernible increase in the frequency of the failure of this equipment is expected at the revised intervals. The proposed change, therefore, does not increase the consequences of a malfunction of equipment important to safety.

5. May create the possibility of an accident of different type than any previously evaluated in the SAR. Yes ☐ No ■

BASIS:

The proposed change only reduces the surveillance frequencies of these interlocks. No change is made in the design or operation of these interlocks or the associated refueling equipment. Reliability of these interlocks has been established through their operating history over the last three refueling outages during which no surveillance test failures were experienced. Engineering judgment was used in conjunction with this operating history of the interlocks to determine the proposed surveillance test frequencies. Due to the demonstrated high reliability of the subject interlocks no discernible increase in the frequency of the failure of this equipment is expected at the revised intervals. Therefore, no new equipment failure modes are introduced as a result of this change. Consequently, the change would not create accident of a different type than any previously evaluated in the SAR.

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

BASIS:

The proposed change only reduces the surveillance frequencies of these interlocks. No change is made in the design or operation of these interlocks or the associated refueling equipment. Reliability of these interlocks has been established through their operating history over the last three refueling outages during which no surveillance test failures were experienced. Engineering judgment was used in conjunction with this operating history of the interlocks to determine the proposed surveillance test frequencies. Due to the demonstrated high reliability of the subject interlocks no discernible increase in the frequency of the failure of this equipment is expected at the revised intervals. Therefore, the proposed change does not create possibility of a malfunction of equipment important to safety of different type than any previously evaluated.

7. Will reduce margin of safety as defined in the Basis for any technical specification. Yes ☐ No ■

BASIS:

The interlocks on which the surveillance intervals are being extended are not required by the Technical Specifications. Reliability of these interlocks has been established through their operating history over the last three refueling outages during which no surveillance test failures were experienced. Engineering judgment was used in conjunction with this operating history of the interlocks to determine the proposed surveillance test frequencies. Due to the demonstrated high reliability of the subject interlocks no discernible increase in the frequency of the failure of this equipment is expected at the revised intervals and as a result the proposed changes are not expected to affect the bases for any Technical Specification limit or surveillance requirement. The proposed change, therefore, does not reduce the margin of safety as defined in the basis for any technical specification.

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:

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:

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

   BASIS:

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:

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

I. Safety Evaluation Overview

Page 1 of 5

A. Reference Data

ORIGINATOR: RON GREEN   DEPT/SECT: P&SE   EVAL. #: 96-0087-R00

DOCUMENT EVALUATED: WO 174234

SYSTEM NO: *B33 REACTOR RECIRCULATION SYSTEM*

REFERENCES:

TECHNICAL SPECIFICATION 3.4.2 FLOW CONTROL VALVES

SAFETY EVALUATION 86-095

FSAR SECTION 15.3.2 RECIRCULATION FLOW CONTROL FAILURE - DECREASING FLOW RATE

FSAR SECTION 15.4.5 RECIRCULATION FLOW CONTROL FAILURE WITH INCREASING FLOW

FSAR 15.4.5.3.2.2 SLOW OPENING OF A RECIRCULATION FLOW CONTROL VALVE

FSAR 5.4.1.1 SAFETY DESIGN BASIS (REACTOR RECIRCULATION SYSTEM)

FSAR 5.4.1.9 FLOW CONTROL SYSTEM DESCRIPTION

VENDOR MANUAL 460000787 REACTOR RECIRCULATION HYDRAULIC AND ASSOCIATED ELECTRONIC AND ELECTRICAL EQUIPMENT

FSAR CHANGE REQUIRED? ☐ Yes ☒ No CR# CR or (n/a)

FSAR SECTIONS TO BE REVISED:

TRM CHANGE REQUIRED? (8) ☐ Yes ☒ No

TECH. SPEC. CHANGE REQUIRED? (9) ☐ 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

EXPLAIN: SYSTEM OPERATION PER SOI IS NOT CHANGED BY THIS EVALUATION

IF YES TO THE LAST QUESTION, HAVE THE ORGANIZATIONS RESPONSIBLE FOR THOSE CHANGES BEEN NOTIFIED? (11) ☐ Yes

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

01-S-06-24	Revision: 102
Attachment V	Page 2 of 5

Signatures and Approvals of Attached Safety and Environmental Evaluation

Evaluated: (12) RON GREEN (PRINT NAME) / ORIGINATOR Ron Green (SIGNATURE) / 9-22-96 DATE

Reviewed: (13) D. R. GRAHAM (PRINT NAME) / INDEPENDENT REVIEWER D. R. Graham (SIGNATURE) / 9-22-96 DATE

Reviewed: (14) (PRINT NAME) / OTHER REVIEWER (IF REQUIRED) (SIGNATURE) / DATE

Plant Safety Review Committee Review

(15) Approved per telcon TOM TANKERSLEY, CHAIRMAN, PSRC 9/22/96 DATE

FOR [signature]

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

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT: *WO 174234 DOCUMENTS TROUBLESHOOTING EFFORTS FOR RECIRCULATION FLOW CONTROL VALVE "B" (1B33F060B). THE RESULTS OF TROUBLESHOOTING HAVE SHOWN THAT THE POSITION FEEDBACK TRANSDUCER, THE RVDT IS NOT RESPONDING AS REQUIRED. THE RESPONSE LAST EVENING (9-21-96) HAS BEEN NOISY AND ERRATIC AT TIMES, THUS THE RVDT IS CONSIDERED TO BE OPERATING IN A DEGRADED CONDITION. THIS SAFETY EVALUATION DOCUMENTS THE EVALUATION OF VALVE OPERATION WITH A DEGRADED OR POTENTIALLY DEGRADED POSITION FEEDBACK SIGNAL FROM THE FCV.*

REASON FOR CHANGE, TEST OR EXPERIMENT: *THE REASON FOR THIS EVALUATION IS TO PROVIDE DOCUMENTATION OF THE REVIEW OF PERIODICALLY UNLOCKING THE HPU TO REPOSITION THE VALVE AS NECESSARY TO MAINTAIN RECIRCULATION FLOW. THIS EVALUATION IS NECESSARY BECAUSE THE FCV B MUST BE REPOSITIONED PERIODICALLY DUE TO AN INTERNAL LOOP HYDRAULIC LEAK CAUSING THE FCV B TO SLOWLY DRIFT CLOSED AND ALSO TO ACCOMPLISH FCV OPENING OR CLOSING FOR NORMAL RECIRCULATION FLOW CHANGES AND NORMAL PLANT SHUTDOWN. THIS OPERATION DOES NOT INVOLVE A SAFETY FUNCTION, AS THE RECIRCULATION SYSTEM IS A POWER GENERATION SYSTEM AND IS NOT REQUIRED FOR SAFETY, NOR REQUIRED TO OPERATE DURING OR AFTER ANY DESIGN BASIS ACCIDENT. FURTHER, THE CONTROLS AND INTERLOCKS ARE NOT REQUIRED OR DESIGNED TO COMPLY WITH ANY SINGLE-FAILURE CRITERIA. CONTROL SYSTEM F[illegible]URES RESULTING IN COMPLETE LOSS OF CONTROL SIGNAL WILL RESULT IN ELECTRICAL/HYDRAULIC LOCKING OF THE FINAL CONTROL VALVE ACTUATOR.*

SAFETY EVALUATION SUMMARY AND CONCLUSIONS: *THIS SAFETY EVALUATION ADDRESSES PERIODIC OPERATION OF THE HPU TO CHANGE THE FCV POSITION UNDER THE CONDITIONS OF A DEGRADED POSITION FEEDBACK SIGNAL. OPERATION OF THE FCV WITH A LOCKED-UP HPU HAS BEEN PREVIOUSLY EVALUATED UNDER SE 86-095. THAT SAFETY EVALUATION IS CONSIDERED STILL APPLICABLE, SINCE THE ONLY CHANGE SINCE THAT EVALUATION WAS COMPLETED IS THE CHANGE TO NEW ITS (IMPROVED TECHNICAL SPECIFICATIONS) CONCERNING FCV OPERABILITY AND RATE OF MOVEMENT, WHICH HAS NO BEARING ON HPU LOCKUP.*

THE POSITION FEEDBACK SIGNAL, WHICH IS SUPPLIED BY THE RVDT, IS COMPARED TO THE POSITION SETPOINT SIGNAL BY THE POSITION CONTROLLER SUMMING JUNCTION. THE RESULTANT POSITION DEVIATION SIGNAL IS AMPLIFIED BY THE POSITION CONTROLLER AMPLIFIERS TO PROVIDE A VELOCITY DEMAND SIGNAL WHICH WILL CAUSE THE ACTUATOR TO TRAVEL IN THE DIRECTION REQUIRED TO REDUCE THE POSITION DEVIATION TO ZERO. THUS, VALVE POSITION WILL RESPOND TO THE POSITION SET POINT SIGNAL. DEGRADATION OF THE RVDT OUTPUT SIGNAL OR LOSS OF THE OUTPUT SIGNAL DOES NOT RESULT IN UNCONTROLLABLE VALVE MOVEMENT DURING THE TIME THE HPU IS UNLOCKED, BECAUSE THE POSITION CONTROLLER LIMITER LIMITS THE MAGNITUDE OF THE VELOCITY SET POINT SIGNAL SO THAT ACTUATOR VELOCITY WILL NOT EXCEED PRESET LIMITS EVEN IF LARGE RAPID CHANGES IN POSITION SETPOINT OCCUR. IF THE POSITION SETPOINT (DEMAND) SIGNAL EXCEEDS PRESET LIMITS, OR EXCEEDS DESIGN RATE OF CHANGE LIMITS, OR IF THE SIGNAL IS UNSTABLE, THE HPU SUBLOOPS ARE AUTOMATICALLY LOCKED, PREVENTING VALVE MOTION.

FURTHER, IT IS NOT POSTULATED THAT A FAILED OR DEGRADED SIGNAL FROM THE RVDT WOULD RESULT IN A RAPID OPENING OF THE FLOW CONTROL VALVE, HOWEVER, IF IN THE UNLIKELY EVENT THIS WERE TO OCCUR DURING VALVE POSITION CHANGES, THE FLOW CONTROL VALVE RATE OF OPENING IS LIMITED ELECTRICALLY AND HYDRAULICALLY (VIA VELOCITY LIMIT ORIFICE) TO PREVENT POWER EXCURSIONS DUE TO EXCESSIVE REACTIVITY ADDITION.

II. Safety Evaluation

(19) ☐ Safety Evaluation not required per Completed Safety and Environmental Pre-Screening or Applicability Review. Proceed to Section III for Environmental Evaluation.

A. Technical Specifications (20)

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: *TECH SPEC 3.4.2 ADDRESSES OPERABILITY OF FLOW CONTROL VALVES. THE TECH SPEC BASES IS CONCERNED WITH OPERABILITY IN REGARD TO VALVE STROKE RATE. THE POSITION FEEDBACK SIGNAL, WHETHER NORMAL OR DEGRADED, DOES NOT AFFECT RATE OF VALVE STROKE, THEREFORE PERIODIC OPERATION OF A FCV WITH A POTENTIALLY DEGRADED RVDT DOES NOT CHANGE OR AFFECT GGNS UNIT 1 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: OPERATION OF THE FCV WITH A POTENTIALLY DEGRADED RVDT DOES NOT INCREASE THE PROBABILITY OF ANY ACCIDENT EVALUATED IN THE SAR. THE EXPECTED ACTION ON DEGRADATION OF THE FEEDBACK SIGNAL IS THE TRIPPING OF THE ASSOCIATED HPU DUE TO EXCESS SERVO ERROR (5% DIFFERENCE BETWEEN POSITION DEMAND AND ACTUAL POSITION). EVEN WITH A COMPLETE FAILURE OF THE ELECTRONICS, THE FCV IS DESIGNED TO LIMIT STROKING RATE TO < 30 % PER SECOND IN THE OPENING DIRECTION AND < 60% IN THE CLOSING DIRECTION.

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

☐ Yes
☒ No

BASIS: THE ONLY POSSIBLE ACCIDENT THAT COULD BE CONSIDERED WOULD BE REACTIVITY ADDITION DUE TO RAPID OPENING OF A FCV. SINCE FAILURE OR DEGRADED OPERATION OF THE RVDT DOES NOT AFFECT THE RATE OF OPENING OR CLOSING OF THE FCV, NO CONSEQUENCES OF ANY ACCIDENT EVALUATED IN THE SAR ARE INCREASED BY PERIODIC OPERATION OF THE FCV WITH A POTENTIALLY DEGRADED RVDT.

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

☐ Yes
☒ No

BASIS: THE RECIRCULATION SYSTEM PERFORMS NO SAFETY FUNCTION AND NO EQUIPMENT IMPORTANT TO SAFETY IS AFFECTED BY UTILIZING A POTENTIALLY DEGRADED RVDT. THE RELIABILITY OF THE RECIRCULATION SYSTEM TO PERFORM ITS FUNCTION OF MAINTAINING REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY IS NOT AFFECTED.

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

☐ Yes
☒ No

BASIS: SINCE NO EQUIPMENT IMPORTANT TO SAFETY IS AFFECTED, THERE ARE NO CONSEQUENCES OF ANY MALFUNCTION TO INCREASE.

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

☐ Yes
☒ No

BASIS: THE SAR EVALUATES BOTH SLOW AND FAST OPENING AND CLOSING OF ONE OR BOTH FCVs. NO OTHER ACCIDENT OTHER THAN VALVE MOVEMENT DUE TO AN INCORRECT FEEDBACK SIGNAL IS POSTULATED AS POSSIBLE.

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 RECIRCULATION SYSTEMS ONLY FUNCTION RELATED TO SAFETY IS TO MAINTAIN THE REACTOR PRESSURE BOUNDARY. NO EQUIPMENT IMPORTANT TO SAFETY IS AFFECTED, THERE FORE NO MALFUNCTION OF ANY OTHER EQUIPMENT IS POSTULATED.

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

☐ Yes
☒ No

BASIS: *No margin of safety is affected by operation of the FCV with a potentially degraded RVDT. The valve stroke rate is not affected by the RVDT.*

III. Environmental Evaluation[22]

☐ Environmental Evaluation not required per Completed Safety and Environmental Pre-Screening or Applicability Review.

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

A. Environmental Protection Plan [23]

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

BASIS: THE RECIRCULATION SYSTEM IS NOT ADDRESSED IN THE EPP.

B. Unreviewed Environmental Question [24]

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 RECIRCULATION SYSTEM IS IN THE DRYWELL. INTEGRITY OF THE REACTOR VESSEL IS NOT IMPACTED AS ADDRESSED IN THE ABOVE SAFETY EVALUATION , THEREFORE THERE IS NO IMPACT ON THE ENVIRONMENT.*

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

BASIS:

OPERATION WITH A POTENTIALLY DEGRADED RVDT HAS NO IMPACT ON EFFLUENTS OR POWER.

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

BASIS: NO IMPACT TO THE ENVIRONMENT.

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

I. Safety Evaluation Overview

Page 1 of 9

A. Reference Data

ORIGINATOR: ALAN J. MALONE DEPT/SECT: P&SE EVAL. #: 96-0098-R00

DOCUMENT EVALUATED: LICENSING DOCUMENT CHANGE REQUEST NO. 96-123

SYSTEM NO(S): N/A (INSERT N/A IF NOT APPLICABLE)

REFERENCES: SEE ATTACHED LICENSING DOCUMENT CHANGE REQUEST

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

EXPLAIN. N/A

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.)

Signatures and Approvals of Attached Safety and Environmental Evaluation

Evaluated: ALAN J. MALONE (PRINT NAME) / ORIGINATOR / [signature] (SIGNATURE) / 10-11-96 DATE

Reviewed: William B. Brice (PRINT NAME) / INDEPENDENT REVIEWER / [signature] (SIGNATURE) / 10-11-96 DATE

Reviewed: N/A (PRINT NAME) / OTHER REVIEWER (IF REQUIRED) / N/A [initials] 10-17-96 (SIGNATURE) / DATE

Plant Safety Review Committee Review

[signature] CHAIRMAN, PSRC 10/18/96 DATE

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

BRIEF DESCRIPTION OF CHANGE, TEST OR EXPERIMENT

This change revises Technical Requirements Manual (TRM) Surveillance Requirement (SR) 7.6.3.3.g.2 to allow the Standby Liquid Control pump relief valves to be tested at least once per 18 months during plant or system shutdowns to verify that they open within the specified 3% tolerance. Currently, TRM SR 7.6.3.3.g.2 requires that the relief valves must be tested "[a]t least once per 18 months, during shutdown," without clarifying whether the "shutdown" is plant or system shutdown. Although the TRM requirement could be interpreted as currently allowing the testing during system shutdown, we have conservatively chosen to address this change as an intent change, rather than an editorial, or non-intent, change.

This change is a specific application of a general clarification of the required intervals specified in ASME Code, Section XI, as stated in NUREG-1482. According to NUREG-1482, testing or examination of any valve in the IST Program can be deferred to a refueling outage interval, without relief, using the guidance in ANSI/ASME OM Standard, Part 10. Additionally, it is allowable to test or examine those valves that are currently tested/examined on a refueling outage frequency during system outages, provided that the testing or examination is done at the same intervals as the refueling outage intervals.

REASON FOR CHANGE, TEST OR EXPERIMENT

Performance of valve set pressure testing, exercising testing and disassembly of valves for inspection of internals is frequently on the critical path schedule during refueling outages and contributes to lengthened outages, thereby requiring purchase of more expensive replacement power, as well as requiring additional personnel resources. Performance of this testing and inspections during system outages while the plant is operating at power contributes to shorter refueling outages and better use of plant staff personnel, resulting in lower outage costs. Although it is anticipated that these tests and examinations will be performed only when a system is being taken out of service for other reasons, the potential exists that the schedule for performing a test or examination may require that the system be taken out of service solely to comply with the required test or examination intervals specified in ASME Code Section XI.

SAFETY EVALUATION SUMMARY AND CONCLUSIONS

Testing or examination of any valve in the IST Program can be deferred to a refueling outage interval, without relief, using the guidance in OM-10. Additionally, it is allowable to test those valves that are currently tested or examined on a refueling outage frequency during system outages, provided that the testing/examination is done at the same intervals as the refueling outage intervals.

Since the systems in which these components are installed must be declared inoperable while they are removed from service for tests and examinations, the systems must be evaluated for the effect on the plant before they are taken out of service, as required by Technical Specifications. In fact, the Tech Specs require that the plant enter a limiting condition for operation (LCO) whenever such safety-related systems are taken out of service. Since the systems are out of service, they cannot influence the operation of the plant, increase the probability of accidents or malfunctions of equipment, nor can they significantly affect the consequences of any accidents or malfunctions of equipment.

II. Safety Evaluation

☐ Safety Evaluation not required per Completed Safety and Environmental Pre-Screening or Applicability Review. Proceed to Section III for Environmental Evaluation.

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: Pump and valve inservice testing (IST) requirements are given in ASME Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," which is invoked by GGNS Unit 1 Technical Specification (Tech Spec) 5.5.6 and Technical Requirements Manual (TRM) Surveillance Requirement (SR) 7.6.3.3.

Tech Spec 5.5.6 is a general testing requirement that simply requires that the testing and examinations specified in ASME Code Section XI shall be performed. It also specifies the required frequencies in days for the intervals (e.g., weekly, monthly, quarterly, etc.) that are given in ASME Code Section XI. It does not get into the specifics of the individual testing and examination requirements, nor does it address testing components during shutdown. As discussed in "Brief Description of Change, Test or Experiment," above, and amplified in Licensing Document Change Request No. 96-123, the NRC has determined, in NUREG-1482, that performance of cold shutdown testing during system outages meets the intent of the ASME Code Section XI requirements.

The general requirements in TRM SR 7.6.3.3 to comply with Section XI were at one time Technical Specification SR 4.0.5, but they were removed from the Tech Specs and transferred to the TRM in Tech Spec Amendment 120. Also, the specific requirement to verify that the SLC "pump relief valve opens within 3% of the system design pressure" was formerly in Tech Spec SR 4.1.5.d.2 but was removed and transferred to the TRM in Tech Spec Amendment 120

Although several other Tech Specs touch on the Section XI valve inservice testing requirements (for example, SR 3.5.1.4, SR 3.6.1.3.4, SR 3.6.1.3.6, SR 3.6.1.7.2, SR 3.6.2.3.2, SR 3.6.4.2.2, SR 3.6.5.3.3, etc., require surveillances to be performed "In accordance with the Inservice Testing Program"), they do not affect the interpretation of the meaning of shutdown as addressed in this Safety Evaluation. Also, this change does not conflict with any other current GGNS technical specification.

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: As stated in Subarticle IWV-1100 ("Scope") of ASME Code Section XI, the Pump and valve IST requirements are intended "to verify operational readiness of certain Class 1, 2 and

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

BASIS (CONTINUED):

3 valves (and their actuating and position indicating systems) in light-water cooled nuclear power plants, which are required to perform a specific function in shutting down a reactor to the cold shutdown condition or in mitigating the consequences of a accident." This change affects only the times at which these IST requirements are performed. It does not change the frequency at which they are performed, nor does it change the types of tests and examinations that are performed or the acceptance criteria for the tests and examinations.

Since the systems in which these components are installed must be declared inoperable while they are removed from service for tests and examinations, the systems must be evaluated for the effect on the plant before they are taken out of service, as required by Technical Specifications. In fact, the Tech Specs require that the plant enter a limiting condition for operation (LCO) whenever such safety-related systems are taken out of service. Since the systems are out of service, they cannot influence the operation of the plant or increase the probability of occurrence of any accident.

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

BASIS:

As stated in Subarticle IWV-1100 ("Scope") of ASME Code Section XI, the Pump and valve IST requirements are intended "to verify operational readiness of certain Class 1, 2 and 3 valves (and their actuating and position indicating systems) in light-water cooled nuclear power plants, which are required to perform a specific function in shutting down a reactor to the cold shutdown condition or in mitigating the consequences of a accident." This change affects only the times at which these IST requirements are performed. It does not change the frequency at which they are performed, nor does it change the types of tests and examinations that are performed or the acceptance criteria for the tests and examinations.

Since the systems in which these components are installed must be declared inoperable while they are removed from service for tests and examinations, the systems must be evaluated for the effect on the plant before they are taken out of service, as required by Technical Specifications. In fact, the Tech Specs require that the plant enter a limiting condition for operation (LCO) whenever such safety-related systems are taken out of service. The LCO conditions usually limit the length of time that the plant may be operated with the systems out of service, and they may also impose additional requirements, such as increased verification of the availability of other mitigation systems. The NRC has evaluated and approved the operation of the plant with the systems out of service under these LCO conditions, thereby indicating (whether stated or not) that the consequences of any accidents and malfunctions of equipment that may occur while the systems are out of service are within the capabilities of the remaining operable systems to mitigate.

All systems and parts of systems which are taken out of service under LCOs have been evaluated for their absence during an accident. As long as the terms of the LCO, including time limits and other requirements, have been complied with, the consequences to the plant have been found to be acceptably low.

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

BASIS (CONTINUED):

Although the unavailability of the system may increase the consequences of an accident by its not being capable of performing its function during and following the accident, the NRC's approval to operate the plant under LCO with the system out of service indicates that the increase is not "significant" within the meaning of 10CFR50.59. Therefore, this change does not increase the consequences of any accidents.

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

☐ Yes
☒ No

BASIS: As stated in Subarticle IWV-1100 ("Scope") of ASME Code Section XI, the Pump and valve IST requirements are intended "to verify operational readiness of certain Class 1, 2 and 3 valves (and their actuating and position indicating systems) in light-water cooled nuclear power plants, which are required to perform a specific function in shutting down a reactor to the cold shutdown condition or in mitigating the consequences of a accident." This change affects only the times at which these IST requirements are performed. It does not change the frequency at which they are performed, nor does it change the types of tests and examinations that are performed or the acceptance criteria for the tests and examinations.

Since the systems in which these components are installed must be declared inoperable while they are removed from service for tests and examinations, the systems must be evaluated for the effect on the plant before they are taken out of service, as required by Technical Specifications. In fact, the Tech Specs require that the plant enter a limiting condition for operation (LCO) whenever such safety-related systems are taken out of service. Since the systems are out of service, they cannot influence the operation of the plant or increase the probability of occurrence of any equipment malfunctions.

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

☐ Yes
☒ No

BASIS: As stated in Subarticle IWV-1100 ("Scope") of ASME Code Section XI, the Pump and valve IST requirements are intended "to verify operational readiness of certain Class 1, 2 and 3 valves (and their actuating and position indicating systems) in light-water cooled nuclear power plants, which are required to perform a specific function in shutting down a reactor to the cold shutdown condition or in mitigating the consequences of a accident." This change affects only the times at which these IST requirements are performed. It does not change the frequency at which they are performed, nor does it change the types of tests and examinations that are performed or the acceptance criteria for the tests and examinations.

Since the systems in which these components are installed must be declared inoperable while they are removed from service for tests and examinations, the systems must be evaluated for the effect on the plant before they are taken out of service, as required by Technical

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

☐ Yes
☒ No

BASIS (CONTINUED):

Specifications. In fact, the Tech Specs require that the plant enter a limiting condition for operation (LCO) whenever such safety-related systems are taken out of service. The LCO conditions usually limit the length of time that the plant may be operated with the systems out of service, and they may also impose additional requirements, such as increased verification of the availability of other mitigation systems. The NRC has evaluated and approved the operation of the plant with the systems out of service under these LCO conditions, thereby indicating (whether stated or not) that the consequences of any accidents and malfunctions of equipment that may occur while the systems are out of service are within the capabilities of the remaining operable systems to mitigate.

All systems and parts of systems which are taken out of service under LCOs have been evaluated for their absence during malfunctions of equipment. As long as the terms of the LCO, including time limits and other requirements, have been complied with, the consequences to the plant have been found to be acceptably low.

Although the unavailability of the system may increase the consequences of an accident by its not being capable of performing its function during and following the accident, the NRC's approval to operate the plant under LCO with the system out of service indicates that the increase is not "significant" within the meaning of 10CFR50.59. Therefore, this change does not increase the consequences of malfunction of equipment previously evaluated in the SAR.

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

☐ Yes
☒ No

BASIS: As stated in Subarticle IWV-1100 ("Scope") of ASME Code Section XI, the Pump and valve IST requirements are intended "to verify operational readiness of certain Class 1, 2 and 3 valves (and their actuating and position indicating systems) in light-water cooled nuclear power plants, which are required to perform a specific function in shutting down a reactor to the cold shutdown condition or in mitigating the consequences of a accident." This change affects only the times at which these IST requirements are performed. It does not change the frequency at which they are performed, nor does it change the types of tests and examinations that are performed or the acceptance criteria for the tests and examinations.

Since the systems in which these components are installed must be declared inoperable while they are removed from service for tests and examinations, the systems must be evaluated for the effect on the plant before they are taken out of service, as required by Technical Specifications. In fact, the Tech Specs require that the plant enter a limiting condition for operation (LCO) whenever such safety-related systems are taken out of service. Since the systems are out of service and are not relied upon for a safety function, they cannot influence the operation of the plant or create the possibility for an accident of a different type than any previously evaluated.

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 stated in Subarticle IWV-1100 ("Scope") of ASME Code Section XI, the Pump and valve IST requirements are intended "to verify operational readiness of certain Class 1, 2 and 3 valves (and their actuating and position indicating systems) in light-water cooled nuclear power plants, which are required to perform a specific function in shutting down a reactor to the cold shutdown condition or in mitigating the consequences of a accident." This change affects only the times at which these IST requirements are performed. It does not change the frequency at which they are performed, nor does it change the types of tests and examinations that are performed or the acceptance criteria for the tests and examinations.

Since the systems in which these components are installed must be declared inoperable while they are removed from service for tests and examinations, the systems must be evaluated before they are taken out of service, as required by Technical Specifications. Since the systems are out of service and are not relied upon to perform a safety function, they cannot influence the operation of the plant nor create the possibility for a malfunction of equipment different from any previously evaluated.

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

☐ Yes
☒ No

BASIS: The Pump and valve IST requirements given in ASME Boiler and Pressure Vessel Code Section XI are intended to provide assurance that pumps and valves which perform a specific function in shutting down the reactor or mitigating the consequences of a accident are capable of performing their functions when called upon to do so. This change affects only the times at which these IST requirements are performed. It does not change the frequency at which they are performed, nor does it change the types of tests and examinations that are performed or the acceptance criteria for the tests and examinations.

Since the systems in which these components are installed must be declared inoperable while they are removed from service for tests and examinations, the systems must be evaluated before they are taken out of service, as required by Technical Specifications. Since the systems are out of service, they cannot influence the operation of the plant, act as precursors to accidents or equipment malfunctions or affect the consequences of any accidents or equipment malfunctions. Therefore, this change cannot affect any margin of safety defined in the Tech Specs or their Bases.

III. Environmental Evaluation ☐ Environmental Evaluation not required per Completed Safety and Environmental Pre-Screening or 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 Environmental Protection Plan does not address the performance of tests and examinations to comply with ASME Code Section XI, nor does it address the intervals at which such tests and examinations are performed, the acceptance criteria for such tests and examinations, or the plant conditions necessary for such tests and examinations.

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 AS'´B. ☐ Yes ☒ No

BASIS: The Final Environmental Statement does not address the performance of tests and examinations to comply with ASME Code Section XI, nor does it address the intervals at which such tests and examinations are performed, the acceptance criteria for such tests and examinations, or the plant conditions necessary for such tests and examinations.

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

BASIS: These tests and examinations can be performed only when the system they are connected into is out of service. Since the system is out of service, performance of these tests and examinations has no effect on effluents or power level.

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

☐ Yes
☒ No

BASIS: The Final Environmental Statement does not address the performance of tests and examinations to comply with ASME Code Section XI, nor does it need to. This change concerns only the times at which tests and examinations specified in ASME Code Section XI are performed. The Pump and valve IST requirements given in ASME Boiler and Pressure Vessel Code Section XI are intended to provide assurance that pumps and valves which perform a specific function in shutting down the reactor or mitigating the consequences of a accident are capable of performing their functions when called upon to do so. This change does not change the frequency at which they are performed, nor does it change the types of tests and examinations that are performed or the acceptance criteria for the tests and examinations.

Although not specifically stated in the Final Environmental Statement or the Environmental Protection Plan, the obvious expectation and assumed requirement in both documents is that the plant will continue to be operated in accordance with the Technical Specifications and Technical Requirements Manual. As discussed in II.A.1 above, this change does not require a change to the GGNS Technical Specifications, and all tests and examinations will continue to be performed under the provisions of the Tech Specs.