Entergy

Entergy Operations, Inc.
Waterloo Road
P.O. Box 756
Port Gibson, MS 39150
Tel 601 437 6299

Charles A. Bottemiller
Manager
Plant Licensing

GNRO-2004/00029

April 15, 2004

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Subject: Report of 10CFR50.59 Safety Evaluations and Commitment Changes – November 01, 2002 through March 31, 2004
Grand Gulf Nuclear Station
Docket No. 50-416
License No. NPF-29

Ladies and Gentlemen:

Pursuant to 10CFR50.59(d)(2), Entergy Operations, Inc. hereby submits the summary of 10CFR50.59 evaluations for the November 01, 2002 through March 31, 2004 period. Also attached is the summary of commitment changes for the same period made in accordance with NEI 95-07 Guidelines.

If you have any questions or require additional information, please contact Chuck Holifield at 601-437-6439.

This letter contains no commitments.

Yours truly,

CAB/CDH;cdh

attachments: 1. Table of Contents
2. 10CFR50.59 Evaluations and Commitment Change Evaluations

cc: (See Next Page)

IE47

cc:	Hoeg	T. L.	(GGNS Senior Resident)	(w/a)
	Levanway	D.E	(Wise Carter)	(w/a)
	Reynolds	N. S.		(w/a)
	Smith	L. J.	(Wise Carter)	(w/a)
	Thomas	H. L.		(w/o)

U.S. Nuclear Regulatory Commission
ATTN: Mr. Bruce S. Mallett (w/2)
Regional Administrator, Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-4005

U.S. Nuclear Regulatory Commission
ATTN: Mr. Bhalchandra Vaidya, NRR/DLPM (w/2)
ATTN: FOR ADDRESSEE ONLY
ATTN: U.S. Postal Delivery Address Only
Mail Stop OWFN/7D-1
Washington, D. C. 20555-0001

bcc:
OUTLOOK MAIL: DISTRIBUTION IS ALL ELECTRONIC

James	D. E.	(ANO-NSAPL)
Miller	O. J.	R. A. (GG-TRNG)
Bottemiller	C. A.	(GG-NSAPL)
Mitchell	T. G.	(ANO-NSA)
Williams	G. A.	(GG-VP)
Edwards	J. B.	(GG-GMPO)
GGN CENTRAL FILE (175)		
GGN PLANT LICENSING		
Sen	G.	(W3-NSAPL)
King	R. J.	(RB-NSA)
Burford	F. G.	(ECH-NSL)
Lorfing	D. N.	(RB-NSAPL)
Krupa	M. A.	(GG-NSA)
Wiles	D. P.	(GG-ENG)

OTHER: File (LRS_DOCS Directory – GNRI or GNRO)

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GRAND GULF NUCLEAR STATION
10CFR50.59 SUMMARY REPORT FOR THE PERIOD
STARTING NOVEMBER 01, 2002 AND ENDING MARCH 31, 2004

MEANING OF ACRONYMS			
ARI	Alarm Response Instruction	LOP	Loss of Power
ASTM	American Society for Testing and Materials	MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
CCE	Commitment Change Evaluation	MCPR	Minimum Critical Power Ratio
CMWT	Core Megawatts Thermal	MNCR	Material Nonconformance Report
CR	Condition Report	MOV	Motor Operated Valve
DCP	Design Change Package	MS	Mechanical Standard
EP	Emergency Procedure	MSIV-LCS	Main Steam Isolation Valve Leakage Control System
EPI	Equipment Performance Instruction	NPE	Nuclear Plant Engineering
EPRI	Electric Power Research Institute	NSSS	Nuclear Steam Supply System
ER	Engineering Request	PDMS	Plant Data Management System
ES	Electrical Standard	PPM	Parts Per Million
ESF	Engineered Safety Feature	PRA	Probabilistic Risk Assessment
GE	General Electric	PSW	Plant Service Water
GG	Grand Gulf	RCIC	Reactor Core Isolation Cooling
GGN	Grand Gulf Nuclear	RFO	Refueling Outage
GPM	Gallons Per Minute	RHR	Residual Heat Removal
IOI	Integrated Operating Instruction	RPV	Reactor Pressure Vessel
ISI	In Service Inspection	SCN	Standard Change Notice
IST	In Service Testing	SERI	System Energy Resources, Inc.
LBDC	License Basis Document Change	SGTS	Standby Gas Treatment System
LDC	License Document Change	SOER	Significant Operating Experience Report
LHGR	Linear Heat Generation Rate	SSW	Standby Service Water
LLRT	Local Leak Rate Test	TRM	Technical Requirements Manual
LOCA	Loss of Coolant Accident	UHS	Ultimate Heat Sink

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SAFETY EVALUATIONS

Evaluation No.	Initiating Document	Summary
SE 2002-0007-R02	LBDR 2002-104	Core Operating Limits Report Cycle 13 Rev. 2
SE 2002-0008-R01	LBDC 2002-116, Revision 1	TRM 7.6.3.3, Item g.1, Inservice Inspection and Testing Program Surveillance 3.6.1.3.6
SE 2003-0001-R00	ER-GG-2003-0152-000 Revision 0	Recirc flow control valve min position change analysis
SE 2003-0002-R00	Calculation XC-Q1N11-94004 Revision 0	Update of Offsite and Control Room doses from a MSL break
SE 2004-0001-R00	LDC 2004-009	Core Operating Limits Report Cycle 14 Rev. 0
SE 2004-0001-R01	LDC 2004-009	Core Operating Limits Report Cycle 14 Rev. 1
SE 2004-0002-R00	Temp Alt 04-005 (Division 1) Revision 0 Temp Alt 04-006 (Division 2) Revision 0	Manual operation of SSW cooling tower fans in Modes 4 and 5 to assist in maintaining basin temperature within admin limits during cold weather

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GRAND GULF NUCLEAR STATION
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COMMITMENT CHANGE EVALUATIONS

Commitment No.	Source Document	Summary
CCE 2002-0007	10CFR, Part 55.45	Annual transient and steady state simulator testing requirement was deleted by a revision to 10CFR55.45.
CCE 2002-0008	AECM-90/0004	Procedural weaknesses identified by response to SALP report for inclusion of appropriate improvements into the 10CFR50.59 process. Completely remove this commitment.
CCE 2002-0009	AECM-82/490	Valve P53F006 seat leakage will be tested at a frequency not to exceed 60 months versus every 18 months.
CCE 2003-0001	AECM-88/0024, Att. 1 Area 5.S3	Delete the drawing control procedure to provide for ready acceptance of NPE reviewed, approved and issued drawings and assignment of drawing coordinator to coordinate design drawings and resolve problems. This process is no longer applicable.
CCE 2003-0002	AECM-89/0003.88-26-01.III 2.B	NMM NF-104 has been revised to include detailed instructions for completing DOE/NRC form 741.
CCE 2003-0003	AECM-89/0003.88-26-01.III.3.a&b	Delete the commitment to reassign responsibility of SNM program to the GGNS General Manager
CCE 2003-0004	GNRO 93/00029 Paragraph 3, Sentence 4	Delete the commitment to track and trend Rosemount transmitters that are susceptible to fill-oil loss.

GRAND GULF NUCLEAR STATION
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COMMITMENT CHANGE EVALUATIONS

CCE 2003-0005	GNRO 94/103.94-13-01.III.S2	Miscellaneous material storage procedure was revised to identify the personnel who are required to maintain copies of the inventory sheets and item location maps for items stored in the pools.
CCE 2003-0006	VOIDED	
CCE 2003-0007	AECM 85/0201 Response 2, Paragraph 3	Commitment specifying that GGNS Admin procedure "Determination of Safety/Quality Classifications" will be used for guidance is no longer valid since the Master Equipment List and its successors (CDB, EDB) have satisfied that requirement.
CCE 2003-0008	GNRO-2000/00011 99-1903 Paragraph 3	Since information tags are now caution tags, "Protective Tagging" procedure will be revised to more clearly describe how caution tags will be used in place of information tags.
CCE 2003-0009	GNRI 99/00047	Check valves E38F002A/B and E38F003A/B will no longer be disassembled and inspected on a sample basis. Instead, they will be full stroke opened and closed on a cold shutdown frequency per Ma-1988, Part 10.
CCE 2003-0010	GNRO 02/00054	Delete the commitment to perform a parametric study at the uprated conditions to quantify the impact of Thermal Power Optimization on GGNS wear rates and update the CHECWORKS model as necessary.
CCE 2003-0011	SIL 156	Delete the commitment to verify neutron monitoring instrument tubes are properly seated after maintenance activities are performed in the vicinity of the tubes.

ATTACHMENT 2

10CFR50.59 Evaluations
and
Commitment Change Evaluations

GGNS 50.59 Safety Evaluation Number

SE 2002-0007-R02

I. OVERVIEW / SIGNATURES

OSRC MEETING # 03/2003
DATE: 1/23/2003

Facility: GGNS

Document Reviewed: Core Operating Limits Report (COLR) **Change/Rev.:** LDC 2002-104

System Designator(s)/Description: N/A

Description of Proposed Change

This evaluation addresses the Cycle 13 reload changes and operation of the Cycle 13 reload core as given in the Core Operating Limits Report (COLR). Revision 1 address the explicit incorporation of the increased GE11 channel bow described in CR-GGN-2002-01810 into the Cycle 13 core design and operation. Revision 2 removes the limitations on equipment out of service allowances previously imposed for Cycle 13 exposures >11 GWd/MTU.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

- ☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.
- ☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2______. (Insert item # from Section 5.2.2.2).
- ☐ The proposed activity is controlled by another regulation per Section 5.2.2.3.

QA RECORD	RT.	NON-QA RECORD	INITIALS	NUMBER of PAGES	DATE	RELATED DOCUMENT NUMBER
✓	2/4.33			12 20	4/23/2003	

4/23/2003

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	**SCREENING**	**Sections I, II, and III required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, III, and IV required**
☑	**50.59 EVALUATION (#: 2002-0007-R. 2)**	**Sections I, II, III, and V required**

Preparer: Guy B. Spikes/ Guy B. Spikes /EOI/Nucl. Eng. – SA/ 1/23/03
Name (print) / Signature / Company / Department / Date

Reviewer: G. E. Broadbent/ G. E. Broadbent /EOI/Nucl. Eng. – SA/ 1/23/03
Name (print) / Signature / Company / Department / Date

OSRC R.V. Moomaw / [signature] EOI/Outage Mgr 1/23/03
Chairman's Signature / Date (N/A for Screenings and 50.59 Evaluation Exemptions)

List of Assisting/Contributing Personnel:

Name:	Scope of Assistance:
J. A. Elam (Echelon Core Design)	Core Design and neutronic input
D. L. Smith (Echelon Fuel Fabrication)	Fuel mechanical input
J. P. Head (Echelon Fuel Fabrication)	Core stability and hydraulic compatibility input
G. W. Smith	EP input

II. SCREENING

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?** (Check "N/A" for those documents that are not applicable to the facility.)

Operating License	YES	NO	N/A	CHANGE # and/or SECTIONS TO BE REVISED
Operating License	☐	☑		
TS	☐	☑		
NRC Orders	☐	☑	☐	
If "YES", obtain NRC approval prior to implementing the change. (See Section 5.1.13 for exceptions.)				

LBDs controlled under 50.59	YES	NO	N/A	CHANGE # and/or SECTIONS TO BE REVISED
FSAR	☑	☐		LDC 2002-106
TS Bases	☐	☑		
Technical Requirements Manual	☐	☑		
Core Operating Limits Report	☑	☐		LDC 2002-104
Offsite Dose Calculations Manual	☐	☑		
NRC Safety Evaluation Reports[1]	☐	☑		
If "YES", perform an Exemption Review per Section IV OR perform a 50.59 Evaluation per Section V.				

LBDs controlled under other regulations	YES	NO	N/A	CHANGE # and/or SECTIONS TO BE REVISED
Quality Assurance Program Manual[2]	☐	☑		
Emergency Plan[2]	☐	☑		
Security Plan[2, 3]	☐	☑		
Fire Protection Program[4] (includes the Fire Hazards Analysis)	☐	☑	☐	
If "YES", evaluate/process any changes in accordance with the appropriate regulation.				

2. **Does the proposed activity involve a test or experiment not described in the FSAR?**
   If "yes," perform an Exemption Review per Section IV OR perform a 50.59 Evaluation per Section V.

   ☐ Yes
   ☑ No

3. **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation? (Check "N/A" if dry fuel storage is not applicable to the facility.)**

   ☐ Yes
   ☐ No
   ☑ N/A

   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

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[1] If "YES," see Section 5.1.5.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed.
[3] The Security Plan is classified as safeguards and can only be reviewed by personnel with the appropriate security clearance. The Preparer should notify the security department of potential changes to the Security Plan.
[4] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

B. Basis

(Provide a clear, concise basis for the answers given in the applicable sections above. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis.)

The current MCPR Safety Limit has been shown to be applicable to the Cycle 13 core. As such, Tech Spec 2.1.1.2 does not need to be revised. There are no other Tech Spec, Bases, or TRM changes required for Cycle 13 operation. There are no NRC orders applicable to the Cycle 13 reload campaign.

The Cycle 13 core will contain fuel types currently described in the FSAR, however, the core characteristics and response will be somewhat different than currently described in the SAR. As such, Cycle 13 analyses have been performed for the new core and the FSAR will need to be updated appropriately as will the COLR.

The Cycle 13 core design and operation will not affect the QAPM, Emergency Plan, Security Plan, or Fire Protection Program.

C. References

[Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101.]

LBDs/Documents Reviewed:	Keywords:
OLM, FSAR, TS, Bases, TRM	

D. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines)

☐ Yes
☑ No

If "Yes," list the required changes.

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review.

Will the proposed Change being evaluated:

Yes	No	
☐	☑	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
☐	☑	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
☐	☑	Involve dredging activities in a lake, river, pond, or stream?
☐	☑	Increase the amount of thermal heat being discharged to the river or lake?
☐	☑	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
☐	☑	Discharge any chemicals new or different from that previously discharged?
☐	☑	Change the design or operation of the intake or discharge structures?
☐	☑	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
☐	☑	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
☐	☑	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
☐	☑	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
☐	☑	Involve the installation or use of equipment that will result in an air emission discharge?
☐	☑	Involve the installation or modification of a stationary or mobile tank?
☐	☑	Involve the use or storage of oils or chemicals?
☐	☑	Involve burial or placement of any solid wastes in the site area that may effect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

V. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after PSRC approval, if available.)

Brief description of change, test, or experiment:

This safety evaluation assesses the reload-related changes associated with Cycle 13 operation as presented in the Core Operating Limits Report (COLR) located in the Operating License Manual (OLM). Cycle 13 has been designed for 492 Effective Full Power Days with a core consisting of 240 fresh ATRIUM-10 assemblies, 204 once-burnt ATRIUM-10 assemblies, 228 twice burnt GE assemblies, and 128 thrice-burnt GE11 assemblies. There are no TS, TS Bases, or TRM changes required to operate with this new core, however, the FSAR does require updates. The Cycle 13 core has been designed and analyzed for a 1.7% power uprate at a rated thermal power of 3898 MWt in support of the Cycle 13 implementation of the Appendix K uprate. As such, the reload analyses are applicable to both the current power level of 3833 MWt as well as the uprated power level of 3898 MWt. The Appendix K uprate is being reviewed by the NRC and is not addressed in this evaluation. Individual design changes on GGNS systems are not addressed in this evaluation. Individual design changes on GGNS systems are assessed in the safety evaluation associated with the specific change package and are not addressed in this evaluation. Attachment 1 provides a detailed description of the Cycle 13 reload analysis and the issues considered in this evaluation. Revision 1 provides additional evaluations associated with the explicit incorporation of the increased GE11 channel bow described in CR-GGN-2002-01810 into the Cycle 13 core design and operation. Revision 2 removes the limitations on equipment out of service allowances previously imposed for Cycle 13 exposures >11 GWd/MTU.

Reason for proposed Change:

Cycle 13 operation will require new core operating limits and the Core Operating Limits Report has been revised to include these new limits. These limits include flow-, power-, and exposure-dependent LHGR, MAPLHGR, and MCPR limits.

50.59 Evaluation summary and conclusions

The Cycle 13 core configuration and operation has been evaluated with respect to mechanical, neutronic, thermal-hydraulic, dose, thermal performance, and methods considerations for GGNS. This evaluation concludes that the reload-related changes associated with Cycle 13 operation will not constitute an unreviewed safety question.

B. License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☑ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☑ No

BASIS:

The Cycle 13 core loading and cycle operation will not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR. The precursors to these events are independent of the core design and the frequency classifications reported in SAR Chapter 15 are unaffected by the core parameters. The following considerations support this conclusion.

Mechanical
The ATRIUM-10 mechanical design has been reviewed for use at Grand Gulf. No unusual failure modes or increased failure frequency have been identified for this fuel design. This is the second reload at GGNS with ATRIUM-10 fuel and this fuel design has accumulated significant problem-free operational experience at other plants. The Cycle 13 bundles will operate within the power history assumptions in the fuel mechanical analyses and will result in exposures within the analyzed burnup limits of the ATRIUM-10 and the GE11 mechanical designs. All design criteria for the GE11 bundles have been shown to meet their respective limits including those that will be irradiated for a fourth cycle.

Nuclear
The neutronic characteristics of the Cycle 13 GE11 and ATRIUM-10 mixed core design have been considered in the safety analysis. Adequate shutdown margin has been predicted by analysis and will be confirmed during startup tests. In addition, the hold-down capability of the standby liquid control system and the subcriticality of Cycle 13 fuel in the spent fuel storage racks have been confirmed. Therefore, the probability of inadvertent criticality has not been increased by the introduction of the Cycle 13 reload fuel.

Thermal-Hydraulic
FRA-ANP's modeling of the GE fuel and the thermal-hydraulic compatibility of the ATRIUM-10 and GE11 fuel have been reviewed and found acceptable. In fact, to accurately model the GE11 bundle hydraulics, a Cycle 11 GE11 bundle was shipped to FRA-ANP for hydraulic testing in their hydraulic test facility. Analyses have been performed to demonstrate that Cycle 13 meets all Enhanced-1A stability performance criteria without changes to the E1A hardware or power-flow map region boundaries. Therefore, the probability of thermal-hydraulic instabilities has not increased.

Analyzed Events
The probability of the occurrence of anticipated operational events is not dependent on the core configuration. No changes to the plant design are required for the Cycle

13 core. The Cycle 13 core loading will not affect the precursors to any of the Chapter 15 events. The probability of an analyzed event therefore has not increased.

As described in UFSAR Section 15A.6.5.3, the Control Rod Drop Accident (CRDA) results from a failure of the control rod-to-drive mechanism coupling after the control rod becomes stuck in its fully inserted position. Although the increased channel bow condition may result in increased friction between the control blade and its corresponding fuel assemblies, there would not be sufficient friction to result in a mechanical failure of the coupling. Additionally, the control rod drive mechanism would not produce enough force to result in a mechanical failure of the coupling even if the channel bow was so severe that the assemblies would preclude blade movement. As such, channel bow is not considered a precursor to the CRDA, and the increased bow associated with the GE11 bundles would not increase the probability of this event.

On these bases, the probability of occurrence of accidents previously identified in the SAR is not increased for the Cycle 13 core with increased channel bow.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☑ No

BASIS:

The Cycle 13 reload fuel is a design that has been shown to be mechanically, neutronically, and thermal-hydraulically compatible with the co-resident GE11 fuel. No plant modifications are required to accommodate the new core design and the only additional loads placed on plant equipment would be potential friction between the control blades and excessively bowed GE11 bundles. Based on previous experience with bowed fuel at GGNS and Clinton Power Station, this increased friction is not expected to impact scram times. Technical Specification scram time testing requirements during Cycle 13 would identify any potential scram time impacts and the appropriate actions would be taken in accordance with Technical Specifications. Additionally, this increased friction would not be sufficient enough to provide any failures associated with the control blades or the control blade drive system.

A conservative vessel overpressurization analysis has been performed, which shows that the vessel pressure limit is not exceeded. The precursors to any malfunction of equipment important to safety are not affected by this change.

Therefore, there is not more than a minimal increase in the likelihood of an occurrence of a malfunction of a SSC important to safety previously evaluated in the FSAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☑ No

BASIS:

As reported in Attachment 1, the acceptance criteria reported in SAR Section 15.0.3.1 and the Technical Specifications are satisfied for each event classification. Core operating limits have been developed to ensure that moderate frequency events do not violate the MCPR safety limit or fuel cladding strain limits. The consequences of infrequent events have been shown to meet the appropriate acceptance criteria while the individual acceptance criteria for the limiting faults have been demonstrated to be satisfied. The following considerations support these conclusions.

Moderate Frequency Events

The Cycle 13 core operating limits have been developed with NRC-approved methodologies such that the MCPR safety limit and the fuel cladding strain limit will not be violated by any analyzed moderate frequency transient initiated from any statepoint available to GGNS. As such, no fuel failures are expected to result from any moderate frequency event. These analyses considered GGNS-specific operational modes such as MEOD, SLO, FWHOS, and EOC-RPT inoperable. These core operating limits consist of MCPR, MAPLHGR and LHGR curves that are functions of flow, power, and exposure. These limits consider conservative channel bow assumptions that bound the current and expected increased bow associated with the GE11 fuel.

These core operating limits will be incorporated into the core monitoring system, however, as with previous cycles, the GE11 operating limits illustrated in the COLR will differ somewhat from those limits in the core monitoring system. GE has generated pellet-based exposure-dependent LHGR and lattice-based MAPLHGR limits for their GE11 fuel bundles; however, for competitive reasons, GE has designated the limits for only the most-limiting or least-limiting lattices as non-proprietary. The COLR, which is submitted to the NRC for information only, will therefore report these non-proprietary limits for reference purposes only, and refer to the appropriate GE proprietary document for the actual limit. It is recognized that most lattices will operate at higher limits than these most-limiting COLR and TRM reference curves, however, this is acceptable since the Technical Specifications 3.2.1 and 3.2.3 require that all APLHGRs and LHGRs, respectively be less than or equal to the limits specified in the COLR, which refers to the proprietary GE document for the actual limit.

Infrequent Events

The consequences of the limiting infrequent events have been evaluated and shown to meet their respective acceptance criteria. These events include the pressure regulator failure downscale, misplaced (*i.e.*, misoriented and mislocated) bundle, and single loop operation pump seizure accidents. Radiological analyses using the alternative source term (AST), which has been recently licensed for GGNS, have been performed to ensure that these events will not result in offsite or control room doses greater than their respective acceptance criteria. These evaluations include conservative channel bow assumptions that bound the current and expected increased bow associated with the GE11 fuel.

Limiting Faults

The limiting faults at GGNS include the fuel handling accident, the control rod drop accident, and the design basis LOCA. The radiological analyses for these events have recently been developed as part of the GGNS AST effort and bound the Cycle 13 core parameters. For the LOCA, MAPLHGR operating limits and single-loop multipliers have been developed for the Cycle 13 core configuration such that the requirements of 10CFR50.46 are satisfied. The containment response for the Cycle 13 core was found to be bounded by previous cycles as is the hydrogen analysis. The seismic/LOCA response of the Cycle 13 core has been confirmed to be acceptable. The Cycle 13 core design results in minor changes to two EP parameters (Mclad and Mfuel), however, the existing EP's remain applicable to Cycle 13.

Therefore, the proposed change does not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☑ No

BASIS:

The Cycle 13 reload fuel is a design that has been shown to be compatible with the co-resident fuel types. The malfunctions of key plant components are analyzed as part of the reload process with the results reported in various sections of the SAR. The consequences of these malfunctions have been shown to meet their respective acceptance criteria.

Therefore, Cycle 13 operation will not result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☑ No

BASIS:

The Cycle 13 reload fuel is similar to and compatible with the fuel that was inserted in previous cycles. The details of this design have been specifically considered in the safety analysis and the core monitoring system. No plant modifications are required to accommodate the new core design or Cycle 13 operation. The GGNS Cycle 13 fuel types have been approved for the Cycle 13 reactor chemistry conditions.

The GGNS operational parameters (water chemistry requirements, spectral-shift core designs, and MEOD rod-lines) have been reviewed and are not expected to result in unusual crud buildup like that observed on the high-power GE11 bundles at

River Bend. Inspection of a high-power, once-burnt representative fuel bundle during GGNS RF10 has confirmed that the high-power GGNS Cycle 10 fuel bundles have no unusual crud buildup.

Therefore, Cycle 13 operation will not create a possibility for an accident of a different type than any previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR?

☐ Yes
☑ No

BASIS

The Cycle 13 reload fuel is a design that has been shown to be mechanically, neutronically, and thermal-hydraulically compatible with the co-resident GE11 fuel. The ATRIUM-10 fuel will not introduce any adverse flow distribution effects such as preferential flow through the ATRIUM-10 bundles that may negatively impact the GE 11 bundles. No plant modifications are required to accommodate the new core design and no additional loads will be imposed on any existing equipment. The ATRIUM-10 bundles provide sufficient clearance for proper control blade operation and allow sufficient bypass flow in the bypass region to provide adequate cooling for control blades and in-core detectors. There are no special operational considerations associated with the Cycle 13 core other than those associated with the increased bow for the GE11 bundles. The higher friction expected between the control blades and GE11 bundles experiencing increased bow is not sufficient to cause a failure of the fuel bundle, control blade, or control rod drive coupling.

Therefore, Cycle 13 operation will not create the possibility for a malfunction of an SSC important to safety with a different result than previously evaluated in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered?

☐ Yes
☑ No

BASIS:

Mechanical analyses have been performed to ensure that all fuel in the Cycle 13 core meet the mechanical design limits for steady-state operation as well as transient conditions including fatigue damage, creep collapse, corrosion, fuel rod internal pressure, rod bow, internal pressure, etc. Additionally, no Cycle 13 fuel will exceed the applicable burn-up or residence time limits.

Core operating limits have been developed using NRC approved codes in order to ensure that the Cycle 13 fuel will not exceed the MCPR safety limits for steady-state operation and anticipated operation occurrences. Similarly, operating limits have been developed to ensure that the Cycle 13 fuel will not exceed the 1% cladding strain limit or experience core-wide fuel melt during steady-state operation or AOO's. The results of these analyses show that the vessel pressure does not exceed the vessel pressure safety limit. Although some vessel blowdown to the suppression pool may be experienced during some AOO's, which would increase the suppression pool temperature, the bulk containment pressure increase is negligible and would not exceed the design limit.

As described in Attachment 1, a bounding pressurization event with a failure of the direct scram has been analyzed for Cycle 13 to ensure compliance with ASME code requirements. This analysis indicates that the vessel pressure safety limit is not exceeded for Cycle 13.

A design basis limit for the peak fuel enthalpy of 280 cal/gm has been established for the control rod drop accident (CRDA) to preclude significant fuel cladding failure such that core geometry and cooling may be impacted. The CRDA has been evaluated for Cycle 13. This evaluation considers all potential withdrawal sequences and concludes that a CRDA will not exceed the 280 cal/gm peak enthalpy limit. Since this accident is a localized event and the peak enthalpy does not exceed 280 cal/gm, there is no impact on the vessel or containment pressures. As such their respective limits are not exceeded.

10CFR50.46 provides limits associated with the ECCS performance analysis (LOCA analysis). Two such limits are Peak Clad Temperature (PCT) and local clad oxidation. Although these limits are not subject to 10CFR50.59, they are discussed in this evaluation for completeness. Grand Gulf specific analyses have been performed for ATRIUM-10 and GE11 fuel in accordance with 10CFR50.46. These analyses, which are applicable to Cycle 13, show that the PCT and local oxidation are well below the limits set forth in 10CFR50.46. These analyses also show that the core-wide metal water reaction, which is used to evaluate compliance with the containment design limit, is less than the 10CFR50.46 limit. The remainder of the existing containment analysis associated with LOCA events is applicable to Cycle 13 as described in Attachment 1. As such, the containment pressure design limit will not be exceeded for Cycle 13.

An ATWS evaluation has also been performed for Cycle 13. As described in Attachment 1, the resulting vessel pressure remains below the ASME emergency vessel pressure limit of 1500 psig and the temperature response used in the existing ATWS containment analysis is applicable to Cycle 13. Thus, the containment pressure design limit will not be exceeded for the ATWS event.

Additional evaluations have been performed for Cycle 13 including Appendix R (Fire Protection), hydrogen analyses, and SBO as described in Attachment 1. These evaluations show that the existing evaluations are applicable to Cycle 13 and that their respective limits are not exceeded.

Therefore, Cycle 13 operation will not result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?

☐ Yes
☑ No

BASIS:

The reload analyses performed by the fuel vendor utilized NRC approved methods as listed in Technical Specification 5.6.5 and described throughout the FSAR. These methods were consistent with those used for Cycle 12. As described in Attachment 1, the uncertainty applied in the Safety Limit calculation associated with each of the equipment out of service combinations was calculated in accordance with Framatome's NRC approved methodology. All remaining GGNS evaluations currently described in the FSAR have been shown to be applicable to Cycle 13. As such, no new methods were used. Finally, the GGNS EP calculation has been updated to consider the minor changes to two fuel parameters. This revision did not incorporate any new or different methods.

Therefore, Cycle 13 operation will not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.

GGNS 50.59 Safety Evaluation Number

SE 2002-0008-R01

I. OVERVIEW / SIGNATURES

Facility: Grand Gulf Nuclear Station

Document Reviewed: LDC 2002-116 Change/Rev. 1

System Designator(s)/Description: B21 – MSIV's

Description of Proposed Change

TRM 7.6.3.3, Item g.1., Inservice Inspection and Testing Programs Surveillance 3.6.1.3.6, Main Steam Isolation Valve (MSIV) requires MSIV full close stroke within 12 hours after reaching 600 psig prior to entry to Mode 1. This evaluation will remove that requirement for the MSIV being full closed stroke at 600 psig and will permit MSIV IST closed stroke testing during "cold" shutdown. The requirement will be based upon the NUREG-1482 definition of "cold" shutdown testing, which allows cold shutdown frequency testing including valves tested either while decreasing power to reactor cold shutdown for a outage or while increasing power to seady state power operaton after a planned or forced shutdown. A method for fast closing the MSIVs while shutting down is currently addressed in Integrated Operating Instruction 03-1-01-3 step 5.29.3 and will be added to 06-OP-1B21-V-0001. This procedure step would now be credited for MSIV fast closure testing to fulfill IST cold shutdown testing requirements. The MSIVs, which are cold shutdown frequency testing valves, will now be treated like other cold shutdown valves. During a refuel outage, all cold shutdown valves are to be stroked. But, during forced outages and non-refuel outages, cold shutdown valves are stroked per a scheduled sequence until startup, at which time testing is curtailed - not all cold shutdown valves (including the MSIVs) are required to be stroked prior to startup after a forced outage.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

- ☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.
- ☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2______. (Insert item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, IV, and V required
☒	50.59 EVALUATION (#: 2002-0008-R01)	Sections I, II, III, IV, and VI required

Preparer: Gary D. Young / Gary D. Young / EOI / Engineering Programs / 8/21/03
Name (print) / Signature / Company / Department / Date

Reviewer: Robert W. Fuller / Robert W Fuller / EOI / Design Engineering / 8-21-03
Name (print) / Signature / Company / Department / Date

OSRC: Jerry C Roberts [signature] 8/25/03

OSRC MEETING # 028/03
DATE: 8/21/2003

Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name:	**Scope of Assistance:**

II. SCREENING

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?**

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	LDC 2002-116, FSAR Section 16B.1-212
TS Bases	☐	☒	
Technical Requirements Manual	☒	☐	LDC 2002-116, TRM 7.6.3.3
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Reports[1]	☐	☒	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2]	☐	☒	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ Yes ☒ No
   If "yes," perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI.

3. **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation?** ☐ Yes ☐ No ☒ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

B. Basis

[1] If "YES," see Section 5.1.4.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

The purpose of the evaluation is to provide the rationale for enhancing the test requirements for MSIV fast closure. The enhancement treats the MSIVs like other IST cold shutdown frequency valves by allowing MSIV fast closure testing (if required per IST frequency requirements) anytime during cold shutdown (including while decreasing power to reactor cold shutdown or while increasing power to steady state power operation) as defined in NUREG-1482. Currently MSIV fast closure testing is done during startup at 600 psig in 06-OP-1B21-V-0001. It will now be acceptable to perform MSIV fast closing IST testing when decreasing power to reactor cold shutdown, during reactor cold shutdown, or while increasing power to steady state power operation. It is preferred to fast closure test the MSIVs at low steam flow because the MSIVs would be "wet" and "hot" but any pressure and temperature in accordance with IST test frequency requirements is acceptable.

During a refuel outage, all cold shutdown valves (including the MSIVs) are to be stroked. But, during forced and non-refuel outages, cold shutdown valves are stroked per a scheduled sequence until startup, at which time testing is curtailed - therefore, not all cold shutdown valves (including the MSIVs) are required to be stroked prior to startup (per NUREG-1482).

Operating License:

The Grand Gulf Nuclear Station (GGNS) operating license does not affect any MSIV fast closure testing. The Technical Specifications and the Environmental Protection Plan are not impacted by this project. Therefore, the proposed activity does not impact the GGNS operating license.

Technical Specifications:

The MSIV fast closure testing is covered by Technical Specification Surveillance Requirement 3.6.1.3.6. However, none of the Technical Specification requirements for MSIV fast closure testing are affected. In addition, the evaluation will not create a system configuration or operating condition such that a Technical Specifications LCO or surveillance requirement is no longer adequate. Likewise, the evaluation will not bypass or invalidate automatic actuation features required to be operable by the Technical Specifications or exceed any limits specified in the Operating License and Technical Specifications. Therefore, no Technical Specifications change is required for the issuance of this evaluation.

UFSAR:

The UFSAR is affected by this evaluation. UFSAR Section 16B.1 page 16B.1-212 (which is the TRM in the UFSAR) will be changed by this evaluation. MSIV fast closure testing is discussed here and the pressure at which the MSIVs are tested. UFSAR Section 16B.1 section 7.6.3.3 g.1. will be deleted allowing MSIV fast closure testing during per IST cold shutdown frequency requirements. This new information will allow MSIV fast closure testing (if required per IST cold shutdown frequency requirements) during cold shutdown as defined in NUREG-1482, including testing while decreasing power to reactor cold shutdown, during reactor cold shutdown, or while increasing power to seady state power operaton, both after a planned or forced shutdown.

NRC Orders:

The NRC Orders issued at Grand Gulf are not affected by this evaluation because this evaluation deals with MSIV fast closure testing and this evaluation is not to be used for security reasons which is what Grand Gulf's current NRC Orders deal with.

Technical Specification Bases:

There are no Technical Specifications or Bases impacted by this activity. The Technical Specification for Containment isolation is 3.6.1.3 and the surveillance requirement under this Technical Specification is 3.6.1.3.6 for fast closure testing will remain the same. This is an evaluation for assuring that MSIV fast closure testing can be accomplished during cold shut as defined in NUREG-1482, including while decreasing power to cold shutdown, during reactor shutdown, or while increasing power to steady state power operation.

Technical Requirements Manual (TRM):

There is impact to the Technical Requirements Manual affected by this activity. The affected TRM section is 7.6.3.3. This section will be changed to delete the requirement of 7.6.3.3.g.1 that specifies at reactor pressure of 600 psig during reactor startup. The method of MSIV fast closure testing during shutdown is already performed in Integrated Operating Instruction 03-1-01-3, step 5.29.3. The procedure for MSIV fast closure testing, 06-OP-1B21-V-0001 will reflect MSIV fast closure testing during cold shutdown as defined in NUREG-1482, including while dereasing power to cold shutdown, during reactor shutdown, or while increasing power to steady state power operation.

Core Operating Limits Report:

This activity does not impact the COLR (GGNS Core Operating Limits Report). This evaluation discusses MSIV fast closure IST cold shutdown frequency testing during cold shutdown as defined in NUREG-1482. It does not have any impact on the COLR and does not affect any licensing activities.

Offsite Dose Calculations Manual:

This activity does not impact any equipment required to monitor offsite dose. Therefore, no changes to the ODCM is required.

NRC Safety Evaluation Reports:

There is no impact to any SERs by providing an evaluation for MSIV fast closure IST cold shutdown frequency testing being performed during cold shutdown as defined in NUREG-1482.

Quality Assurance Program Manual:

This evaluation complies with all requirements of the Entergy Quality Assurance Program Manual, as applicable. This activity does not change any commitments contained in the QAPM. Therefore, this activity does not require a change to the QAPM.

Emergency Plan:

This evaluation does not impact the interaction of GGNS personnel and offsite agencies in response to an emergency.

Security Plan:

This evaluation does not impact the Security Plan since it does not require the breaching of security barriers. Therefore, no change to the Security Plan is required.

Fire Protection Program (includes the Fire Hazards Analysis):

This evaluation does not impact the Fire Protection Program by providing MSIV fast closure testing during controlled reactor vessel shutdown.

C. <u>References</u>

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. **NOTE: Ensure that electronic and manual searches are performed using controlled copies of documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
UFSAR 16.1 (page 16B.1-212), Technical Specification 3.6.1.3 (SR 3.6.1.3.6),	MSIV, Main Steam Isolation Valve, 600 psig

TRM 7.6.3.3 g 1

LBDs/Documents reviewed manually:

N/A

D. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines.)

☐ **Yes**
☒ **No**

If "Yes," list the required changes.

N/A

References

1. GIN 2001/00986
2. Integrated Operating Instruction (IOI) 03-1-01-3
3. 06-OP-1B21-V-0001, MSIV Operability Test
4. TRM 7.6.3.3 section g paragraph 1
5. Technical Specification Surveillance 3.6.1.3.6
6. NUREG-1482, Guidelines for Inservice Testing at Nuclear Power Plant

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☒ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) ______ ☒ No

____

Name of Security Plan reviewer (print) / Signature / Date

VI. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

Currently TRM 7.6.3.3 g.1. requires MSIV fast closure testing to be performed during startup at 600 psig reactor pressure. This testing is done for Technical Specification 3.6.1.3 under Surveillance Requirement 3.6.1.3.6 which is MSIV fast closure testing. This evaluation deletes the TRM requirement. Procedure 06-OP-1B21-V-0001 will need to be revised to allow MSIV fast closure testing during cold shutdown as defined in NUREG-1482, including while decreasing power to cold shutdown, during reactor shutdown, or while increasing power to steady state power operation. There is a minor procedure change for IOI 03-1-01-3. The tables will be removed from the IOI 03-1-01-3 to the 06-OP-1B21-V-0001 because it currently fast closes the MSIV and records the required information. This will eliminate the need for MSIV fast closure testing during startup after an outage, unless required for post-maintainence retest per the 01-S-07-28 ASME Section XI Repair/Replacement Program checklist.

Reason for proposed Change:

During controlled reactor shutdown MSIV fast closure testing is performed by IOI 03-1-01-03. IST MSIV fast closure testing is currently performed and credited during startup. This may not be the optimum time to test the MSIVs. During controlled shutdown it would be desirable to fast closure test the MSIVs during shutdown around 60 psig reactor pressure. Fast closure testing during shut down reduces the possibility of preconditioning the MSIVs prior to fast closure testing. Currently, the valves may have maintenance performed on them during an outage, thus giving the impression of preconditioning of the MSIVs. Fast Closure testing of the MSIVs during controlled shutdown of the reactor would eliminate this situation and would be an IST enhancement because no perceived MSIV preconditioning occurs.
Additionally, MSIV post-maintenance retest is currently performed during restart at approximately 600 psig. It may be desireable to perform those retests during reactor cold shutdown. This evaluation will provide flexibility to perform testing and retesting at the optimum time to support Operation's activities and outage schedules.

50.59 Evaluation summary and conclusions

For the MSIV fast closure testing, the requirements are going to be deleted in TRM Section 7.6.3.3 g.1. This testing currently performed by Integrated Operating Instruction 03-1-01-3 but not utilized for IST testing will be moved to 06-OP-1B21-V-0001. This will allow MSIV fast closure testing during cold shutdown as defined in NUREG-1482, including while decreasing power to cold shutdown, during reactor shutdown, or while increasing power to steady state power operation. The procedure for this testing is 06-OP-1B21-V-0001. Procedure 06-OP-1B21-V-0001 needs to be revised to include MSIV fast closure testing during cold shutdown as defined in NUREG-1482, including while decreasing power to cold shutdown, during reactor shutdown, or while increasing power to steady state power operation.

Stroking of the MSIVs will be performed per the limitations/requirements of GIN 2001/00986. If the valve is "wet" (i.e., the steam line is filled with steam) the valve can be repeatedly stroked with no minimum wait time between valve cycles and the valve will not experience valve galling or damage due to stroking. Although experience has shown that "cold" stroke of the MSIVs are consistent with the "hot" stroke of the MSIVs, the 06-OP-1B21-V-0001 will be revised with enhanced stroke time acceptance criteria.

If fast closure testing of the MSIVs is performed at 60 psig reactor pressure, the saturation temperature is greater than 300°F. The MSIVs are still "hot" even with the test pressure reduced from 600 psig to 60 psig. The MSIV fast closure time will be unaffected by reducing the test pressure to 60 psig. The MSIVs fast closure stroke time can be performed at any pressure and temperature. Limitations about stroking the MSIVs "cold" and "dry" has been communicated to operations via GIN 2001/00986.

The requirement to MSIV fast closure test at 600 psig was not found in any of the MSIV technical specifications, vendor manual or design documents for the MSIVs. It appears that there was no basis for this reactor pressure other than the MSIVs was desired to be tested "wet" and "hot". Therefore 600 psig was chosen as the test pressure apparently arbitrarily chosen based on steam conditions.

Changing the TRM to allow MSIV fast closure testing is an IST enhancement. It eliminates preconditioning of the MSIVs prior to as-found testing. This TRM change does not affect MSIV design functions and continues to provide safety pressure boundary, seismic, and tornado protection requirements established in the design and licensing basis for the MSIVs. All essential plant systems and equipment will function as assumed in the Accident Analysis. Therefore, this change will have no effect on any consequences of the accidents evaluated previously in the UFSAR, will not change offsite dose to the public, will not affect any fission product barriers, and does not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the UFSAR. As a result, this change will not increase the consequences of an accident or create an accident of a different type than any evaluated previously in the UFSAR.

The MSIV are designed to meet the current licensing and design requirements. This evaluation does not change the MSIV system actuation, flow parameters, or the pressure boundary requirements and they will function as assumed in the Accident Analysis. Therefore, this change will not increase the consequences of a malfunction of equipment important-to-safety or create the possibility of a malfunction of equipment important-to-safety of a different type than any evaluated previously in the USAR. Also, this change will not reduce the margin of safety as defined in the basis for any Technical Specification.

Based on the results of this safety-evaluation, the affects associated with this evaluation are inconsequential and, therefore, do not constitute an Unreviewed Safety Question (USQ).

This change is an enhancement to MSIV fast closure testing allowing testing during cold shutdown as defined in NUREG-1482, including while decreasing power to cold shutdown, during reactor shutdown, or while increasing power to steady state power operation. It does not affect plant influents or effluents. Therefore, this evaluation does not represent a change to the Environmental Protection Plan or a change that will affect the environment. There is no potential for an Unreviewed Environmental Question, therefore, there is no need to perform an environmental evaluation.

B. License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.

☐ Yes
☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

This evaluation does not make a physical change to the Plant. It changes the TRM requirements for MSIV fast closure testing. This enhancement will allow MSIV fast closure testing during cold shutdown as defined in NUREG-1482, including while decreasing power to cold shutdown, during reactor shutdown, or while increasing power to steady state power operation. This is a change to testing conditions and does not change frequency of occurrence of an accident. A change to the TRM is made to allow fast closure testing of the MSIVs during cold shutdown as defined in NUREG-1482, including while decreasing power to cold shutdown, during reactor shutdown, or while increasing power to steady state power operation. The MSIV isolation mode of operation is not changed or affected by this evaluation.

Chapters 3, 6 and 15 of the USAR were reviewed to determine the impact of this evaluation on any accidents previously analyzed. Several accidents and transients analyzed in Chapter 15 were determined to be unaffected this testing enhancement.

The change in TRM 7.6.3.3 allowing testing during cold shutdown as defined in NUREG-1482, including while decreasing power to cold shutdown, during reactor shutdown, or while increasing power to steady state power operation has no impact on the probability of occurrence of allowing reactor vessel inventory to leak into the environment. Based upon the above discussion, it is concluded that the proposed evaluation does not result in an increase in the frequency of occurrence of an accident previously evaluated in the USAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The TRM change and evaluation meets the original licensing and design requirements for the associated MSIVs. The change does not affect required MSIV design or system operation, meets containment isolation requirements, does not invalidate existing seismic, pipe stress, or tornado design criteria, moderate energy line crack flooding and spray analysis, high energy line break jet impingement and pipe whip evaluations, pose a fire hazard, and does not impact RPV Internals, core analysis or thermal hydraulic design criteria and analysis, or affect any UFSAR Chapter 15 Accident Analysis. Based upon the above discussion, it is concluded that this change does not increase the likelihood of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR.

The change in the TRM allows MSIV fast closure testing to occur either during cold shutdown as defined in NUREG-1482, including while decreasing power to cold shutdown, during reactor shutdown, or while increasing power to steady state power operation.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

This evaluation allows MSIV fast closure testing to occur during cold shutdown as defined in NUREG-1482, including while decreasing power to cold shutdown, during reactor shutdown, or while increasing power to steady state power operation. The TRM 6.7.3.3 will be revised to reflect these changes. Procedures to collect the MSIV fast closure data exist and will be utilized as needed to meet the test conditions. All affected piping, fittings, and valve pressure boundaries are qualified to the appropriate fluid transients and operational conditions and meet the design and licensing requirements for pressure boundary integrity and containment isolation provisions.

This TRM change and evaluation affects equipment located within the primary containment and the auxiliary building, and does not penetrate any structural wall or barriers. Therefore there is no affect to the boundary integrity of any fire area. Therefore this evaluation will not compromise the function nor integrity of structures, systems or components important to safety and has no effect on the Fire Hazards Analysis Report.

A review of USAR Chapter 15 Accident Analysis was performed. The proposed TRM change was evaluated against the existing safety analyses to determine if any of the analyses are impacted. The criteria used in this evaluation were that the change should not impact the ability of MSIVs to provide containment isolation within the Technical Specification Surveillance requirement, should not create an event of a type not previously analyzed, and previous component analyses should not be negatively impacted. The proposed TRM and FSAR change satisfies the evaluation criteria, and the TRM and FSAR change is within the bounds of the existing safety analyses.

The other potentially impacted accidents are a FSAR Chapter 6 analysis. These accidents are considered limiting faults. For the case of the recirculation line break inside containment (i.e., drywell), the reactor vessel depressurizes very rapidly and the MSIVs would go closed within the required Technical Specification Surveillance requirement (SR 3.6.1.3.6).

MSIVs reactor/containment isolation remains unchanged. All essential plant systems and equipment will function as assumed in the Accident Analysis. There is no increase in offsite dose due to any accident previously evaluated. Therefore the proposed activity does not increase the consequences of an accident evaluated previously in the UFSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

This evaluation does not make a physical change to the Plant. MSIV fast closure testing is currently performed and credited during startup. This may not be the optimum time to test the MSIVs. During controlled shutdown it may be desirable to fast closure test the MSIVs during shutdown around 60 psig reactor pressure. Fast closure testing during shut down eliminates the possibility of preconditioning the MSIVs prior to fast closure as-found testing. Currently, the valves may be worked on during an outage, thus giving the impression of preconditioning of the MSIVs. Fast Closure testing of the MSIVs during cold shutdown as defined in NUREG-1482, including while decreasing power to cold shutdown, during reactor shutdown, or while increasing power to steady state power operation.

As an IST enhancement, the TRM is changed to allow testing during restart or shutdown. Integrating Operating Instructions 03-1-01-3 is a procedure that currently tests MSIV fast closure during Shutdown. Procedure 06-OP-1B21-V-0001 would be revised with the MSIV fast closure testing instructions from IOI 03-1-01-3 when shutting down.

This TRM change and evaluation meets the current design and licensing basis such that all affected systems, structures, and components, including the RPV and its internals that are Important to safety meet all required operational modes and will function as assumed in the Accident Analysis. Fast Closure Containment isolation of the MSIVs does not compromise containment integrity because the inboard and outboard MSIV containment isolation valves will still meet the Technical Specification Surveillance Requirement 3.6.1.3.6 for closure time. Evaluation of the various piping and hydraulic transients were performed to ensure that no adverse impact to the MSIV lines were created. Any breaks and/or malfunctions of safety related or important to safety equipment and the mitigating actions for these failures or malfunctions remain the same. As such there is no change in the radiological consequences at the site boundary. Therefore this evaluation will not increase the consequences of a malfunction of equipment important to safety evaluated previously in the UFSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:
This evaluation meets and does not invalidate the current design and licensing basis for the following:

- Containment isolation provisions.
- USAR Chapter 15 - Accident Analysis.
- Fluid transient and system operational conditions.

The TRM is being changed to allow MSIV fast closure testing during either restart or shutdown. There are procedures in place that currently collect the MSIV fast closure times. The MSIV isolation time is not be changed and the enhancement to the TRM provides a test condition that is conservative with respect to the existing test condition of reactor vessel pressure at 600 psig. There are no other events postulated as a result of this evaluation which could create the possibility of an accident of a different type than any evaluated previously in the USAR.

Therefore this evaluation as previously described will not create the possibility of an accident of a different type than evaluated previously in the UFSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS
This evaluation meets the current design and licensing basis such that all affected systems, structures, and components including the RPV and its internals that are important to safety meet all required plant operational modes and events. This includes tornado, seismic, and pipe stress criteria, moderate energy line crack flooding and spray analysis, high energy pipe break jet impingement and pipe whip target evaluations, fire hazards, and UFSAR Chapter 15 Accident Analysis.

MSIV fast closure testing is currently performed and credited during startup. This condition is not conservative with respect to the proposal to test the MSIVs during cold shutdown as defined in NUREG-1482, including while decreasing power to cold shutdown, during reactor shutdown, or while increasing power to steady state power operation. During controlled shutdown it may be desirable to fast closure test the MSIVs during shutdown around 60 psig reactor pressure. Fast closure testing during shut down eliminates the possibility of preconditioning the MSIVs prior to fast closure IST as-found testing. Currently, the valves may be worked on during an outage, thus possibly giving the impression of preconditioning of the MSIVs. Fast Closure testing of the MSIVs during cold shutdown as defined in NUREG-1482, including while decreasing power to cold shutdown, during reactor shutdown, or while increasing power to steady state power operation would eliminate this situation.

There are no other postulated events which could create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the UFSAR. Therefore this evaluation as previously described will not create the possibility of a malfunction of equipment important to safety of a different type than any evaluated previously in the UFSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

The MSIV capability of containment isolation is not changed or affected by this evaluation. The MSIV fast stroke time for containment isolation will perform as required in accordance with the design and licensing basis including USAR Chapter 15 Accident Analysis. Therefore, this evaluation causes no change to plant equipment or the design basis that is addressed by the Operating License. A review of the Facility Operating License Conditions and the Technical Specification show that these documents are not affected by this evaluation. Technical Specifications, Section 3.6, Containment Systems, Subsection 3.6.1.3, Primary Containment Isolation Valve (PCIVS) address containment isolation. No change to these sections of the Technical Specification is required. The Technical Specification Surveillance Requirement 3.6.1.3.6 will maintain the current isolation valve stroke times.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

This evaluation does not make a physical change to the Plant. Containment isolation is achieved by closing the MSIVs. The TRM section 7.6.3.3 g.1 will be deleted . A change to the 06-OP-1B21-V-0001 is made to incorporate Integrated Operating Instruction 03-1-01-3 for MSIV fast closure. The MSIV mode of operation is not changed or affected by this evaluation.

As an IST enhancement, the 06-OP-1B21-V-0001 is changed to allow testing during cold shutdown as defined in NUREG-1482, including while decreasing power to cold shutdown, during reactor shutdown, or while increasing power to steady state power operation. Integrating Operating Instructions 03-1-01-3 is a procedure that currently tests MSIV fast closure during Shutdown. Procedure 06-OP-1B21-V-0001 would be revised with the MSIV fast closure testing instructions from IOI 03-1-01-3 at reactor pressure of low steam flow during controlled shutdown. The MSIV fast closure testing would be allowed during cold shutdown as defined in NUREG-1482, including while decreasing power to cold shutdown, during reactor shutdown, or while increasing power to steady state power operation.

GGNS 50.59 Safety Evaluation Number

SE 2003-0001-R00

I. OVERVIEW / SIGNATURES

OSRC MEETING # 022/13
DATE: 6/12/2003

Facility: Grand Gulf Nuclear Station

Document Reviewed: ER-GG-2003-0152-000 Change/Rev. 0

System Designator(s)/Description: B33

Description of Proposed Change

ER-GG-2003-0152 performs the analysis to increase the recirc flow control valve (FCV) min position (MIN ED) from 7% to 12% to alleviate FCV sticking problems after recirc pump upshift.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.

☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2______.
(Insert item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, IV, and V required
☒	50.59 EVALUATION (#: 2003-0001 R.0)	Sections I, II, III, IV, and VI required

Preparer: G.E. Broadbent / G.E. Broadbent / EOI / Engineering / 6/12/03
Name (print) / Signature / Company / Department / Date

Reviewer: S.C. Stanchfield/ S.C. Stanchfield / EOI / Engineering / 6/12/03
Name (print) / Signature / Company / Department / Date

OSRC: Ron Moomaw / [signature] 6/12/03
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name:	Scope of Assistance:

II. SCREENING

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?**

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	LDC 2003-051
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Reports[1]	☐	☒	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2]	☐	☒	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ Yes ☒ No
   If "yes," perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI.

3. **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation?** ☐ Yes ☐ No ☒ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

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[1] If "YES," see Section 5.1.4.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

B. Basis

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

The FCV MIN ED position is not governed by the Operating License, Tech Specs, Tech Spec Bases, TRM or NRC Orders. It is described in the FSAR and is credited in the Idle Loop Startup analysis in FSAR Section 15.4.4. A revised reload analysis has confirmed the operating limits in the current Core Operating Limits Report are unaffected by this change in MIN ED.

C. References

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. **NOTE: Ensure that electronic and manual searches are performed using controlled copies of documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
FSAR, Operating License Manual	"recirc", "flow control valve", "loop startup", "min ed"

LBDs/Documents reviewed manually:

D. **Is the validity of this Review dependent on any other change?** (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines.)

☐ Yes
☒ No

If "Yes," list the required changes.

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☐ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) ____________ ☐ No

____

Name of Security Plan reviewer (print) / Signature / Date

V. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A, above.

A. Check the applicable boxes below. If any of the boxes are checked, a 50.59 Evaluation is not required. If none of the boxes are checked, perform a 50.59 Evaluation in accordance with Section VI. Provide supporting documentation or references as appropriate.

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.6.1.1:

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; AND

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; AND

The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.6.1.2. Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.6.1.3. Reference: ______________________

☐ The proposed activity is controlled by another regulation per Section 5.6.1.4.

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10CFR50.59 Review Program Guidelines for guidance.

VI. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

ER-GG-2003-0152 000 increases the recirc system flow control valve MIN ED position from 7% to 12%.

Reason for proposed Change:

As described in CR-GGN-2003-00352, plant startups have been adversely affected by the recirc flow control valves sticking after recirc pump upshift. An increase in the MIN ED position will allow the recirc pumps to be upshifted with the valves in a more open position which will alleviate this problem.

50.59 Evaluation summary and conclusions

This 50.59 Evaluation concludes that the only analysis impacted by the change to MIN ED is the idle loop startup. This analysis has been revised with approved methodology and remains a non-limiting event. No fission product barriers are impacted since no fuel failure is calculated to occur for this event. A detailed evaluation in the reviewed document concludes that this change will not adversely impact the recirculation system, cause an unplanned scram, or violate fuel requirements. Consequently, this change does not increase the frequency of occurrence of accidents or SSC malfunctions previously evaluated in the FSAR. Since there is no fuel failure, there are no increases in the consequences of accidents or SSC malfunctions previously evaluated in the FSAR.

B. License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.

☐ Yes
☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?

   ☐ Yes
   ☒ No

   BASIS:

   The only FSAR event that is impacted by an increase in MIN ED is the idle loop startup (ILS), described in FSAR Section 15.4.4. This event involves the startup of a second cold recirc loop with an accompanying operator error. The increased FCV position when the idle loop is started results in a faster influx of cold water into the reactor and consequently a larger reactivity-induced power swing. The ILS is classified as a moderate frequency event, which may occur at a frequency of once per calendar year to once every 20 years per FSAR Section 15.0.3.1.

   The increase in MIN ED would not impact the number of times an idle loop would need to be started nor the potential for an operator error in starting this loop. Consequently, this MIN ED change will not result in an increase in the frequency of occurrence of an accident previously evaluated in the FSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?

   ☐ Yes
   ☒ No

   BASIS:

   ER-GG-2003-0152 evaluated the impact of this change on the recirculation system, cause an unplanned scram, or violate fuel requirements. For the recirc system, the normal and structural loads on the jet pumps were found to be unaffected since the maximum flow through the jet pumps is not increased. The recirc pump motor was found to have only a small increase in Brake Horsepower and winding heating which would have no appreciable affect on motor life or reliability. The neutron flux transient generated by the pump upshift was calculated to not exceed 80% power, thereby maintaining significant margin to the flux scram setpoint at 118% power. The width of the neutron flux spike generated by the pump upshift is predicted to be sufficiently small that a thermal power scram would also not occur. The ILS event has been re-analyzed and confirmed to meet the applicable fuel thermal requirements.

   On these bases, this ER evaluation concludes that this change will not adversely impact the recirculation system, cause an unplanned scram, or violate fuel requirements. Therefore, this MIN ED change will not increase the likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated in the FSAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR?

   ☐ Yes
   ☒ No

BASIS:

As described in the Basis to Question 1, the only FSAR event that is impacted by an increase in MIN ED is the ILS. Per FSAR Section 15.4.4, the ILS was found to be a non-limiting event, whose results met the MCPR safety limit and LHGR overpower criteria.

As described in detail in ER-GG-2003-0152, the ILS event has been re-analyzed with the increased MIN ED. This evaluation concluded that the MCPR safety limit and LHGR overpower criteria are still not violated in an ILS. Therefore, fuel failures are still not predicted to occur as a result of this change in the event of an ILS.

With no fuel failures, there is no radiological release associated with the ILS. Consequently, this MIN ED change will not result in an increase in the consequences of an accident previously evaluated in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The FSAR analyses postulate a number of different malfunctions of the recirc system that may be impacted by the flow control valves. Specifically, these failures include fast opening, fast closing, and line break.

The increase in MIN ED does not impact the rate at which the FCV can open or close such that the flow-dependent reload analyses are not impacted. In the LOCA dose analysis, the source term release is governed by the non-mechanistic NRC guidance in Reg Guide 1.183, which is independent of FCV position at the onset of the event. In the ECCS LOCA performance analysis (FSAR Section 6.3.3), the break area is governed by the flow area at the jet pump nozzles and is not impacted by the position of the FCV.

Consequently, this MIN ED change will not result in an increase in the consequences of a malfunction of a SSC important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The evaluation in ER-GG-2003-0152 concludes that this change will not adversely impact the recirculation system, cause an unplanned scram, or violate fuel requirements. Therefore, this MIN ED change will not create the possibility for an accident of a different type than previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS

No new SSCs are required and no new operational modes or failure modes of the FCV are introduced with this MIN ED change. The impacts on the recirc system have been evaluated in ER-GG-2003-0152 and concluded to be acceptable. Therefore, this MIN ED change will not create the possibility for a malfunction of a SSC important to safety with a different result than any previously analyzed in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered?

☐ Yes
☒ No

BASIS:

The fission product barriers potentially impacted by this change are the fuel cladding and reactor coolant system. ER-GG-2003-0152 concludes that this change will not adversely impact the recirculation system or violate fuel requirements. Therefore, this MIN ED change will not result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?

☐ Yes
☒ No

BASIS:

SAR 15.4.4.3.1 describes the analytical model used to evaluate the ILS event during Cycle 1. GE applied their point kinetics code called REDY to confirm that this event was non-limiting for the operating limits. Per SAR Table 15D.4-1, this same methodology was applied when GE re-analyzed this event for the GGNS Maximum Extended Operating Domain (MEOD).

The current GGNS fuel vendor, Framatome, has performed the ILS evaluation supporting this increase in MIN ED. This evaluation applied Framatome's one-dimensional transient code, called COTRANSA2. This methodology has been used for other events at GGNS, is part of the GGNS COLR methods in Technical Specification 5.6.5, and has been NRC-approved for analysis of the ILS transient (letter: S Richards (NRC) to J.F. Mallay (Siemens), dated May 31, 2000).

This re-analysis is conservative in that the results are more severe (*i.e.*, larger ΔCPR) since the increased FCV position during the pump startup results in a faster influx of cold water into the reactor and consequently a larger reactivity-induced power swing.

Since the elements of the method described in the FSAR were updated in a manner to generate more conservative results and the new methodology is NRC-approved for this application, this MIN ED change does not represent a departure from a method of evaluation described in the FSAR.

GGNS 50.59 Safety Evaluation Number

SE 2003-0002-R00

I. OVERVIEW / SIGNATURES

OSRC ~~OSCR~~ MEETING #028/03
DATE: 8/21/2003
CJ 10/21/03

Facility: Grand Gulf Nuclear Station

Document Reviewed: Calculation XC-Q1N11-94004 Change/Rev. 0

System Designator(s)/Description: N11

Description of Proposed Change

This calculation updates the offsite and control room doses from a main steam line break outside containment.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

- ☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.
- ☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2______. (Insert item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, IV, and V required
☒	50.59 EVALUATION (#: 2003-0002 R.0.)	Sections I, II, III, IV, and VI required

Preparer: G.E. Broadbent / G.E. Broadbent / EOI / Engineering / 7/1/03
Name (print) / Signature / Company / Department / Date

Reviewer: [signature] / ENS / NE-SA / 7-01-03
Name (print) / Signature / Company / Department / Date

OSRC: [signature] 10/20/03 Jerry C Roberts
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name: **Scope of Assistance:**

CJ 10/21/03

X	QA RECORD B14.33
	~~RT = A7.01~~
	~~NON QA RECORD~~
	INITIAL: CJN
	NUMBER of PA: 12
	DATE: 10-20-03

RELATED DOCUMENT NUMBER = SE 2003-0002

II. SCREENING

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?**

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☒	☐	LDC 2003-059
TS Bases	☐	☒	
Technical Requirements Manual	☐	☒	
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Reports[1]	☐	☒	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2]	☐	☒	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculations Manual	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ Yes ☒ No
   If "yes," perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI.

3. **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation?** ☐ Yes ☐ No ☒ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

[1] If "YES," see Section 5.1.4.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

B. Basis

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

This calculation updates the offsite and control room doses from a main steam line break outside containment. The requirements in the Tech Specs, TRM, and Operating License are inputs to the analysis and are not affected by the calculation. NRC Security Orders are also not impacted. The TS Bases do not describe the radiological consequences of this event. The COLR is unaffected since this event is bounded by the recirc line break which is addressed by the current limits in the COLR. This event is described in SAR Section 15.6.4 to demonstrate that the applicable regulatory acceptance criteria are satisfied. This event is not affected by the QA manual, Emergency plan, Fire Protection program, or ODCM.

C. References

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. NOTE: **Ensure that electronic and manual searches are performed using controlled copies of documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
FSAR, Operating License Manual	"steamline break", "radiological dose"

LBDs/Documents reviewed manually:

D. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines.)

☐ Yes
☒ No

If "Yes," list the required changes.

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☐ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) ________ ☐ No

Name of Security Plan reviewer (print) / Signature / Date

V. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A, above.

A. Check the applicable boxes below. If any of the boxes are checked, a 50.59 Evaluation is not required. If none of the boxes are checked, perform a 50.59 Evaluation in accordance with Section VI. Provide supporting documentation or references as appropriate.

- ☐ The proposed activity meets all of the following criteria regarding design function per Section 5.6.1.1:

  The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; AND

  The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; AND

  The proposed activity does not adversely affect a method of evaluation that demonstrates intended design function(s) of an SSC described in the FSAR will be accomplished.

- ☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed activity already exists per Section 5.6.1.2. Reference 50.59 Evaluation # ____________ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

- ☐ The NRC has approved the proposed activity or portions thereof per Section 5.6.1.3. Reference: ______________________

- ☐ The proposed activity is controlled by another regulation per Section 5.6.1.4.

B. Basis

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10CFR50.59 Review Program Guidelines for guidance.

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VI. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

Calculation XC-Q1N11-94004, Rev. 0 updates the offsite and control room radiological doses associated with a main steam line break outside containment. This 50.59 evaluation addresses the changes made to the inputs of the analysis and a new dose methodology. The input changes include higher reactor coolant source term concentrations and updated dose conversion factors and diffusion factors (χ/Qs), which have been-approved by the NRC in association with the alternative source term licensing effort. This calculation was performed with a new dose methodology called RAPTOR, which has been determined to be essentially the same as previous NRC-approved methods via benchmark evaluations.

Reason for proposed Change:

A calculation revision is necessary to correct the deficiencies identified in CR-GG-2003-01876, and to include revised atmospheric diffusion factors as tracked under LO-GLO-2001-00034, CA-12. As described in CR-GG-2003-01876, the original Bechtel calculation was found to be deficient in that it did not (1) address the control room dose although GDC-19 reported explicit regulatory acceptance criteria for the control room doses for accidents, nor (2) consider this worst case iodine spiking scenario (4 uCi/g DE I-131) by only evaluating the equilibrium iodine case (0.2 uCi/g DE I-131).

50.59 Evaluation summary and conclusions

This 50.59 Evaluation concludes that, based on a comparison to the original Bechtel calculation, this revised calculation does not increase the consequences of an accident previously evaluated in the FSAR. The computer code and inputs applied in the analysis were found to not be a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses. Since this calculation does not impact any SSC or plant procedure, this change does not impact fission product barriers, or increase the frequency of occurrence of accidents or SSC malfunctions previously evaluated in the FSAR, or create the possibility for an accident or malfunction of a different type than previously evaluated in the FSAR.

B. License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   This calculation updates the offsite and control room doses associated with a main steamline break outside containment and does not impact any SSC or plant procedure. Since this calculation only impacts the plant analysis, it will not impact the frequency of occurrence of an accident previously evaluated in the FSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   This calculation updates the plant dose analysis. No changes are being made to any SSC or plant procedure. Therefore, this calculation will not impact the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   The radiological consequences of a main steamline break outside containment are reported in SAR Section 15.6.4 as developed in Bechtel Calculation 5.3.27. As described in CR-GG-2003-01876, this Bechtel calculation was found to be deficient in that it did not (1) address the control room dose although GDC-19 reported explicit regulatory acceptance criteria for the control room doses for accidents, nor (2) consider this worst case iodine spiking scenario (4 uCi/g DE I-131) by only evaluating the equilibrium iodine case (0.2 uCi/g DE I-131). To assess whether this change represents a significant increase in consequences, the new dose results are compared to the original Bechtel results in the following table.

Equilibrium Iodine Case (0.2 uCi/g DE I-131)

	Exclusion Area Boundary			Control Room		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
Bechtel Calculation 5.3.27	5.82	5.5E-2		Not Calc'd.	Not Calc'd.	
Updated Calculation XC-Q1J11-94004, R.0	2.43	3.8E-2	1.19E-1	4.55	2.99E-3	1.53E-1

As shown above, the only available case for comparison is the equilibrium iodine case at the EAB location. The new calculation reports EAB thyroid and whole body doses of 2.43 and 3.8E-2 Rem, respectively which are less than the original Bechtel results of 5.82 and 5.5E-2 Rem, respectively. The only available direct comparison therefore shows a reduction in consequences. The analysis input changes that caused this dose reduction are addressed in Question #8 since they relate to the methodology.

As described in CR-GG-2003-01876, the original Bechtel analysis did not calculate a control room dose. This oversight on Bechtel's part does not mean the control room dose was zero, only it was not quantified. Based on the offsite comparison performed above, the control room would also be expected to see a similar trend, with a reduction in the calculated dose from the Bechtel calculation.

CR-GG-2003-01876 also reports that Bechtel did not evaluate another more-limiting case, in which iodine spiking is assumed to put the coolant iodine concentration at the maximum allowed Technical Specification limit of 4.0 µCi/g Dose Equivalent I-131. As shown in the following table, the new analysis demonstrates that this case meets the NRC acceptance criteria. The results from this more-limiting case with higher doses will be added to the FSAR representing a separate case from the equilibrium case currently reported in the FSAR. Based on the offsite comparison performed above, these results would be expected to be lower than the doses calculated by Bechtel, if Bechtel had generated them.

Iodine Spiking Case (4.0 uCi/g DE I-131)

	Exclusion Area Boundary			Control Room		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
Bechtel Calculation 5.3.27	Not Calc'd.	Not Calc'd.		Not Calc'd.	Not Calc'd.	
Updated Calculation XC-Q1J11-94004, R.0	47.7	7.21E-1	2.32	89.2	5.88E-2	3.01
10 CFR 50.67 Acceptance Limit			25			5.0

In summary, for the cases in which a comparison can be made, the new analysis reports lower doses than the previous Bechtel analysis. This trend would extend to the new dose point (control room) and the worse scenario (iodine spiking) such that the new doses would be less than the associated Bechtel results, if Bechtel had generated them. On these bases, the results of the new MSLB radiological analysis do not represent an increase in consequences of an accident previously evaluated in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

   BASIS:

   This calculation updates the radiological analysis of the main steam line break outside containment. Consequently, this change will not affect the consequences of a malfunction of a SSC important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

This calculation updates the plant dose analysis. No changes are being made to any SSC or plant procedures. Therefore, this calculation will not create the possibility for an accident of a different type than previously evaluated in the FSAR.

6. *Create a possibility for a malfunction of a structure, system, or component* important to safety with a different result than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS

This calculation updates the plant dose analysis. No changes are being made to any SSC or plant procedures. Therefore, this calculation will not create the possibility for a malfunction of a SSC important to safety with a different result than any previously analyzed in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered?

☐ Yes
☒ No

BASIS:

The fission product barriers potentially impacted by this change are the fuel cladding and reactor coolant system. This calculation updates the plant dose analysis and makes no changes to the RCS, MSIV closure timing, or plant procedures. On these bases, this calculation will not result in *a design basis limit for a fission product barrier as described in the FSAR being exceeded* or altered.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses? ☐ Yes ☒ No

BASIS:

Methodology

The dose analysis methodology is not described in the FSAR in any detail. The computer code originally applied by Bechtel in the analysis of this event was called @PUFF. The GGNS computer code applied for the updated calculation is RAPTOR. As documented in Engineering Report GG-SA-2003-0002, RAPTOR has been benchmarked to the NRC's RADTRAD code and to the TRANSACT computer code used for the GGNS alternative source term (AST) analyses. Since RAPTOR has been shown to provide results that are essentially the same as another methodology previously accepted by the NRC, the use of the RAPTOR computer code is not a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.

Elements of the Methodology

As discussed in the response to Question 3, the results of the dose analysis decreased with the new analysis. Per the EOI 50.59 guidelines, this change could be construed as "non-conservative" such that margin to the acceptance criteria is gained via the re-analysis. However, the dose reduction is due to changes to the code inputs as a result of the AST analyses, which were NRC-approved in GNRI-2001/00032.

The three significant differences that resulted in the reduced EAB dose results are the (i) FGR-11 basis for reactor coolant iodine source terms, (ii) FGR-11 dose conversion factors, and (iii) updated diffusion factors. Each of these inputs were specifically submitted by GGNS and reviewed and approved by the NRC in association with the AST licensing efforts. The new calculation merely updates the analysis based on the NRC-approved inputs.

On these bases, although the results change in the non-conservative direction, this new analysis does not represent a departure from the method of evaluation described in the FSAR used in the safety analyses since it applies NRC-approved changes to the elements of the analysis.

GGNS 50.59 Safety Evaluation Number

SE 2004-0001-R00

I. OVERVIEW / SIGNATURES

Facility: GGNS

Document Reviewed: Core Operating Limits Report (COLR) **Change/Rev.** LDC 2004-009

System Designator(s)/Description: J11

Description of Proposed Change

This evaluation addresses the Cycle 14 reload changes and operation of the Cycle 14 reload core as given in the Core Operating Limits Report (COLR).

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

- ☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.
- ☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2______.
(Insert item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): **(Only the sections indicated must be included in the Review.)**

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION 2.19.04 CJ Moore	Sections I, II, III, IV, and V required
☑	50.59 EVALUATION (#: ~~SE-0001-2004~~)	Sections I, II, III, IV, and VI required

SE-2004-0001-R00

Preparer: Guy B. Spikes/ Guy B. Spikes /EOI/Nucl. Eng. – SA/ 2-4-04
Name (print) / Signature / Company / Department / Date

Reviewer: Scott Stanchfield / Scott Stanchfield / ENS / Nuc Eng-SA / 2/4/04
Name (print) / Signature / Company / Department / Date

OSRC: M. A. Krupa 2/19/04 OSRC MTG #006-2004
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name:	Scope of Assistance:
J. A. Elam (Central Engineering BWR Fuels)	Core design and neutronic input
D. L. Smith (Central Engineering BWR Fuels)	Fuel mechanical input
J. P. Head (Central Engineering BWR Fuels)	Core stability and hydraulic compatibility input
G. W. Smith (GGNS-PSA)	EOP Input

II. SCREENING

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?**

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☑	
TS	☐	☑	
NRC Orders	☐	☑	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☑	☐	LDC 2004-010
TS Bases	☐	☑	
Technical Requirements Manual	☑	☐	LDC 2004-012
Core Operating Limits Report	☑	☐	LDC 2004-009
NRC Safety Evaluation Reports[1]	☐	☑	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☑	
Emergency Plan[2]	☐	☑	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	☑	
Offsite Dose Calculations Manual	☐	☑	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ Yes ☑ No
   If "yes," perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI.

3. **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation?** ☐ Yes ☐ No ☑ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

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[1] If "YES," see Section 5.1.4.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

B. Basis

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

The current MCPR Safety Limit has been shown to be applicable to the Cycle 14 core. As such, Tech Spec 2.1.1.2 does not need to be revised.

Tech Spec 3.4.11 includes exposure-dependent P/T limit curves. The cumulative reactor vessel service life (EFPY) is expected to increase beyond the applicability of the P/T limit curve currently in use (≤16 EFPY) approximately 5 months after BOC14. The TRM will need to be updated to reflect the change in applicable P/T limit curve in TS 3.4.11.

There are no other Tech Spec, Bases, or TRM changes required for Cycle 14 startup and initial operation. There are no NRC orders applicable to the Cycle 14 reload campaign.

The Cycle 14 core will contain FANP 9x9-5 reinsert fuel bundles last used in Cycle 11 and the core characteristics and response will also be somewhat different than currently described in the FSAR. As such, Cycle 14 analyses have been performed for the new core and the FSAR will need to be updated appropriately as will the COLR.

The Cycle 14 core design and operation will not affect the OCDM, QAPM, Emergency Plan, Security Plan, Fire Protection Program, or any NRC SERs.

C. References

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. **NOTE: Ensure that electronic and manual searches are performed using controlled copies of documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
OLM, FSAR, TS Bases, TRM	Fuel, reload, channel, COLR, P/T

LBDs/Documents reviewed manually:

D. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines.)

☑ Yes
☐ No

If "Yes," list the required changes.

An acceptable final core loading. For a final core loading not exactly as provided in FAB03-1155, an evaluation of the as loaded core would need to be performed to ensure that the Cycle 14 reload analyses would continue to be acceptable.

Revise the TRM prior to BOC14+5 months to reflect change in applicable TS 3.4.11 P/T limit curve (LCTS 35421)

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☑	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☑	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☑	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☑	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☑	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☑	Discharge any chemicals new or different from that previously discharged?
7.	☐	☑	Change the design or operation of the intake or discharge structures?
8.	☐	☑	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☑	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☑	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☑	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☑	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☐	☑	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☑	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☑	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☑	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☑	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☑	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☑	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☑	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☑	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☑	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☑	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☑	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☑	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☐ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) ________ ☐ No

____

Name of Security Plan reviewer (print) / Signature / Date

VI. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

This safety evaluation assesses the reload-related changes associated with Cycle 14 operation as presented in the Core Operating Limits Report (COLR) located in the Operating License Manual (OLM). Cycle 14 has been designed for 538 Effective Full Power Days with a core consisting of 244 fresh ATRIUM-10 assemblies, 239 once-burnt ATRIUM-10 assemblies, 204 twice-burnt ATRIUM-10 assemblies, 28 thrice burnt GE11 assemblies, and 85 previously loaded twice-burnt FANP 9x9-5 assemblies. Eight of the fresh ATRIUM-10 assemblies are supplied with lead fuel channels manufactured by Kobe Steel Ltd. There are no TS or TS Bases changes required to operate with this new core, however, the FSAR does require updates. A TRM change will also be needed later during Cycle 14 to change the applicable TS 3.4.11 P/T limit curve. The Cycle 14 core has been designed and analyzed for a rated thermal power of 3898 MWt. Attachment 1 provides a detailed description of the Cycle 14 reload analysis and the issues considered in this evaluation. Increased (abnormal) channel bow described in CR-GGN-2002-01810 has been explicitly considered in the Cycle 14 reload analysis for the GE11 and potentially susceptible high exposure ATRIUM-10 fuel types using the advanced (thick-thin) channel design.

Reason for proposed Change:

Cycle 14 operation will require new core operating limits and the Core Operating Limits Report has been revised to include these new limits. These limits include flow-, power-, and exposure-dependent LHGR, MAPLHGR, and MCPR limits.

50.59 Evaluation summary and conclusions

The Cycle 14 core configuration and operation has been evaluated with respect to mechanical, neutronic, thermal-hydraulic, dose, thermal performance, and methods considerations for GGNS. This evaluation concludes that the reload-related changes associated with Cycle 14 operation does not require NRC review.

B. License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☑ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☑ No

BASIS:

The Cycle 14 core loading and cycle operation will not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR. The precursors to these events are independent of the core design and the frequency classifications reported in FSAR Chapter 15 are unaffected by the core parameters. The following considerations support this conclusion.

Mechanical

The ATRIUM-10 mechanical design has been reviewed for use at Grand Gulf. No unusual failure modes or increased failure frequency have been identified for this fuel design. This is the third reload at GGNS with ATRIUM-10 fuel and this fuel design has accumulated significant problem-free operational experience at other plants. The FANP 9x9-5 fuel type was introduced at GGNS in Cycle 5 and was last used in Cycle 11. The mechanical design of this fuel has been reviewed for use at GGNS in the mixed (ATRIUM-10, GE11, 9x9-5) Cycle 14 core. The Cycle 14 bundles will operate within the power history assumptions in the fuel mechanical analyses and will result in exposures within the analyzed burnup limits of the ATRIUM-10, GE11, and 9x9-5 mechanical designs. All design criteria for the GE11 bundles have been shown to meet their respective limits including those that will be irradiated for a fourth cycle.

Nuclear

The neutronic characteristics of the Cycle 14 9x9-5, GE11 and ATRIUM-10 mixed core design have been considered in the safety analysis. Adequate shutdown margin has been predicted by analysis and will be confirmed during startup tests. In addition, the hold-down capability of the standby liquid control system and the subcriticality of Cycle 14 fuel in the spent fuel storage racks have been confirmed. Therefore, the probability of inadvertent criticality has not been increased by the introduction of the Cycle 14 reload fuel.

Thermal-Hydraulic

FANP's modeling of the GE fuel and the thermal-hydraulic compatibility of the ATRIUM-10, GE11, and 9x9-5 fuel have been reviewed and found acceptable. To accurately model the GE11 bundle hydraulics, a Cycle 11 GE11 bundle was shipped to FANP for hydraulic testing in their hydraulic test facility. Analyses have been performed to demonstrate that Cycle 14 meets all Enhanced-1A stability performance criteria without changes to the E1A hardware or power-flow map region boundaries. Therefore, the probability of thermal-hydraulic instabilities has not increased.

Analyzed Events
The probability of the occurrence of anticipated operational events is not dependent on the core configuration. No changes to the plant design are required for the Cycle 14 core. The Cycle 14 core loading will not affect the precursors to any of the Chapter 15 events. The probability of an analyzed event therefore has not increased.

As described in FSAR Section 15A.6.5.3, the Control Rod Drop Accident (CRDA) results from a failure of the control rod-to-drive mechanism coupling after the control rod becomes stuck in its fully inserted position. Although an increased channel bow condition could result in increased friction between the control blade and its corresponding fuel assemblies, analyses have shown that there would not be sufficient friction to result in a mechanical failure of the coupling. Additionally, the control rod drive mechanism would not produce enough force to result in a mechanical failure of the coupling even if the channel bow was so severe that the assemblies would preclude blade movement. As such, channel bow is not considered a precursor to the CRDA, and any increased bow associated with the high exposure ATRIUM-10 bundles (the GE11 bundles are loaded in unrodded locations) would not increase the probability of this event.

On these bases, the probability of occurrence of accidents previously identified in the FSAR is not increased for the Cycle 14 core with increased channel bow.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☑ No

BASIS:

The Cycle 14 ATRIUM-10 reload fuel and 9x9-5 reinsert fuel are designs that have been shown to be mechanically, neutronically, and thermal-hydraulically compatible with the co-resident GE11 fuel. No plant modifications are required to accommodate the new core design and the only additional loads placed on plant equipment would be potential friction between the control blades and excessively bowed ATRIUM-10 bundles (the GE11 bundles are loaded in unrodded locations). Based on previous experience with bowed fuel at GGNS and Clinton Power Station, this increased friction is not expected to impact scram times. Technical Specification scram time testing and appropriate channel bow surveillances will be performed during Cycle 14. These actions would identify any potential scram time or other impacts and the appropriate actions would be taken. Additionally, this increased friction would not be sufficient to provide any failures associated with the control blades or the control blade drive system.

A conservative vessel overpressurization analysis has been performed, which shows that the vessel pressure limit is not exceeded. The precursors to any malfunction of equipment important to safety are not affected by this change.

Therefore, there is not more than a minimal increase in the likelihood of an occurrence of a malfunction of a SSC important to safety previously evaluated in the FSAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☑ No

BASIS:

As reported in Attachment 1, the acceptance criteria reported in FSAR Section 15.0.3.1 and the Technical Specifications are satisfied for each event classification. Core operating limits have been developed to ensure that moderate frequency events do not violate the MCPR safety limit or fuel cladding strain limits. The consequences of infrequent events have been shown to meet the appropriate acceptance criteria while the individual acceptance criteria for the limiting faults have been demonstrated to be satisfied. The following considerations support these conclusions.

Moderate Frequency Events

The Cycle 14 core operating limits have been developed with NRC-approved methodologies such that the MCPR safety limit and the fuel cladding strain limit will not be violated by any analyzed moderate frequency transient initiated from any statepoint available to GGNS. As such, no fuel failures are expected to result from any moderate frequency event. These analyses considered GGNS-specific operational modes such as MEOD, SLO, FHOOS, and EOC-RPT inoperable. These core operating limits consist of MCPR, MAPLHGR and LHGR curves that are functions of flow, power, and exposure. These limits consider conservative channel bow assumptions for the GE11 and highly exposed ATRIUM-10 fuel that bound the increased bow previously observed in the GE11 fuel.

These core operating limits will be incorporated into the core monitoring system, however, as with previous cycles, the GE11 operating limits illustrated in the COLR will differ somewhat from those limits in the core monitoring system. GE has generated pellet-based exposure-dependent LHGR and lattice-based exposure-dependent MAPLHGR limits for their GE11 fuel bundles; however, for competitive reasons, GE has designated the limits for only the most-limiting or least-limiting lattices as non-proprietary. The COLR, which is submitted to the NRC for information only, will therefore report the non-proprietary limits for the most limiting enriched lattice at each exposure for reference purposes only, and refer to the appropriate GE proprietary document for the actual limits. The actual LHGR and MAPLHGR limits for most lattices will be higher than the COLR reference curves and it is recognized that most lattices could operate at higher values than the reference limits. However, this is acceptable since Technical Specifications 3.2.1 and 3.2.3 require that all APLHGRs and LHGRs, respectively be less than or equal to the limits specified in the COLR, and the COLR refers to the proprietary GE document for the actual limits.

Infrequent Events

The consequences of the limiting infrequent events have been evaluated and shown to meet their respective acceptance criteria. These events include the pressure regulator failure downscale, misplaced (*i.e.,* misoriented and mislocated) bundle, and single loop operation pump seizure accidents. Radiological analyses using the alternative source term (AST) have been performed to ensure that these events will not result in offsite or control room doses greater than their respective acceptance criteria. These evaluations consider conservative channel bow assumptions for the GE11 and highly exposed ATRIUM-10 fuel that bound the increased bow previously observed in the GE11 fuel.

Limiting Faults

The limiting faults at GGNS include the fuel handling accident, the control rod drop accident, and the design basis LOCA. The radiological analyses for these events have been developed as part of the GGNS AST effort and bound the Cycle 14 core parameters. For the LOCA, MAPLHGR operating limits and single-loop multipliers have been developed for the Cycle 14 core configuration such that the requirements of 10CFR50.46 are satisfied. The containment response for the Cycle 14 core was found to be bounded by previous cycles as is the hydrogen analysis. The seismic/LOCA response of the Cycle 14 core has been confirmed to be acceptable. The Cycle 14 core design results in minor changes to two EP parameters (Mclad and Mfuel); however, the existing EP's remain applicable to Cycle 14.

Therefore, the proposed change does not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☑ No

BASIS:

The Cycle 14 ATRIUM-10 reload fuel and 9x9-5 reinsert fuel are designs that have been shown to be compatible with the co-resident GE11 fuel type. The malfunctions of key plant components are analyzed as part of the reload process with the results reported in various sections of the FSAR. The consequences of these malfunctions have been shown to meet their respective acceptance criteria.

Therefore, Cycle 14 operation will not result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR? ☐ Yes ☑ No

BASIS:

The Cycle 14 ATRIUM-10 reload fuel and FANP 9x9-5 reinsert fuel are similar to and compatible with the GE11 fuel that was inserted in previous cycles. The details of this design have been specifically considered in the safety analysis and the core monitoring system. No plant modifications are required to accommodate the new core design or Cycle 14 operation. The GGNS Cycle 14 fuel types have been approved for the Cycle 14 reactor chemistry conditions.

The GGNS operational parameters (water chemistry requirements, spectral-shift core designs, and MEOD rod-lines) have been reviewed and are not expected to result in unusual crud buildup like that observed on the high-power GE11 bundles at River Bend. Inspection of a high-power, once-burnt representative fuel bundle during GGNS RF10 has confirmed that the high-power GGNS Cycle 10 fuel bundles have no unusual crud buildup.

Therefore, Cycle 14 operation will not create a possibility for an accident of a different type than any previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR? ☐ Yes ☑ No

BASIS

The Cycle 14 ATRIUM-10 reload fuel and 9x9-5 reinsert fuel are designs that have been shown to be mechanically, neutronically, and thermal-hydraulically compatible with the co-resident GE11 fuel. The ATRIUM-10 and 9x9-5 fuel will not introduce any adverse flow distribution effects such as preferential flow through the ATRIUM-10 bundles that may negatively impact the GE 11 bundles. No plant modifications are required to accommodate the new core design and no additional loads will be imposed on any existing equipment. The GE11, ATRIUM-10 and 9x9-5 bundles provide sufficient clearance for proper control blade operation and allow sufficient bypass flow in the bypass region to provide adequate cooling for control blades and in-core detectors. There are no special operational considerations associated with the Cycle 14 core other than those associated with the increased bow condition. The higher friction expected between the control blades and any ATRIUM-10 bundles experiencing increased bow would not be sufficient to cause a failure of the fuel bundle, control blade, or control rod drive coupling. Appropriate channel bow surveillances will be performed during Cycle 14 to monitor this condition. The GE11 bundles are loaded in unrodded locations on the core periphery and increased bow in these bundles will not affect control rod movement.

Therefore, Cycle 14 operation will not create the possibility for a malfunction of an SSC important to safety with a different result than previously evaluated in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered? ☐ Yes ☑ No

BASIS:

Mechanical analyses have been performed to ensure that all fuel in the Cycle 14 core meet the mechanical design limits for steady-state operation as well as transient conditions including fatigue damage, creep collapse, corrosion, fuel rod internal pressure, rod bow, internal pressure, etc. Additionally, no Cycle 14 fuel will exceed the applicable burn-up or residence time limits.

Core operating limits have been developed using NRC approved codes in order to ensure that the Cycle 14 fuel will not exceed the MCPR safety limits for steady-state operation and anticipated operation occurrences. Similarly, operating limits have been developed to ensure that the Cycle 14 fuel will not exceed the 1% cladding strain limit or experience core-wide fuel melt during steady-state operation or AOO's. Although some vessel blowdown to the suppression pool may be experienced during some AOO's, which would increase the suppression pool temperature, the bulk containment pressure increase is negligible and would not exceed the design limit.

As described in Attachment 1, a bounding pressurization event with a failure of the direct scram has been analyzed for Cycle 14 to ensure compliance with ASME code requirements. This analysis indicates that the vessel pressure safety limit is not exceeded for Cycle 14.

A design basis limit for the peak fuel enthalpy of 280 cal/gm has been established for the control rod drop accident (CRDA) to preclude significant fuel cladding failure such that core geometry and cooling may be impacted. The CRDA has been evaluated for Cycle 14. This evaluation considers all potential withdrawal sequences and concludes that a CRDA will not exceed the 280 cal/gm peak enthalpy limit. Since this accident is a localized event and the peak enthalpy does not exceed 280 cal/gm, there is no impact on the vessel or containment pressures. As such their respective limits are not exceeded.

10CFR50.46 provides limits associated with the ECCS performance analysis (LOCA analysis). Two such limits are Peak Clad Temperature (PCT) and local clad oxidation. Although these limits are not subject to 10CFR50.59, they are discussed in this evaluation for completeness. Grand Gulf specific analyses have been performed for ATRIUM-10, 9x9-5, and GE11 fuel in accordance with 10CFR50.46. These analyses, which are applicable to Cycle 14, show that the PCT and local oxidation are well below the limits set forth in 10CFR50.46. These analyses also show that the core-wide metal water reaction, which is used to evaluate compliance with the containment design limit, is less than the 10CFR50.46 limit. The remainder of the existing containment analysis associated with LOCA events is applicable to Cycle 14 as described in Attachment 1. As such, the containment pressure design limit will not be exceeded in Cycle 14.

An ATWS evaluation has also been performed for Cycle 14. As described in Attachment 1, the resulting vessel pressure remains below the ASME emergency vessel pressure limit of 1500 psig and the temperature response used in the existing ATWS containment analysis is applicable to Cycle 14. Thus, the containment pressure design limit will not be exceeded for the ATWS event.

Additional evaluations have been performed for Cycle 14 including Appendix R (Fire Protection), hydrogen analyses, and SBO as described in Attachment 1. These evaluations show that the existing evaluations are applicable to Cycle 14 and that their respective limits are not exceeded.

Therefore, Cycle 14 operation will not result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?

☐ Yes
☑ No

BASIS:

The reload analyses performed by the fuel vendor utilized NRC approved methods as listed in Technical Specification 5.6.5 and described throughout the FSAR. Since the MCPR operating limits developed for the ATRIUM-10 fuel are conservatively applied to the GE11 and 9x9-5 fuel types, the references listed in Technical Specification 5.6.5 need not include the supporting NRC approved methodologies for these non-limiting fuel types. As described in Attachment 1, uncertainty applied in the Safety Limit calculation associated with each of the equipment out of service combinations was calculated in accordance with Framatome-ANP's NRC approved methodology. All remaining GGNS evaluations currently described in the FSAR have been shown to be applicable to Cycle 14. As such, no new methods were used. Finally, the GGNS EP calculation has been updated to consider the minor changes to two fuel parameters. This revision did not incorporate any new or different methods.

Therefore, Cycle 14 operation will not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.

GGNS 50.59 Safety Evaluation Number

SE 2004-0001-R01

Entergy	EN-S NUCLEAR MANAGEMENT MANUAL	QUALITY RELATED ADMINISTRATIVE	LI-101	Revision 3		
		INFORMATION USE				
ATTACHMENT 9.1		50.59 REVIEW FORM	Page	1	of	13

I. OVERVIEW / SIGNATURES

Facility: GGNS

Document Reviewed: Core Operating Limits Report (COLR) **Change/Rev.** LDC 2004-009

System Designator(s)/Description: J11

Description of Proposed Change

This evaluation addresses the Cycle 14 reload changes and operation of the Cycle 14 reload core as given in the Core Operating Limits Report (COLR). Revision 1 evaluates the impact on the Cycle 14 core design and operation of the identification of fuel failures during RF13 and changes to the reference core loading to reflect the replacement of one failed once-burnt ATRIUM-10 fuel bundle.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

- ☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.
- ☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2______. (Insert item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, IV, and V required
☑	50.59 EVALUATION (#: SE-2004-0001-R. 1)	Sections I, II, III, IV, and VI required

Preparer: Guy B. Spikes/ Guy B. Spikes /EOI/Nucl. Eng. – SA/ 3-16-04
Name (print) / Signature / Company / Department / Date

Reviewer: Scott Stanchfield / Scott Stanchfield / ENS / NE-SA / 3/17/04
Name (print) / Signature / Company / Department / Date

OSRC #010-2004

OSRC: M. A Krupa 3-17-04
Chairman's Name (print) / Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name:	Scope of Assistance:
J. A. Elam (Central Engineering BWR Fuels)	Core design and neutronic input
D. L. Smith (Central Engineering BWR Fuels)	Fuel mechanical input
J. P. Head (Central Engineering BWR Fuels)	Core stability and hydraulic compatibility input
G. W. Smith (GGNS-PSA)	EOP input

II. SCREENING

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?**

Operating License	YES	NO	CHANGE # and/or SECTIONS IMPACTED
Operating License	☐	☑	
TS	☐	☑	
NRC Orders	☐	☑	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
FSAR	☑	☐	LDC 2004-010
TS Bases	☐	☑	
Technical Requirements Manual	☑	☐	LDC 2004-012
Core Operating Limits Report	☑	☐	LDC 2004-009
NRC Safety Evaluation Reports[1]	☐	☑	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # (if applicable) and/or SECTIONS IMPACTED
Quality Assurance Program Manual[2]	☐	☑	
Emergency Plan[2]	☐	☑	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	☑	
Offsite Dose Calculations Manual	☐	☑	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ Yes ☑ No
   If "yes," perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI.

3. **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation?** ☐ Yes ☐ No ☑ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

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[1] If "YES," see Section 5.1.4.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

B. Basis

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

The current MCPR Safety Limit has been shown to be applicable to the Cycle 14 core. As such, Tech Spec 2.1.1.2 does not need to be revised.

Tech Spec 3.4.11 includes exposure-dependent P/T limit curves. The cumulative reactor vessel service life (EFPY) is expected to increase beyond the applicability of the P/T limit curve currently in use (≤16 EFPY) approximately 5 months after BOC14. The TRM will need to be updated to reflect the change in applicable P/T limit curve in TS 3.4.11.

There are no other Tech Spec, Bases, or TRM changes required for Cycle 14 startup and initial operation. There are no NRC orders applicable to the Cycle 14 reload campaign.

The Cycle 14 core will contain FANP 9x9-5 reinsert fuel bundles last used in Cycle 11 and the core characteristics and response will also be somewhat different than currently described in the FSAR. As such, Cycle 14 analyses have been performed for the new core and the FSAR will need to be updated appropriately as will the COLR.

The Cycle 14 core design and operation will not affect the ODCM, QAPM, Emergency Plan, Security Plan, Fire Protection Program, or any NRC SERs.

C. References

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. **NOTE: Ensure that electronic and manual searches are performed using controlled copies of documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
OLM, FSAR, TS Bases, TRM	Fuel, reload, channel, COLR, P/T

LBDs/Documents reviewed manually:

D. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines.)

☑ Yes
☐ No

If "Yes," list the required changes.

Revise the TRM prior to BOC14+5 months to reflect change in applicable TS 3.4.11 P/T limit curve (LCTS 35421)

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☑	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☑	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☑	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☑	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☑	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☑	Discharge any chemicals new or different from that previously discharged?
7.	☐	☑	Change the design or operation of the intake or discharge structures?
8.	☐	☑	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☑	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☑	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1].
11.	☐	☑	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☑	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☐	☑	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☑	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☑	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☑	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☑	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☑	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☑	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☑	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☑	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☑	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☑	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☑	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☑	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☐ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) ________ ☐ No

____

Name of Security Plan reviewer (print) / Signature / Date

VI. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

This safety evaluation assesses the reload-related changes associated with Cycle 14 operation as presented in the Core Operating Limits Report (COLR) located in the Operating License Manual (OLM). Cycle 14 has been designed for 538 Effective Full Power Days with a core consisting of 244 fresh ATRIUM-10 assemblies, 239 once-burnt ATRIUM-10 assemblies, 204 twice-burnt ATRIUM-10 assemblies, 28 thrice burnt GE11 assemblies, and 85 previously loaded twice-burnt FANP 9x9-5 assemblies. Eight of the fresh ATRIUM-10 assemblies are supplied with lead fuel channels manufactured by Kobe Steel Ltd. There are no TS or TS Bases changes required to operate with this new core, however, the FSAR does require updates. A TRM change will also be needed later during Cycle 14 to change the applicable TS 3.4.11 P/T limit curve. The Cycle 14 core has been designed and analyzed for a rated thermal power of 3898 MWt. Attachment 1 provides a detailed description of the Cycle 14 reload analysis and the issues considered in this evaluation. Increased (abnormal) channel bow described in CR-GGN-2002-01810 has been explicitly considered in the Cycle 14 reload analysis for the GE11 and potentially susceptible high exposure ATRIUM-10 fuel types using the advanced (thick-thin) channel design. Revision 1 incorporates the results of evaluations associated with a revision to the reference core loading pattern. The original core design considered replacement of one assumed failed once-burnt ATRIUM-10 fuel bundle. The revised core loading reflects the identification of that failed bundle and its replacement with another non-failed once-burnt ATRIUM-10 bundle. The revised reference core loading is provided in FAB04-072 and the Cycle 14 core has been verified to be correctly loaded per 17-S-02-108-RF13-031504.

Reason for proposed Change:

Cycle 14 operation will require new core operating limits and the Core Operating Limits Report has been revised to include these new limits. These limits include flow-, power-, and exposure-dependent LHGR, MAPLHGR, and MCPR limits.

50.59 Evaluation summary and conclusions

The Cycle 14 core configuration and operation has been evaluated with respect to mechanical, neutronic, thermal-hydraulic, dose, thermal performance, and methods considerations for GGNS. This evaluation concludes that the reload-related changes associated with Cycle 14 operation does not require NRC review.

B. License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.

☐ Yes
☑ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR?

   ☐ Yes
   ☑ No

BASIS:

The Cycle 14 core loading and cycle operation will not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR. The precursors to these events are independent of the core design and the frequency classifications reported in FSAR Chapter 15 are unaffected by the core parameters. The following considerations support this conclusion.

Mechanical

The ATRIUM-10 mechanical design has been reviewed for use at Grand Gulf. No unusual failure modes or increased failure frequency have been identified for this fuel design. This is the third reload at GGNS with ATRIUM-10 fuel and this fuel design has accumulated significant problem-free operational experience at other plants. The FANP 9x9-5 fuel type was introduced at GGNS in Cycle 5 and was last used in Cycle 11. The mechanical design of this fuel has been reviewed for use at GGNS in the mixed (ATRIUM-10, GE11, 9x9-5) Cycle 14 core. The Cycle 14 bundles will operate within the power history assumptions in the fuel mechanical analyses and will result in exposures within the analyzed burnup limits of the ATRIUM-10, GE11, and 9x9-5 mechanical designs. All design criteria for the GE11 bundles have been shown to meet their respective limits including those that will be irradiated for a fourth cycle.

Nuclear

The neutronic characteristics of the Cycle 14 9x9-5, GE11 and ATRIUM-10 mixed core design have been considered in the safety analysis. Adequate shutdown margin has been predicted by analysis and will be confirmed during startup tests. In addition, the hold-down capability of the standby liquid control system and the subcriticality of Cycle 14 fuel in the spent fuel storage racks have been confirmed. Therefore, the probability of inadvertent criticality has not been increased by the introduction of the Cycle 14 reload fuel.

Thermal-Hydraulic

FANP's modeling of the GE fuel and the thermal-hydraulic compatibility of the ATRIUM-10, GE11, and 9x9-5 fuel have been reviewed and found acceptable. To accurately model the GE11 bundle hydraulics, a Cycle 11 GE11 bundle was shipped to FANP for hydraulic testing in their hydraulic test facility. Analyses have been performed to demonstrate that Cycle 14 meets all Enhanced-1A stability performance criteria without changes to the E1A hardware or power-flow map region boundaries. Therefore, the probability of thermal-hydraulic instabilities has not increased.

Analyzed Events
The probability of the occurrence of anticipated operational events is not dependent on the core configuration. No changes to the plant design are required for the Cycle 14 core. The Cycle 14 core loading will not affect the precursors to any of the Chapter 15 events. The probability of an analyzed event therefore has not increased.

As described in FSAR Section 15A.6.5.3, the Control Rod Drop Accident (CRDA) results from a failure of the control rod-to-drive mechanism coupling after the control rod becomes stuck in its fully inserted position. Although an increased channel bow condition could result in increased friction between the control blade and its corresponding fuel assemblies, analyses have shown that there would not be sufficient friction to result in a mechanical failure of the coupling. Additionally, the control rod drive mechanism would not produce enough force to result in a mechanical failure of the coupling even if the channel bow was so severe that the assemblies would preclude blade movement. As such, channel bow is not considered a precursor to the CRDA, and any increased bow associated with the high exposure ATRIUM-10 bundles (the GE11 bundles are loaded in unrodded locations) would not increase the probability of this event.

On these bases, the probability of occurrence of accidents previously identified in the FSAR is not increased for the Cycle 14 core with increased channel bow.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☑ No

BASIS:

The Cycle 14 ATRIUM-10 reload fuel and 9x9-5 reinsert fuel are designs that have been shown to be mechanically, neutronically, and thermal-hydraulically compatible with the co-resident GE11 fuel. No plant modifications are required to accommodate the new core design and the only additional loads placed on plant equipment would be potential friction between the control blades and excessively bowed ATRIUM-10 bundles (the GE11 bundles are loaded in unrodded locations). Based on previous experience with bowed fuel at GGNS and Clinton Power Station, this increased friction is not expected to impact scram times. Technical Specification scram time testing and appropriate channel bow surveillances will be performed during Cycle 14. These actions would identify any potential scram time or other impacts and the appropriate actions would be taken. Additionally, this increased friction would not be sufficient to provide any failures associated with the control blades or the control blade drive system.

A conservative vessel overpressurization analysis has been performed, which shows that the vessel pressure limit is not exceeded. The precursors to any malfunction of equipment important to safety are not affected by this change.

Therefore, there is not more than a minimal increase in the likelihood of an occurrence of a malfunction of a SSC important to safety previously evaluated in the FSAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☑ No

BASIS:

As reported in Attachment 1, the acceptance criteria reported in FSAR Section 15.0.3.1 and the Technical Specifications are satisfied for each event classification. Core operating limits have been developed to ensure that moderate frequency events do not violate the MCPR safety limit or fuel cladding strain limits. The consequences of infrequent events have been shown to meet the appropriate acceptance criteria while the individual acceptance criteria for the limiting faults have been demonstrated to be satisfied. The following considerations support these conclusions.

Moderate Frequency Events

The Cycle 14 core operating limits have been developed with NRC-approved methodologies such that the MCPR safety limit and the fuel cladding strain limit will not be violated by any analyzed moderate frequency transient initiated from any statepoint available to GGNS. As such, no fuel failures are expected to result from any moderate frequency event. These analyses considered GGNS-specific operational modes such as MEOD, SLO, FHOOS, and EOC-RPT inoperable. These core operating limits consist of MCPR, MAPLHGR and LHGR curves that are functions of flow, power, and exposure. These limits consider conservative channel bow assumptions for the GE11 and highly exposed ATRIUM-10 fuel that bound the increased bow previously observed in the GE11 fuel.

These core operating limits will be incorporated into the core monitoring system, however, as with previous cycles, the GE11 operating limits illustrated in the COLR will differ somewhat from those limits in the core monitoring system. GE has generated pellet-based exposure-dependent LHGR and lattice-based exposure-dependent MAPLHGR limits for their GE11 fuel bundles; however, for competitive reasons, GE has designated the limits for only the most-limiting or least-limiting lattices as non-proprietary. The COLR, which is submitted to the NRC for information only, will therefore report the non-proprietary limits for the most limiting enriched lattice at each exposure for reference purposes only, and refer to the appropriate GE proprietary document for the actual limits. The actual LHGR and MAPLHGR limits for most lattices will be higher than the COLR reference curves and it is recognized that most lattices could operate at higher values than the reference limits. However, this is acceptable since Technical Specifications 3.2.1 and 3.2.3 require that all APLHGRs and LHGRs, respectively be less than or equal to the limits specified in the COLR, and the COLR refers to the proprietary GE document for the actual limits.

Infrequent Events

The consequences of the limiting infrequent events have been evaluated and shown to meet their respective acceptance criteria. These events include the pressure regulator failure downscale, misplaced (*i.e.*, misoriented and mislocated) bundle, and single loop operation pump seizure accidents. Radiological analyses using the alternative source term (AST) have been performed to ensure that these events will not result in offsite or control room doses greater than their respective acceptance criteria. These evaluations consider conservative channel bow assumptions for the GE11 and highly exposed ATRIUM-10 fuel that bound the increased bow previously observed in the GE11 fuel.

Limiting Faults

The limiting faults at GGNS include the fuel handling accident, the control rod drop accident, and the design basis LOCA. The radiological analyses for these events have been developed as part of the GGNS AST effort and bound the Cycle 14 core parameters. For the LOCA, MAPLHGR operating limits and single-loop multipliers have been developed for the Cycle 14 core configuration such that the requirements of 10CFR50.46 are satisfied. The containment response for the Cycle 14 core was found to be bounded by previous cycles as is the hydrogen analysis. The seismic/LOCA response of the Cycle 14 core has been confirmed to be acceptable. The Cycle 14 core design results in minor changes to two EP parameters (Mclad and Mfuel); however, the existing EP's remain applicable to Cycle 14.

Therefore, the proposed change does not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?　☐ Yes　☑ No

BASIS:

The Cycle 14 ATRIUM-10 reload fuel and 9x9-5 reinsert fuel are designs that have been shown to be compatible with the co-resident GE11 fuel type. The malfunctions of key plant components are analyzed as part of the reload process with the results reported in various sections of the FSAR. The consequences of these malfunctions have been shown to meet their respective acceptance criteria.

Therefore, Cycle 14 operation will not result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR?　☐ Yes　☑ No

BASIS:

The Cycle 14 ATRIUM-10 reload fuel and FANP 9x9-5 reinsert fuel are similar to and compatible with the GE11 fuel that was inserted in previous cycles. The details of this design have been specifically considered in the safety analysis and the core monitoring system. No plant modifications are required to accommodate the new core design or Cycle 14 operation. The GGNS Cycle 14 fuel types have been approved for the Cycle 14 reactor chemistry conditions.

The GGNS operational parameters (water chemistry requirements, spectral-shift core designs, and MEOD rod-lines) have been reviewed and are not expected to result in unusual crud buildup like that observed on the high-power GE11 bundles at River Bend. Inspection of high-power, once- and twice-burnt representative fuel bundles during GGNS RF13 has confirmed that the high-power GGNS Cycle 13 fuel bundles have no unusual crud buildup. As described in Attachment 1, the fuel failures during Cycle 13 are not indicative of generic or core-wide fuel performance issues.

Therefore, Cycle 14 operation will not create a possibility for an accident of a different type than any previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR?

☐ Yes
☑ No

BASIS

The Cycle 14 ATRIUM-10 reload fuel and 9x9-5 reinsert fuel are designs that have been shown to be mechanically, neutronically, and thermal-hydraulically compatible with the co-resident GE11 fuel. The ATRIUM-10 and 9x9-5 fuel will not introduce any adverse flow distribution effects such as preferential flow through the ATRIUM-10 bundles that may negatively impact the GE 11 bundles. No plant modifications are required to accommodate the new core design and no additional loads will be imposed on any existing equipment. The GE11, ATRIUM-10 and 9x9-5 bundles provide sufficient clearance for proper control blade operation and allow sufficient bypass flow in the bypass region to provide adequate cooling for control blades and in-core detectors. There are no special operational considerations associated with the Cycle 14 core other than those associated with the increased bow condition. The higher friction expected between the control blades and any ATRIUM-10 bundles experiencing increased bow would not be sufficient to cause a failure of the fuel bundle, control blade, or control rod drive coupling. Appropriate channel bow surveillances will be performed during Cycle 14 to monitor this condition. The GE11 bundles are loaded in unrodded locations on the core periphery and increased bow in these bundles will not affect control rod movement.

Therefore, Cycle 14 operation will not create the possibility for a malfunction of an SSC important to safety with a different result than previously evaluated in the FSAR.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered?

☐ Yes
☑ No

BASIS:

Mechanical analyses have been performed to ensure that all fuel in the Cycle 14 core meet the mechanical design limits for steady-state operation as well as transient conditions including fatigue damage, creep collapse, corrosion, fuel rod internal pressure, rod bow, internal pressure, etc. Additionally, no Cycle 14 fuel will exceed the applicable burn-up or residence time limits.

Core operating limits have been developed using NRC approved codes in order to ensure that the Cycle 14 fuel will not exceed the MCPR safety limits for steady-state operation and anticipated operation occurrences. Similarly, operating limits have been developed to ensure that the Cycle 14 fuel will not exceed the 1% cladding strain limit or experience core-wide fuel melt during steady-state operation or AOO's. Although some vessel blowdown to the suppression pool may be experienced during some AOO's, which would increase the suppression pool temperature, the bulk containment pressure increase is negligible and would not exceed the design limit.

As described in Attachment 1, a bounding pressurization event with a failure of the direct scram has been analyzed for Cycle 14 to ensure compliance with ASME code requirements. This analysis indicates that the vessel pressure safety limit is not exceeded for Cycle 14.

A design basis limit for the peak fuel enthalpy of 280 cal/gm has been established for the control rod drop accident (CRDA) to preclude significant fuel cladding failure such that core geometry and cooling may be impacted. The CRDA has been evaluated for Cycle 14. This evaluation considers all potential withdrawal sequences and concludes that a CRDA will not exceed the 280 cal/gm peak enthalpy limit. Since this accident is a localized event and the peak enthalpy does not exceed 280 cal/gm, there is no impact on the vessel or containment pressures. As such their respective limits are not exceeded.

10CFR50.46 provides limits associated with the ECCS performance analysis (LOCA analysis). Two such limits are Peak Clad Temperature (PCT) and local clad oxidation. Although these limits are not subject to 10CFR50.59, they are discussed in this evaluation for completeness. Grand Gulf specific analyses have been performed for ATRIUM-10, 9x9-5, and GE11 fuel in accordance with 10CFR50.46. These analyses, which are applicable to Cycle 14, show that the PCT and local oxidation are well below the limits set forth in 10CFR50.46. These analyses also show that the core-wide metal water reaction, which is used to evaluate compliance with the containment design limit, is less than the 10CFR50.46 limit. The remainder of the existing containment analysis associated with LOCA events is applicable to Cycle 14 as described in Attachment 1. As such, the containment pressure design limit will not be exceeded in Cycle 14.

An ATWS evaluation has also been performed for Cycle 14. As described in Attachment 1, the resulting vessel pressure remains below the ASME emergency vessel pressure limit of 1500 psig and the temperature response used in the existing ATWS containment analysis is applicable to Cycle 14. Thus, the containment pressure design limit will not be exceeded for the ATWS event.

Additional evaluations have been performed for Cycle 14 including Appendix R (Fire Protection), hydrogen analyses, and SBO as described in Attachment 1. These evaluations show that the existing evaluations are applicable to Cycle 14 and that their respective limits are not exceeded.

Therefore, Cycle 14 operation will not result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?

☐ Yes
☑ No

BASIS:

The reload analyses performed by the fuel vendor utilized NRC approved methods as listed in Technical Specification 5.6.5 and described throughout the FSAR. Since the MCPR operating limits developed for the ATRIUM-10 fuel are conservatively applied to the GE11 and 9x9-5 fuel types, the references listed in Technical Specification 5.6.5 need not include the supporting NRC approved methodologies for these non-limiting fuel types. As described in Attachment 1, uncertainty applied in the Safety Limit calculation associated with each of the equipment out of service combinations was calculated in accordance with Framatome-ANP's NRC approved methodology. All remaining GGNS evaluations currently described in the FSAR have been shown to be applicable to Cycle 14. As such, no new methods were used. Finally, the GGNS EP calculation has been updated to consider the minor changes to two fuel parameters. This revision did not incorporate any new or different methods.

Therefore, Cycle 14 operation will not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.

GGNS 50.59 Safety Evaluation Number

SE 2004-0002-R00

I. OVERVIEW / SIGNATURES

Facility: Grand Gulf Nuclear Station

Document Reviewed: Temp Alt 04-005 (Division 1) **Change/Rev.:** 0
Temp Alt 04-006 (Division 2) **Change/Rev.:** 0

System Designator(s)/Description: P41, Standby Service Water System

Description of Proposed Change

Manual operation of SSW cooling tower fans will assist in maintaining SSW basin temperature within administrative limits during cold weather. During Shutdown Cooling operations while in Modes 4 and 5, manual operation of the SSW system and fans will control the heat removal capability of the UHS. This will be accomplished by placing a jumper around the contacts for a SSW system initiation based upon manual starting of the RHR pump (A or B) that is in service for shutdown cooling. The effect will be that the SSW system will still manually and automatically initiate as designed, except for the response to a RHR pump manual start. Any automatic initiation signal required in modes 4 and 5 will cause SSW to operate as designed. There is no automatic initiation signal that will only start the RHR pumps. Installation of the temporary alteration will require manual alignment of SSW valves and manual start of the SSW pump prior to manually starting the RHR pump. This is in accordance with the current procedures and places no additional burden on operations (Ref. SOI 04-1-01-E12-1, Rev. 122 and SOI 04-1-01-E12-2, Rev 101). This temporary alteration does not impact the ability to control SSW or RHR from the remote shutdown panels.

Temporary Alteration 04-005 and 04-006 discusses temperature limits for starting and stopping SSW fans. The fans can be cycled to maintain the SSW pump discharge temperature between 50°F and 75°F. The temporary alteration is only applicable in modes 4 and 5.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

- ☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.
- ☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2______. (Insert item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

☐	SCREENING	Sections I, II, III, and IV required
☐	50.59 EVALUATION EXEMPTION	Sections I, II, III, IV, and V required
☒	50.59 EVALUATION (#: SE-2004-002-R0	Sections I, II, III, IV, and VI required

Preparer: [signature] / Scott Cameron / EOI / DE – Mech / 2-27-04
Name (print) / Signature / Company / Department / Date

Reviewer: Robert W Fuller / Robert Fuller EOI / DE – Mech / 2-27-04
Name (print) / Signature / Company / Department / Date

OSRC M. A Krupa 2-27-04 OSRC MEETING #007-2004
Chairman's Signature / Date
[Required only for Programmatic Exclusion Screenings (see Section 5.9) and 50.59 Evaluations.]

List of Assisting/Contributing Personnel:

Name:	**Scope of Assistance:**
Jeff Brown, Robert Burrell, Andrew Fox, Amanda Gallagher, Gerald Lantz, Tom Matson, Dennis Chipley, and Ron Roma.	Preparation of 50.59, Additional review and comments on 50.59. Preparation of Temp Alts

II. SCREENING

A. Licensing Basis Document Review

1. **Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents?**

Operating License	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
Operating License	☐	☒	
TS	☐	☒	
NRC Orders	☐	☒	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

LBDs controlled under 50.59	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
FSAR	☒	☐	Various sections addressed in Section II.B. No LDC will be issued since the temp alt will only be in place during the outage.
TS Bases	☒	☐	TS Bases addressed in Section II.B. No LDC will be issued since the temp alt will only be in place during the outage.
Technical Requirements Manual	☒	☐	TRM addressed in Section II.B. No LDC will be issued since the temp alt will only be in place during the outage.
Core Operating Limits Report	☐	☒	
NRC Safety Evaluation Reports[1]	☐	☒	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

LBDs controlled under other regulations	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
Quality Assurance Program Manual[2]	☐	☒	
Emergency Plan[2]	☐	☒	
Fire Protection Program[3] (Includes the Fire Hazards Analysis)	☐	☒	
Offsite Dose Calculation Manual[3]	☐	☒	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. **Does the proposed activity involve a test or experiment not described in the FSAR?** ☐ Yes ☒ No
   If "yes," perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI.

3. **Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation?** ☐ Yes ☐ No ☒ N/A
   (Check "N/A" if dry fuel storage is not applicable to the facility.)
   If "yes," perform a 72.48 Review in accordance with NMM Procedure LI-112.
   (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

[1] If "YES," see Section 5.1.4. No LBD change is required.
[2] If "YES," notify the responsible department and ensure a 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES," evaluate the change in accordance with the requirements of the facility's Operating License Condition.

B. Basis

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License/Technical Specifications and/or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.)

The Temp Alts allow for bypassing the applicable contacts in the SSW initiation logic to prevent an SSW automatic initiation due to a manual RHR pump (A or B) start for Shutdown Cooling mode of RHR. All other initiation signals are unaffected and will start the respective SSW subsystems. For shutdown cooling, fuel pool cooling assist, and any other manual RHR system start it will be necessary to manually align and start SSW prior to starting the RHR pump. This is required by current procedures (Ref. SOI 04-1-01-E12-1, Rev. 122 and SOI 04-1-01-E12-2, Rev 101) and can be performed in the control room. Therefore, the temporary alteration will not result in any additional operator burdens. This method will allow manual control of the applicable cooling tower fans in order to aid in maintaining SSW temperature within administrative limits. The temp alts do not bypass or disable any other automatic initiation signal. There is no automatic initiation signal that will only start the RHR pump. Any automatic initiation signal will also start other ESF equipment and result in the initiation of SSW as designed. SSW response to LOCA, LOP, EDG, or the other component start signals will remain as designed. The Temp Alts do not change the ability of the SSW system to perform its design safety functions.

Temporary Alteration 04-005 and 04-006 discusses temperature limits for starting and stopping SSW fans. The fans can be cycled to maintain the SSW pump discharge temperature between 50°F and 75°F. The temporary alteration is only applicable in modes 4 and 5.

<u>OPERATING LICENSE</u>: The operating license was reviewed and does not contain specific limits or requirements of the SSW fans or RHR Shutdown Cooling. Therefore, no change is required to the Operating License.

<u>TECHNICAL SPECIFICATIONS</u>: The TS were reviewed for applicability. The Temp Alts are only allowed for use with Shutdown Cooling in Modes 4 and 5. Tech Spec 3.7.1 addresses the SSW and UHS operability during Modes 1, 2, and 3. This is not applicable since the temporary alterations are only applicable during modes 4 and 5. No other TS LCOs other than TRM TR3.7.1 (addressed separately below) pertain to the SSW system and fans being manually controlled for the specific purpose of maintaining SSW basin temperature within administrative limits during shutdown cooling. Therefore the TS are not affected by this change.

<u>NRC ORDERS</u>: The GGNS NRC Orders required licensees to establish interim compensatory measures to delineate licensee responsibility in response to the high level threat environment. Licensing was contacted because of the change to the Ultimate Heat Sink. The proposed changes to the SSW system and its control during Modes 4 and 5 do not pertain to station security. Therefore the NRC Orders are not affected by this change.

FSAR: The SSW System is discussed in several areas of the FSAR.

Section 9.2.1 discusses the SSW system in detail. Since the proposed Temp Alts will not disable or otherwise prevent an SSW automatic initiation from a LOCA, LOP, EDG start, or other component start, it still fulfills the design safety function as described in section 9.2.1. Manual control of the SSW system and fans for shutdown cooling is in accordance with the intent of section 9.2.1.1.2 which states that the SSW system "provides cooling to plant components, as required, during normal, reactor shutdown, and reactor isolation modes." The use of a temporary alteration to allow manual control of SSW system and fans during shutdown cooling will aid in keeping SSW temperature above administrative limits. The UHS temperature can still be controlled by the fans in the cooling tower; however, fan operation will only be manual during Shutdown Cooling or any other non-automatic function. Any SSW initiation signal except for RHR pump start will result in the system and fans running as designed. The worst case design basis accident for the UHS (LOCA/LOP with single failure) in Section 9.2 assumes that the fans will run for Post-LOCA heat removal. This assumption remains valid since the LOCA signals from high drywell pressure or low reactor water level will still result in an automatic SSW system initiation. The Temp Alts also do not change the requirement that the SSW system respond automatically to a LOCA with no operator action (Section 9.2.1.5). With the Temp Alts installed, the UHS still functions as described in section 9.2.5 because it will provide the designed cooling for LOCA conditions, and will provide cooling as required for normal shutdown operations.

Section 7.3.1.1.7 describes the SSW system instrumentation and logic. With the Temp Alts installed, the system will initiate as described with the exception of response to an RHR pump being started without another automatic SSW initiation signal present. The Temp Alts require Operations to manually align and start SSW to the applicable RHR pump to provide cooling to the pump seal cooler and pump room cooler prior to manually aligning RHR for Shutdown Cooling. Per current procedures (Ref. SOI 04-1-01-E12-2, Rev 101), SSW is manually aligned to RHR (valves aligned and SSW pump started) prior the RHR pumps being started. Additionally, the requirement that the SSW system fans and other essential components are prevented from being shutdown with an automatic initiation signal present is still satisfied except for manual start the RHR pump. Therefore in a LOCA or LOP SSW will function as designed.

Section 7.3.2.7.1 describes conformance with 10 CFR 50 General Design Criteria. GDC 20 requires that "protection systems shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that the specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety." This temp alt does not (1) impede the automatic operation of systems to protect fuel limits nor does it (2) impede the ability of the plant to sense accident conditions and respond to them.

Response to Criterion 20, Protection System Functions, states that SSW operates to serve other ESF systems and is initiated when they start to perform their ESF functions. This will remain true for all components performing automatic ESF functions except for the manual start of the RHR pumps (A or B), which are affected by the temp alt. For the pump used for Shutdown Cooling, SSW will be manually aligned before the pump is started to ensure the pump is protected from overheating as required by current procedures.

TS BASES: According to the TS Bases for the SSW system (p. B3.7-3) and TRM 3.7.1, an SSW subsystem is considered operable when the associated pump is operable and the associated piping, valves, instrumentation, and controls required to perform the safety related functions (including LOCA/LOP) are operable. Operability of the UHS requires four fans (two per cell) to be operable. Since the SSW system will automatically start on any required automatic signal besides manual start of the RHR pump, the system will be fully capable of performing the safety related functions for LOCA, LOP, and component starts other than RHR pump A or B. Additionally, the system will be manually placed in service with the fans manually controlled to maintain SSW temperature when supplying the RHR pump for shutdown cooling, and are therefore performing the design function of protecting the RHR pump, although not automatically.

TRM: Section 3.7.1 discusses SSW operability in modes 4 and 5. Conditions A and B address inoperability of one or more SSW fans. These requirements are not affected by Temporary Alterations 04-005 and 04-006. Although the fans will not automatically start on a manual RHR pump (A or B) start they will automatically start on all other initiation signals. Since the SSW basin temperatures will be maintained within administrative limits, thus within design limits, the UHS will be fully capable of removing all residual heat from the reactor during manual control of the fans, and in the event of any subsequent automatic initiation. SSW fans can be started and stopped from the control room, if required to maintain basin temperature within the acceptable band, thus ensuring that the associated RHR pump is adequately cooled.

COLR: The Core Operating Limits Report does not contain discussion of methods of controlling SSW temperature during Shutdown Cooling. Therefore there is no impact to the COLR.

NRC SERs: The NRC SER describes the NRC's response to the design of the plant as described in the FSAR. The intent of the SSW design as described in the FSAR is not affected by these temporary alterations as described above. Therefore, there is no impact to the SERs..

QAPM/EP/FPP/ODCM: These documents do not contain information relevant to control of the SSW system. Therefore they are unaffected by this change.

The proposed changes do not represent a test or experiment not previously described in the FSAR because proper cooling will be provided to the components served by SSW as required. Automatic response to LOCA/LOP is maintained, and the system will be capable of performing its design safety function. Additionally, the proposed change does not pertain to spent fuel storage.

C. References

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. **NOTE: Ensure that electronic and manual searches are performed using controlled copies of documents. If you have any questions, contact your site Licensing department.**

LBDs/Documents reviewed via keyword search:	Keywords:
FSAR, TS, TS Bases, Operating License, TRM	Shutdown cooling, fans, SSW

LBDs/Documents reviewed manually:

NRC Orders, NRC SERs, QAPM, ODCM, COLR, EP, FPP

D. Is the validity of this Review dependent on any other change? (See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines)

☐ Yes
☒ No

If "Yes," list the required changes.

N/A

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	☒	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	☒	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	☒	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	☒	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	☒	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	☒	Discharge any chemicals new or different from that previously discharged?
7.	☐	☒	Change the design or operation of the intake or discharge structures?
8.	☐	☒	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	☒	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	☒	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	☒	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	☒	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☐	☒	Involve the installation or modification of a stationary or mobile tank?
14.	☐	☒	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	☒	Involve burial or placement of any solid wastes in the site area that may affect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

IV. SECURITY PLAN SCREENING

If any of the following questions is answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	☒	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	☒	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	☒	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	☒	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	☒	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiber optics)?
6.	☐	☒	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	☒	Modify or otherwise affect (block, move, or alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	☒	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	☒	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	☒	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes".

B. Is the Security Plan actually impacted by the proposed activity? ☐ Yes ☐ No

C. Is a change to the Security Plan required? ☐ Yes Change # (optional) ☐ No

Name of Security Plan reviewer (print) / Signature / Date

V. 50.59 EVALUATION EXEMPTION

Enter this section only if a "yes" box was checked in Section II.A, above.

A. **Check the applicable boxes below. If any of the boxes are checked, a 50.59 Evaluation is not required. If none of the boxes are checked, perform a 50.59 Evaluation in accordance with Section V. Provide supporting documentation or references as appropriate.**

☐ The proposed activity meets all of the following criteria regarding design function per Section 5.6.1.1:

The proposed activity does not adversely affect the design function of an SSC as described in the FSAR; AND

The proposed activity does not adversely affect a method of performing or controlling a design function of an SSC as described in the FSAR; AND

The proposed activity does not adversely affect a method of evaluation that demonstrates intended functions of an SSC described in the FSAR will be accomplished.

☐ An approved, valid 50.59 Review(s) covering associated aspects of the proposed change already exists per Section 5.6.1.2. Reference 50.59 Evaluation # _____ (if applicable) or attach documentation. Verify the previous 50.59 Review remains valid.

☐ The NRC has approved the proposed activity or portions thereof per Section 5.6.1.3. Reference: _____

☐ The proposed activity is controlled by another regulation per Section 5.6.1.4.

B. **Basis**

Provide a clear, concise basis for determining the proposed activity may be exempted such that a third-party reviewer can reach the same conclusions. See Section 5.6.6 of the EOI 10CFR50.59 Review Program Guidelines for guidance.

N/A

VI. 50.59 EVALUATION

A. Executive Summary (Serves as input to NRC summary report. Limit to one page or less. Send an electronic copy to the site licensing department after OSRC approval, if available.)

Brief description of change, test, or experiment:

Temp Alt 04-005 and 04-006 will allow manual control of SSW cooling tower fans during Shutdown Cooling for plant Modes 4 and 5. The manual control of the SSW system and fans will control the heat removal capability when the fans are secured during cold weather which will aid in maintaining the basin water temperature within administrative limits. The SSW fans will be cycled to maintain SSW temperature between 50°F and 75°F. SSW temperatures are indicated/recorded on several instruments in the control room and logged hourly. SSW fans can be controlled in the control room. This temp alt adds minimal operator responsibility.

The Temp Alt will bypass the RHR pump (A or B) signal to the associated SSW initiation logic. Therefore, the SSW division for the affected RHR pump will not start automatically in response to the manual start of the RHR pump. The affected division of SSW will respond as designed to all other automatic initiation signals. There are no automatic initiation signals that only initiate the RHR pumps. The Temp Alt will include instructions that require manual alignment of SSW to the RHR pump prior to starting the pump with the Temp Alt installed. Operation procedures currently require manually starting and aligning SSW prior to starting a RHR pump for shutdown cooling. All operation of the RHR pumps and SSW system can be performed from the control room. Therefore there is no additional burden placed on operations by implementing the temporary alterations. There is no impact on the operation of the remote shutdown panel (except that a RHR pump start will not auto start SSW).

Reason for proposed Change:

During shutdown cooling operations and cold weather, SSW temperature may approach the administrative temperature limits for water in the SSW basin. By providing a method to run shutdown cooling with the ability to start and stop fans manually, SSW basin temperature can more easily be controlled within administrative limits.

50.59 Evaluation summary and conclusions

Bypassing the RHR pump (A or B) signal to the SSW initiation logic is an acceptable method of allowing manual control of SSW system and fans during shutdown cooling in mode 4 or 5 during cold weather operation. The SSW system will still perform required safety related functions with the Temp Alts installed, with the exception of automatic initiation in response to RHR A or B pump start. Manual operation of SSW will be required to ensure SSW is properly lined up to the RHR pump for shutdown cooling, as is currently done per the procedure for shutdown cooling, 04-1-01-E12-2. All other automatic initiation signals will cause SSW to automatically initiate as designed. The ability of the RHR system to automatically respond to ECCS signals is also not affected. Relying on manual operation to cool the RHR pumps for shutdown cooling does not pose more than a minimal increase in risk to the ability of the RHR system to perform its design safety function.

B. License Amendment Determination

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below. ☐ Yes ☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

As mitigating systems, the SSW system and UHS are not the direct subject of any UFSAR Chapter 15 accident analyses. The Temp Alt bypasses the RHR pump (A or B) manual start signal in the associated SSW division initiation logic. All other initiation signals remain in place and will function as designed. The temp alts do not bypass or disable any other initiation signals from causing an automatic initiation to occur as designed. There is no automatic initiation signal that will only start the RHR pump. Any automatic initiation signal will also start other ESF equipment and result in the initiation of SSW as designed. These temp-alts are only applicable for the manual alignment of the Shutdown Cooling Mode of RHR during plant modes 4 and 5.

Although the RHR pump manual start initiation signal is bypassed, these temp alts will not adversely affect the function or operation of the SSW system or the interfacing systems and will not compromise any safety related system, structure or component. The proposed change does not change the applicable SSW system or UHS design (other than the RHR A/B pump automatic SSW initiation), material, or construction standards, does not degrade overall SSW system or UHS reliability, and does not cause the SSW system or UHS to operate outside design limits. Thus, the temporary alterations do not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

The shutdown cooling mode of RHR is a manually aligned safety related function that requires proper functioning of both the RHR system and SSW system. The temp-alts provide means to manually control the SSW fans during Shutdown Cooling Mode of RHR in plant modes 4 and 5 in order to maintain SSW basin temperature within administrative limits. This would be achieved by bypassing the RHR pump start initiation signal in the SSW logic. This will provide the ability to manually start and stop the SSW fans from the control room, provided there is no other SSW initiation signal present. Operator action to manually align SSW to the RHR system prior to starting the RHR pumps is being relied upon to provide cooling to the RHR pumps (pump protection). If this action was not performed, a potential exist to damage the RHR pumps and lose associated decay heat removal capability. The likelihood of this occurrence is no greater than the likelihood of occurrence of failure to perform similar operator actions to manually align SSW for shutdown cooling since the current procedures for shutdown cooling requires that SSW be in service prior to starting the RHR pumps. Thus, installation of the temp alts would not result in more than a minimal increase in the likelihood of a malfunction of the RHR pumps or system.

The possibility for a human error allowing SSW basin temperature to increase above the design limit of 90°F also requires evaluation. Allowing basin temperature to exceed 90°F is unlikely. The SSW SOI 04-1-01-P41-1 requires SSW temperature to be maintained at or below 85°F during normal operation. RHR SOI 04-1-01-E12-2 requires that SSW temperature be logged hourly during shutdown cooling operation. SSW temperature exceeding 90°F is not considered credible since the Temp Alts are intended only to be used in cold weather and with minimal heat load. Temperature limits have been specified for cycling the SSW fans to ensure that the administrative and design limits are maintained.

If an operator were to fail to shutdown the SSW fans prior to the temperature dropping below 50°F then the SSW system would continue to function as currently designed. The upper limit (75°F) is sufficiently low that if operators failed to manually start the fans prior to reaching 75°F there would be multiple opportunities to realize this mistake prior to reaching the current administrative limit of 85°F. Although, given the low heat load in modes 4 and 5, and restriction on use in cold weather, it is unlikely that the administrative limit of 85°F could be reached at all.

SSW temperature indication and fan controls are located in the control room, therefore, this temporary alteration does not introduce additional operator burden.

The fans will still start upon an automatic initiation signal with the exception of a RHR pump start, thus maintaining basin temperature below the design limit of 90 degrees. Therefore, the operability of SSW/UHS is not challenged by operating the fans manually in Mode 4 and 5 since there is adequate assurance that temperature will be maintained within limits.

Therefore, installation of the temp alts do not result more than a minimal increase in the likelihood of a malfunction of the SSW or RHR system (components, etc) since it will have no adverse affect on any equipment which is important to safety and does not cause the RHR and SSW system (including components) to operate outside design limits.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR? ☐ Yes ☒ No

BASIS:

There is no increase in the consequences of an accident previously evaluated in the FSAR. The temp alts aids operation in maintaining the SSW basin temperature within administrative limits by manually controlling the SSW fans during shutdown cooling mode of RHR in cold weather operation and in mode 4 and 5. The SSW fans would operate as designed with all automatic initiation signals with the exception of a manual RHR pump start. The SSW system would still be able to perform its design safety function since the system will still be operating within it design limits. Any signal generated by an accident (e.g. drywell pressure, reactor water level) would result in SSW initiating as designed. Therefore, any accident would have consequences bounded by those currently analyzed within the FSAR.

With the temp alts installed, the automatic pump protection feature for the RHR system is bypassed for RHR pumps A and B. Operator action to manually align SSW to the RHR system prior to starting the RHR pumps is being relied upon to provide cooling the RHR pumps (pump protection). However, the likelihood of an operator failing to perform this action is no greater than the likelihood of an operator failing to perform this action under current operating conditions. If this action was not performed, a potential exist to damage the RHR pumps and lose decay heat removal capability. The likelihood of this occurrence is very low since the current procedures for shutdown cooling requires that SSW be in service prior to starting the RHR pumps. In the unlikely situation that this occurred, it would not result in a different consequence than those already evaluated in the UFSAR. The Technical Specifications provide requirements for adequate decay heat removal capabilities during modes 4 and 5 to ensure that component failures do not result in a total loss of decay heat removal capability.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

With the Temp Alt installed, the SSW system will respond as designed to all automatic initiation signals with the exception of the RHR pump start. Operator action will be relied upon to manually align SSW to RHR prior to starting the RHR pumps. If this action was not performed, a potential exists to damage the affected RHR pump and lose the associated decay heat removal capability. However, the likelihood of an operator failing to perform this action is no greater than the likelihood of an operator failing to perform this action under current operating conditions. The current procedures for shutdown cooling require that SSW be in service prior to starting the RHR pumps. A pump failure is an active failure that is already considered in station design. No new initiating conditions or design operating parameters are allowed by these Temp Alts.

The possibility for a human error allowing SSW basin temperature to increase above the design limit of 90°F also requires evaluation and is discussed in more detail in question 2 above.

Thus, the installation of the temp alts do not result in more than an increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The temp alts aids operation in maintaining the SSW basin temperature within administrative limits by manually controlling the SSW fans during shutdown cooling mode of RHR in cold weather operation with the plant in mode 4 or 5.
The temporary alterations do not disable the essential automatic SSW responses to ECCS and EDG equipment. The SSW system will operate within current design basis requirements and these temp alts do not create the possibility for a different type of accident than previously evaluated in the FSAR.

With the temp alts installed, the automatic pump protection feature for the RHR system is bypassed. Operator action to manually align SSW to the RHR system prior to starting the RHR pumps is being relied upon to provide cooling the RHR pumps (pump protection). A potential exist to damage the RHR pump if this action is not performed, this could lead to a loss the associated decay heat removal capability. It is not likely to occur since current procedures for shutdown cooling requires SSW to be in service prior to starting the RHR pumps.

Also, loss of an RHR pump (loss of decay heat removal capability) is already analyzed. Thus, installation of the temp alts does not create the possibility of an accident of a different type than any previously evaluated in the FSAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the FSAR?

☐ Yes
☒ No

BASIS:

The temp alt does not create a new failure mode. By removing the automatic SSW start on a manual RHR pump start the potential exists to operate an RHR pump without adequate cooling. However, current procedural guidance reduces the likelihood of this occurrence. Current operating procedures require manually starting and aligning SSW prior to manually starting the RHR pump. This temp alt removes the built in operational backup should procedural guidance not be followed. The active failure of an RHR pump is already evaluated and is part of the reason for backup decay heat removal capability. Therefore, the possibility of a malfunction of a different result than previously evaluated does not exist.

7. Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered?

☐ Yes
☒ No

BASIS:

The Temp Alts do not adversely affect any fission product barrier. The Temp Alts will allow manual operation of the SSW system to maintain SSW water temperature within administrative limits during shutdown cooling mode of RHR while in plant mode 4 and 5. The SSW basin temperature will be maintained within design limits as specified by Tech Specs and TRM requirements. Thus since design limits will be maintained, SSW will perform as designed to protect the fission product barrier.

8. Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses?

☐ Yes
☒ No

BASIS:

The Temp Alts will not allow SSW to be operated outside of administrative limits, which are within existing design limits. No design bases are being changed as a result of the temp alt. The manual control of the SSW fans during shutdown cooling will help maintain the SSW water temperature within administrative limits. The Temp Alts do not make any changes to the methodology for establishing any design bases, nor do they make a change in establishing a safety limit.

GGNS Commitment Change Evaluation Number

CCE 2002-007

COMMITMENT CHANGE EVALUATION FORM

Commitment Number:	A-29293	Plant Licensing Tracking Number:	CCE 2002-0007
Source Document:	10CFR,Part 55 45		

Commitment: Deletion? ☒ Revision? ☐

Has the original commitment been implemented? ☐ YES ☐ NO, Notify Plant Licensing

Original Commitment Description:

Perform transient tests and steady state tests on the simulator annually

Revised Commitment Description:

Summary of Justification for Change or Deletion:

10CFR55 45 was revised earlier this year and deleted a requirement to report on simulator test failures and their resolution and also deleted a requirement to submit the test plan for the upcoming four years
Entergy has implemented the 1998 version of ANSI 3 5 and issued TQ-202,Simulator Configuration Control, Rev 1
This procedure implements the testing requirements of ANSI 3 5 as it applies to GGNS,ANO,RBS, and WF3

(Attach additional sheets if necessary)
Refer to Attachment 9.4 for a flow diagram that outlines the commitment change evaluation process.

	Print Name/Signature	Date
Prepared By:	S R Kaskie *S R Kaskie*	8-21-02
Management Approval:	Robert Goldman *[signature]*	8/21/02
Plant Licensing Management Concurrence:	C A Bottemiller *[signature]*	8-22-02

10 CFR 55.45 was significantly eliminated.

QA RECORD
RT
NON-QA RECORD
INITIALS
NUMBER OF PAGES 1
DATE 8-29-02
RELATED DOCUMENT NUMBER

REF LI-110

PART I

1.1	Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Program, Fire Protection Program, or Security Plan?

☐ YES STOP. Do not proceed with this evaluation. Instead use appropriate codified process (e.g., 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment

☒ NO Go to Part II.

PART II

2.1 Could the change negatively impact the ability of a System, Structure, or Component (SSC) to perform its safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its intended safety function?

☐ YES Go to Question 2.2.

☒ NO Continue with Part III. Briefly describe rationale:

2 2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ YES ☐ NO

Basis:

Does the revised commitment create the possibility of a new or different kind of accident from any previously evaluated?

☐ YES ☐ NO

Basis:

Does the revised commitment involve a significant reduction in a margin of safety?

☐ YES ☐ NO

Basis:

If any of the above questions are answered Yes, STOP. Do not proceed with the revision , OR discuss change with NRC and obtain necessary approvals prior to implementation of the proposed change. If all three questions are answered NO, go to Part III

(Attach additional sheets as necessary.)

PART III

3.1 Was the original commitment (e.g., response to NOV, etc.) to restore an Obligation (i.e., rule, regulation, order or license condition)?

☐ YES Go to question 3.2.

☒ NO Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ YES Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

Rationale:

☐ NO STOP. Do not proceed with the revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ YES Go to Question 4.2.

☒ NO Go to Part V

4.2 Has the original commitment been implemented?

☐ YES STOP, You have completed this evaluation. Revise the commitment and notify NRC of revised commitment in summary report

☐ NO Go to Question 5.1.

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ YES Go to Question 5.2.

☒ NO STOP. You have completed this evaluation. Revise the commitment. No NRC notification required

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ YES Revise the commitment and notify NRC of revised commitment in next annual/RFO interval summary report.

☐ NO Revise commitment: no NRC notification is required:

REFERENCES

List documents (e.g., procedures, NRC submittals, etc.) affected by this change.

Doc. Number	Description

GGNS Commitment Change Evaluation Number

CCE 2002-008

COMMITMENT CHANGE EVALUATION FORM

Commitment Number.	P-32395	Plant Licensing Tracking Number:	CCE-2002-0008
Source Document:	AECM-90/0004		
Commitment:	Deletion? ➔	Revision? ☐	
Has the original commitment been implemented?	➔YES	☐ NO, Notify Plant Licensing	

Original Commitment Description:

PROCEDURAL WEAKNESSES IDENTIFIED BY RESPONSE TO SALP REPORT FOR INCLUSION OF APPROPRIATE IMPROVEMENTS INTO THE 10CFR50 59 PROCESS

Revised Commitment Description:

Completely remove the commitment

Summary of Justification for Change or Deletion:

There is no regulatory requirement or basis for this commitment

(Attach additional sheets if necessary)
Refer to Attachment 9.4 for a flow diagram that outlines the commitment change evaluation process.

Prepared By:	Guy H Davant *Guy H Davant*	9/27/02
	Print Name/Signature	Date
Management Approval:	F.G. BURFORD [signature]	9.27.02
	Print Name/Signature	Date
Plant Licensing Management Concurrence:	[signature] [illegible]	10-21-02
	Print Name/Signature	Date

QA RECORD
RT [illegible]
[illegible] QA RECORD
[illegible]
NUMBER of PAGES 4
DATE 10-23-02
RELATED DOCUMENT NUMBER

REF LI-110

PART I

1.1 **Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Program, Fire Protection Program, or Security Plan?**

☐ **YES** **STOP. Do not proceed with this evaluation. Instead use appropriate codified process (e.g., 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.**

➜ **NO** **Go to Part II.**

PART II

2.1 **Could the change negatively impact the ability of a System, Structure, or Component (SSC) to perform its safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its intended safety function?**

☐ **YES** **Go to Question 2.2**

➜ **NO** **Continue with Part III. Briefly describe rationale:**

Deleting this commitment has no impact on the safety function of any SSC There is no regulatory requirement or basis to establish a 50 59 Review Program This commitment was made in response to a SALP report that stated, " 10CFR50 59 evaluations and events continue to be weak "

2.2 **Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:**

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ **YES** ☐ **NO**

Basis:

Does the revised commitment create the possibility of a new or different kind of accident from any previously evaluated?

☐ **YES** ☐ **NO**

Basis:

Does the revised commitment involve a significant reduction in a margin of safety?

☐ **YES** ☐ **NO**

Basis:

If any of the above questions are answered Yes, STOP. Do not proceed with the revision , OR discuss change with NRC and obtain necessary approvals prior to implementation of the proposed change. If all three questions are answered NO, go to Part III.
(Attach additional sheets as necessary.)

PART III

3.1 Was the original commitment (e.g., response to NOV, etc.) to restore an Obligation (i.e., rule, regulation, order or license condition)?

☐ YES Go to question 3.2.

➜ NO Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ YES Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

Rationale:

☐ NO STOP. Do not proceed with the revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ YES Go to Question 4.2.

➜ NO Go to Part V.

4.2 Has the original commitment been implemented?

☐ YES STOP, You have completed this evaluation. Revise the commitment and notify NRC of revised commitment in summary report.

☐ NO Go to Question 5.1.

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ YES Go to Question 5.2.

➜ NO STOP. You have completed this evaluation. Revise the commitment. No NRC notification required

5.2 **Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?**

☐ **YES** **Revise the commitment and notify NRC of revised commitment in next annual/RFO interval summary report.**

☐ **NO** **Revise commitment: no NRC notification is required:**

REFERENCES

List documents (e.g., procedures, NRC submittals, etc.) affected by this change.

Doc. Number	Description
AECM-90/0004	Response to SALP Report, dated November 22, 1989
10 CFR 50 59, *Changes, Tests, and Experiments*	Regulation
NEI 96-07, *Guidelines for 10 CFR 50 59 Implementation*	Industry guidance document endorsed by the NRC in Reg Guide 1 187

GGNS Commitment Change Evaluation Number

CCE 2002-009

COMMITMENT CHANGE EVALUATION FORM CCE

Commitment Number:	P-23936	**Plant Licensing Tracking Number:**	2002-0009
Source Document:	AECM-82/490		
Commitment:	**Deletion?** ☐	**Revision?** ☒	
Has the original commitment been implemented?	☒ **YES**	☐ **NO, Notify Plant Licensing**	

Original Commitment Description:

Valve P53F006 of the ADS SYS will undergo valve seat leakage test every 18 mos.

Revised Commitment Description:

Valve P53F006 of the ADS SYS will undergo valve seat leakage test at a frequency not to exceed once every 60 months.

Summary of Justification for Change or Deletion:

The local leak rate test program was changed to a performance based testing program to be consistent with NRC approved industry guidance. The change to the program was approved under SE 96-0099-R00.
A performance based testing program allows for a test frequency of 60 months, changed from 18/24 month frequency, for valves that pass two consecutive local leak rate test. P53F006 has passed two consecutive tests and is currently on a 60 month frequency.

(Attach additional sheets if necessary)
Refer to Attachment 9.4 for a flow diagram that outlines the commitment change evaluation process.

	Print Name/Signature	Date
Prepared By:	Ed Burton / Ed Burton	11/6/2002
Management Approval:	[illegible] for M. Renfroe	11/7/02
Plant Licensing Management Concurrence:	C A Bottemiller [illegible]	11-12-02

✓	QA RECORD
	RT- 514.39
	NON-QA RECORD
	INITIALS [illegible]
NUMBER of PAGES	4
DATE 11/12/02	
RELATED DOCUMENT NUMBER-	

REF: LI-110

PART I

1.1 **Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Program, Fire Protection Program, or Security Plan?**

☐ **YES** **STOP. Do not proceed with this evaluation. Instead use appropriate codified process (e.g., 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.**

☒ **NO** **Go to Part II.**

PART II

2.1 **Could the change negatively impact the ability of a System, Structure, or Component (SSC) to perform its safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its intended safety function?**

☐ **YES** **Go to Question 2.2.**

☒ **NO** **Continue with Part III. Briefly describe rationale:**

Changing the test frequency has no impact on the safety function of any SCC. There is no regulatory requirement to perform a seat leakage test every 18 months. SE 96-0099-R00 documented the approval of going to a performance based test program. The frequency for P53F006 was/is determined using the criteria set forth in a performance based test program.

2.2 **Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:**

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ **YES** ☒ **NO**

Basis:

Changing the frequency does not significantly increase in the probability or consequences of an accident previously evaluated.

Does the revised commitment create the possibility of a new or different kind of accident from any previously evaluated?

☐ **YES** ☒ **NO**

Basis:

A test frequency change to the original commitment does not create the possibility of a new or different kind of accident from any previously evaluated.

Does the revised commitment involve a significant reduction in a margin of safety?

☐ **YES** ☒ **NO**

Basis:

Revising the commitment does not involve a significant reduction in a margin of safety.

If any of the above questions are answered Yes, STOP. Do not proceed with the revision , OR discuss change with NRC and obtain necessary approvals prior to implementation of the proposed change. If all three questions are answered NO, go to Part III.
(Attach additional sheets as necessary.)

PART III

3.1 Was the original commitment (e.g., response to NOV, etc.) to restore an Obligation (i.e., rule, regulation, order or license condition)?

☐ YES Go to question 3.2.

☒ NO Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ YES Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

Rationale:

☐ NO STOP. Do not proceed with the revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ YES Go to Question 4.2.

☒ NO Go to Part V.

4.2 Has the original commitment been implemented?

☐ YES STOP, You have completed this evaluation. Revise the commitment and notify NRC of revised commitment in summary report.

☐ NO Go to Question 5.1.

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ YES Go to Question 5.2.

☒ NO STOP. You have completed this evaluation. Revise the commitment. No NRC notification required.

5.2 **Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?**

☐ **YES** **Revise the commitment and notify NRC of revised commitment in next annual/RFO interval summary report.**

☐ **NO** **Revise commitment: no NRC notification is required:**

REFERENCES

List documents (e.g., procedures, NRC submittals, etc.) affected by this change.

Doc. Number	Description
Commitment P-23936	P53F006 (AD System valve) will undergo valve seat leakage at a frequency not to exceed once every 60 months.

GGNS Commitment Change Evaluation Number

CCE 2003-001

COMMITMENT CHANGE EVALUATION FORM

Commitment Number:	24290	**Plant Licensing Tracking Number:**	CCE 2003-001
Source Document:	AECM-88/0024 Att. 1 Area 5.S3		
Commitment:	Deletion? ☒	Revision? ☐	
Has the original commitment been implemented?	☒ YES	☐ NO, Notify Plant Licensing	

Original Commitment Description:

Plant Staff revised its drawing control procedure to provide for ready acceptance of NPE reviewed, approved and issued drawings and the assignment of the drawing coordinator to coordinate design drawings and resolve drawings problems.

Revised Commitment Description:

None

Summary of Justification for Change or Deletion:

This process is no longer applicable, documents do not go to a pending file they are transmitted directly to records and within a 24 hour time frame records issues and populates EDMS or rejects and sends drawing/document back to NPE for corrections and NPE has 30 days to re-submitted corrected drawings/documents.

(Attach additional sheets if necessary)

Refer to Attachment 9.4 for a flow diagram that outlines the commitment change evaluation process.

Prepared By:	Cindy Edwards / Cindy Edwards Print Name/Signature	2/28/03 Date
Management Approval:	H.D. Bo[illegible] Print Name/Signature	2/28/03 Date
Plant Licensing Management Concurrence:	[signature] Print Name/Signature	2-28-03 Date

2/28/03

RELATED DOCUMENT

NUMBER OF PAGES 4

DATE 2-28-03

RELATED DOCUMENT NUMBER

REF: LI-110

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Program, Fire Protection Program, or Security Plan?

☐ YES STOP. Do not proceed with this evaluation. Instead use appropriate codified process (e.g., 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ NO Go to Part II.

PART II

2.1 Could the change negatively impact the ability of a System, Structure, or Component (SSC) to perform its safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its intended safety function?

☐ YES Go to Question 2.2.

☒ NO Continue with Part III. Briefly describe rationale:

This was an old process that is no longer applicable.

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ YES ☐ NO

Basis:

Does the revised commitment create the possibility of a new or different kind of accident from any previously evaluated?

☐ YES ☐ NO

Basis:

Does the revised commitment involve a significant reduction in a margin of safety?

☐ YES ☐ NO

Basis:

If any of the above questions are answered Yes, STOP. Do not proceed with the revision , OR discuss change with NRC and obtain necessary approvals prior to implementation of the proposed change. If all three questions are answered NO, go to Part III.
(Attach additional sheets as necessary.)

PART III

3.1 Was the original commitment (e.g., response to NOV, etc.) to restore an Obligation (i.e., rule, regulation, order or license condition)?

☐ YES Go to question 3.2.

☒ NO Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ YES Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

Rationale:

☐ NO STOP. Do not proceed with the revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ YES Go to Question 4.2.

☒ NO Go to Part V.

4.2 Has the original commitment been implemented?

☐ YES STOP, You have completed this evaluation. Revise the commitment and notify NRC of revised commitment in summary report.

☐ NO Go to Question 5.1.

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ YES Go to Question 5.2.

☒ NO STOP. You have completed this evaluation. Revise the commitment. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ YES Revise the commitment and notify NRC of revised commitment in next annual/RFO interval summary report.

☐ NO Revise commitment: no NRC notification is required:

REFERENCES

List documents (e.g., procedures, NRC submittals, etc.) affected by this change.

Doc. Number	Description

GGNS Commitment Change Evaluation Number

CCE 2003-002

COMMITMENT CHANGE EVALUATION FORM

Commitment Number:	P-24401	Plant Licensing Tracking Number:	CCE 2003-0002
Source Document:	AECM-89/0003.88-26-01.III 2.b		
Commitment:	Deletion? ☐	Revision? ☒	
Has the original commitment been implemented?	☒ YES	☐ NO, Notify Plant Licensing	

Original Commitment Description:

PAP 01-S-06-15 has been revised to include detailed instructions for completing DOE/NRC form 741. The personnel responsible for SNM receipt and completion of the form have been made aware of the requirement to fill out the form as required by NUREG/BR-0006.

Revised Commitment Description:

NMM NF-104 has been revised to include instructions for completing DOE/NRC form 741.. The personnel responsible for SNM receipt and completion of the form have been made aware of the requirement to fill out the form as required by NUREG/BR-0006.

Summary of Justification for Change or Deletion:

Under NMM procedure NF-104 rev. 2 (to be issued with an effective date of 7/1/2003), overall control of the Special Nuclear Materials Program is assigned to the Manager of the Nuclear Engineering Analysis department at Echelon. PAP 01-S-06-15 will be superceded by NF-104. The SNM Management and Control process will be standardized, with NEAD responsible for all EN-S sites. Thus, this commitment will be met by NF-104.

(Attach additional sheets if necessary)

Refer to Attachment 9.4 for a flow diagram that outlines the commitment change evaluation process.

	Print Name/Signature	Date
Prepared By:	Robert B Martin / Robert B Martin	6-16-03
Management Approval:	Ken Walker / Ken Walker	6-16-03
Plant Licensing Management Concurrence:	CABarr... / CABotter...	6-16-03

☑	QA RECORD
	RT = 1314.37
	NON-QA RECORD
	INITIALS
NUMBER of PAGES	4
DATE	6-26-03
RELATED DOCUMENT NUMBER =	

REF: LI-110

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Program, Fire Protection Program, or Security Plan?

☐ YES STOP. Do not proceed with this evaluation. Instead use appropriate codified process (e.g., 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ NO Go to Part II.

PART II

2.1 Could the change negatively impact the ability of a System, Structure, or Component (SSC) to perform its safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its intended safety function?

☐ YES Go to Question 2.2.

☒ NO Continue with Part III. Briefly describe rationale:

Administrative change with no impact on SSC's.

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ YES ☐ NO

Basis:

Does the revised commitment create the possibility of a new or different kind of accident from any previously evaluated?

☐ YES ☐ NO

Basis:

Does the revised commitment involve a significant reduction in a margin of safety?

☐ YES ☐ NO

Basis:

If any of the above questions are answered Yes, STOP. Do not proceed with the revision , OR discuss change with NRC and obtain necessary approvals prior to implementation of the proposed change. If all three questions are answered NO, go to Part III.
(Attach additional sheets as necessary.)

PART III

3.1 Was the original commitment (e.g., response to NOV, etc.) to restore an Obligation (i.e., rule, regulation, order or license condition)?

☐ YES Go to question 3.2.

☒ NO Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ YES Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

Rationale:

☐ NO STOP. Do not proceed with the revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ YES Go to Question 4.2.

☒ NO Go to Part V.

4.2 Has the original commitment been implemented?

☐ YES STOP, You have completed this evaluation. Revise the commitment and notify NRC of revised commitment in summary report.

☐ NO Go to Question 5.1.

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☒ YES Go to Question 5.2.

☐ NO STOP. You have completed this evaluation. Revise the commitment. No NRC notification required.

5.2 **Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?**

☐ **YES** **Revise the commitment and notify NRC of revised commitment in next annual/RFO interval summary report.**

☒ **NO** **Revise commitment: no NRC notification is required:**

REFERENCES

List documents (e.g., procedures, NRC submittals, etc.) affected by this change.

Doc. Number	Description
01-S-06-15	GGNS SNM procedure. Procedure will be superceded by NF-104, rev. 2

GGNS Commitment Change Evaluation Number

CCE 2003-003

COMMITMENT CHANGE EVALUATION FORM

Commitment Number:	P-24402,24403	**Plant Licensing Tracking Number:**	CCE 2003-0003
Source Document:	AECM-89/0003.88-26-01.III.3 .a&b		

Commitment: Deletion? ☒ Revision? ☐

Has the original commitment been implemented? ☒ YES ☐ NO, Notify Plant Licensing

Original Commitment Description:

SERI has determined that the SNM Management and Control process could be streamlined by assigning complete control of the SNM program to the GGNS General Manager. This reassignment of responsibilities has been implemented consolidating activities under one department and location. Future SNM reports will be sent out by the GGNS General Manager removing the NMM and the Nuclear Licensing Section from the SNM reporting process.

PAP 01-S-06-15 has also been revised to reflect transfer of responsibility of the SNM program to the GGNS General Manager.

Revised Commitment Description:

DELETE these commitments.

Summary of Justification for Change or Deletion:

Under NMM procedure NF-104 rev. 2 (to be issued with an effective date of 7/1/2003), overall control of the Special Nuclear Materials Program is assigned to the Manager of the Nuclear Engineering Analysis department at Echelon. PAP 01-S-06-15 will be superceded by NF-104. The SNM Management and Control process will be standardized, with NEAD responsible for all EN-S sites. In addition, per NEI 99-04 guidelines, these are not commitments.

(Attach additional sheets if necessary)
Refer to Attachment 9.4 for a flow diagram that outlines the commitment change evaluation process.

Prepared By:	Robert B Martin / Robert B Martin **Print Name/Signature**	6-13-03 **Date**
Management Approval:	Ken Walker / Ken Walker **Print Name/Signature**	6-16-03 **Date**
Plant Licensing Management Concurrence:	C A Bottemiller **Print Name/Signature**	6-16-03 **Date**

☑	QA RECORD
	RT = 13.4.3.7
	NON-QA RECORD
	INITIALS
NUMBER of PAGES	4
DATE	6-26-03
RELATED DOCUMENT NUMBER =	

REF: LI-110

PART I

1.1 **Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Program, Fire Protection Program, or Security Plan?**

☐ **YES** **STOP. Do not proceed with this evaluation. Instead use appropriate codified process (e.g., 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.**

☒ **NO** **Go to Part II.**

PART II

2.1 **Could the change negatively impact the ability of a System, Structure, or Component (SSC) to perform its safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its intended safety function?**

☐ **YES** **Go to Question 2.2.**

☒ **NO** **Continue with Part III. Briefly describe rationale:**

Administrative change with no impact on SSC's.

2.2 **Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:**

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ **YES** ☐ **NO**

Basis:

Does the revised commitment create the possibility of a new or different kind of accident from any previously evaluated?

☐ **YES** ☐ **NO**

Basis:

Does the revised commitment involve a significant reduction in a margin of safety?

☐ **YES** ☐ **NO**

Basis:

If any of the above questions are answered Yes, STOP. Do not proceed with the revision , OR discuss change with NRC and obtain necessary approvals prior to implementation of the proposed change. If all three questions are answered NO, go to Part III.
(Attach additional sheets as necessary.)

PART III

3.1 Was the original commitment (e.g., response to NOV, etc.) to restore an Obligation (i.e., rule, regulation, order or license condition)?

☐ YES Go to question 3.2.

☒ NO Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ YES Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

Rationale:

☐ NO STOP. Do not proceed with the revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ YES Go to Question 4.2.

☒ NO Go to Part V.

4.2 Has the original commitment been implemented?

☐ YES STOP, You have completed this evaluation. Revise the commitment and notify NRC of revised commitment in summary report.

☐ NO Go to Question 5.1.

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☒ YES Go to Question 5.2.

☐ NO STOP. You have completed this evaluation. Revise the commitment. No NRC notification required.

5.2 **Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?**

☐ **YES** **Revise the commitment and notify NRC of revised commitment in next annual/RFO interval summary report.**

☒ **NO** **Revise commitment: no NRC notification is required:**

REFERENCES

List documents (e.g., procedures, NRC submittals, etc.) affected by this change.

Doc. Number	Description
01-S-06-15	GGNS SNM procedure. Procedure will be superceded by NF-104, rev. 2

GGNS Commitment Change Evaluation Number

CCE 2003-004

COMMITMENT CHANGE EVALUATION FORM

Commitment Number:	25027	**Plant Licensing Tracking Number:**	CCE 2003-0004
Source Document:	GNRO-93/00029 Paragraph 3, Sentence 4		
Commitment:	**Deletion?** ☒	**Revision?** ☐	
Has the original commitment been implemented?	☒ YES	☐ NO, Notify Plant Licensing	

Original Commitment Description:

"Transmitters confirmed of having a loss of fill-oil failure will be dispositioned in accordance with our non-conformance process or the applicable technical specification."

Revised Commitment Description:

Delete commitment

Summary of Justification for Change or Deletion:

Instruction 17-S-06-3 provides the means to track and trend Rosemount transmitters that are susceptible to fill-oil loss. GNRO-93/00029, Paragraph 3 states that trending is required to detect this condition until replacement or the appropriate psi-month threshold recommended by Rosemount is achieved. Grand Gulf has either replaced or achieved the psi-month threshold for these transmitters (see attached). This commitment has been has been implemented and is now complete. *Current non-conformance programs will continue to identify component issues.*

(Attach additional sheets if necessary)

Refer to Attachment 9.4 for a flow diagram that outlines the commitment change evaluation process.

Prepared By:	Lee Eaton / Lee E. Eaton Print Name/Signature	7/30/03 Date
Management Approval:	ORt 8/5/03 A.D. Barfield [signature] Print Name/Signature	8/7/03 Date
Plant Licensing Management Concurrence:	CABottemiller [signature] Print Name/Signature	8-11-03 Date

QA RECORD
RT 614.39
NON-QA RECORD
INITIALS WMM
NUMBER of PAGES 17
DATE 8-11-03
RELATED DOCUMENT NUMBER

REF: LI-110

PART I

1.1 **Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Program, Fire Protection Program, or Security Plan?**

☐ **YES** **STOP. Do not proceed with this evaluation. Instead use appropriate codified process (e.g., 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.**

☒ **NO** **Go to Part II.**

PART II

2.1 **Could the change negatively impact the ability of a System, Structure, or Component (SSC) to perform its safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its intended safety function?**

☐ **YES** **Go to Question 2.2.**

☒ **NO** **Continue with Part III. Briefly describe rationale:**

Susceptible Rosemount transmitters in service less than the psi-month threshold are vulnerable to fill-oil loss failures. GGNS achieved the psi-month threshold for all susceptible transmitters. Therefore, removing the trending requirements will have no negative impact. (See Attached)

2.2 **Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:**

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ **YES** ☐ **NO**
Basis:

Does the revised commitment create the possibility of a new or different kind of accident from any previously evaluated?

☐ **YES** ☐ **NO**
Basis:

Does the revised commitment involve a significant reduction in a margin of safety?

☐ **YES** ☐ **NO**
Basis:

If any of the above questions are answered Yes, STOP. Do not proceed with the revision , OR discuss change with NRC and obtain necessary approvals prior to implementation of the proposed change. If all three questions are answered NO, go to Part III.
(Attach additional sheets as necessary.)

PART III

3.1 Was the original commitment (e.g., response to NOV, etc.) to restore an Obligation (i.e., rule, regulation, order or license condition)?

☐ YES Go to question 3.2.

☒ NO Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ YES Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

Rationale:

☐ NO STOP. Do not proceed with the revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☒ YES Go to Question 4.2.

☐ NO Go to Part V.

4.2 Has the original commitment been implemented?

☒ YES STOP, You have completed this evaluation. Revise the commitment and notify NRC of revised commitment in summary report.

☐ NO Go to Question 5.1.

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ YES Go to Question 5.2.

☐ NO STOP. You have completed this evaluation. Revise the commitment. No NRC notification required.

5.2 **Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?**

☐ **YES** **Revise the commitment and notify NRC of revised commitment in next annual/RFO interval summary report.**

☐ **NO** **Revise commitment: no NRC notification is required:**

REFERENCES

List documents (e.g., procedures, NRC submittals, etc.) affected by this change.

Doc. Number	Description
GNRO-93/00029	Response to NRC Bulletin 90-01, Supplement 1
17-S-06-3	Rosemount Enhanced Monitoring Program

Attachment to Commitment Change Evaluation
CCE 2003-0004

Background
Commitment #25027 is addressed in GNRO-93/00029, Paragraph 3, sentence 4. It is implemented within directive 17-S-06-3. The committed sentence states "Transmitters confirmed of having a loss of fill-oil failure will be dispositioned in accordance with our non-conformance process of the applicable technical specification." 17-S-06-3 is the means to trend these susceptible transmitters. GNRO-93/00029, Paragraph 3 also states: "This program will remain in effect, with the transmitters listed in Tables 2, 3, and 4 being monitored on an 18-month (cycle) basis until replacement or the appropriate psi-month threshold recommended by Rosemount is achieved. The transmitters listed in Table 1 will be replaced during the next refueling outage, currently scheduled for October 1993."

This attachment will evaluate if the remaining PSI-Months has been achieved, that GGNS is no longer susceptible to this type failure, and that the commitment is no longer applicable.

Transmitter Scope
NOTE: Reference GNRO-93/00029 for more table information.
Table 1 transmitters
Accumulated 53,450 PSI-Months as of May 1993
130,000 PSI-Month Threshold
Worst case 76,550 PSI-Months remaining as of May 1993
NOTE: All five transmitters were replaced. No further actions are required for these transmitters.

Transmitter	Replacement WO	Date
1E31N086C	98003	10/6/93
1E31N089A	98308 and	10/6/93
	141807	4/18/95
1E31N089B	98313	10/5/93
1E31N089C	98314	10/1/93
1E31N089D	98318	10/4/93

Table 2 transmitters
Accumulated 53,320 PSI-Months as of May 1993
60,000 PSI-Month Threshold
Worst case 6,680 PSI-Months remaining as of May 1993

Table 3 transmitters
Accumulated 53,320 PSI-Months as of May 1993
60,000 PSI-Month Threshold
Worst case 6,680 PSI-Months remaining as of May 1993

Table 4 transmitters
Lowest accumulated transmitter is 6,640 PSI-Months as of May 1993
60,000 PSI-Month Threshold
Worst case 53,360 PSI-Months remaining as of May 1993

On-Line Data
On-line data was referenced from the Unit's Generation Performance Indicators to acquire the number of GGNS on-line hours since May 1993. The affected transmitters are pressurized whenever the unit is on-line. Therefore, on-line hours will be used to determine when these transmitters were pressurized.

Year	**Hours On-Line**	**Comments**
1993 starting in June	3223	6846 hrs (1993 total) – 3623 hrs (hrs before June)
1994	8286	
1995	6832	
1996	7698	
1997	8760	
1998	7642	
1999	6946	
2000	8634	
2001	8041	
2002	8140	
2003 to July 17th	4566	
	78,768 total	

Assumptions
- All Rosemount transmitters requiring trending by GNRO-93/00029 are sensing either Main Steam Line (MSL) pressure or Reactor pressure. Reactor and MSL pressure is approximately 1030 psig or above when the unit is synchronized to the grid. To ensure the calculation is conservative 1000 psig will be used.
- Since particular off-line months were not determined the calculation will use 31 days per month. This will ensure a conservative calculation.

Calculation

This calculation will determine the cumulative PSI-Months since the GNRO-93/00029 was issued on May 1993:

Total on-line hours from June 1993 to July 17, 2003 = 78,768 hours (see summed hours above)
On-line months = 78,768 hrs / 24 hrs / 31 months = 105.9
PSI-Months = 1000 psig * 105.9 on-line months = 105,900

105,900 PSI-Months have accumulated since May 1993. This is well above the remaining 53,360 PSI-Months necessary to satisfy the trending requirements of GNRO-93/00029. Therefore, trending transmitters identified in GNRO-93/00029 is no longer required, and commitment #25027 is no longer applicable.

Performed by Lee E. Eaton 12/30/03

Reviewed by C Matson 12-30-03

GGNS Commitment Change Evaluation Number

CCE 2003-005

PART I

1.1 **Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Program, Fire Protection Program, or Security Plan?**

☐ **YES** **STOP. Do not proceed with this evaluation. Instead use appropriate codified process (e.g., 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.**

☒ **NO** **Go to Part II.**

PART II

2.1 **Could the change negatively impact the ability of a System, Structure, or Component (SSC) to perform its safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its intended safety function?**

☐ **YES** **Go to Question 2.2.**

☒ **NO** **Continue with Part III. Briefly describe rationale:**

The change allows whoever is responsible for what is stored of items in the pools to also be responsible for maintaining an inventory of what is stored in the pools.

2.2 **Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:**

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ **YES** ☐ **NO**
Basis:

Does the revised commitment create the possibility of a new or different kind of accident from any previously evaluated?

☐ **YES** ☐ **NO**
Basis:

Does the revised commitment involve a significant reduction in a margin of safety?

☐ **YES** ☐ **NO**
Basis:

If any of the above questions are answered Yes, STOP. Do not proceed with the revision , OR discuss change with NRC and obtain necessary approvals prior to implementation of the proposed change. If all three questions are answered NO, go to Part III.
(Attach additional sheets as necessary.)

COMMITMENT CHANGE EVALUATION FORM

Commitment Number:	P-29077	Plant Licensing Tracking Number:	CCE 2003-0005
Source Document:	GNRO 94/103.94-13-01.III.S2		
Commitment:	Deletion? ☐	Revision? ☒	
Has the original commitment been implemented?	☒ YES	☐ NO, Notify Plant Licensing	

Original Commitment Description:

The miscellaneous material storage procedure was revised to require copies of the inventory sheets and item location maps to be maintained by Health Physics personnel.

Revised Commitment Description:

The miscellaneous material storage procedure was revised to identify the personnel who are required to maintain copies of the inventory sheets and item location maps for items stored in the pools.

Summary of Justification for Change or Deletion:

Original violation was that written records of items stored in the pools were not available to inform workers of hazards present for items stored in pools. At the time Health Physics was assigned the responsibility of maintaining the inventory. Currently Reactor Engineering is responsible for all items in the refueling pools and has the responsibility to maintain an inventory. Following RF-13 the Outage Group will be responsible for storage of miscellaneous items in the pools. The issue was not properly maintaining an inventory available and not who was responsible for maintaining the inventory. The change meets the intent of the commitment by identifying in the procedure who is responsible for maintaining the list.

(Attach additional sheets if necessary)

Refer to Attachment 9.4 for a flow diagram that outlines the commitment change evaluation process.

Prepared By:	John D. Williams / John D. Williams	08-20-2003
	Print Name/Signature	Date
Management Approval:	N. A. Edney II / N. A. [signature] II for R. A. Wilson	8/21/03
	Print Name/Signature	Date
Plant Licensing Management Concurrence:	C A M. Acker [signature]	8-27-03
	Print Name/Signature	Date

REF: LI-110

PART III

3.1 Was the original commitment (e.g., response to NOV, etc.) to restore an Obligation (i.e., rule, regulation, order or license condition)?

☐ YES Go to question 3.2.

☒ NO Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ YES Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

Rationale:

☐ NO STOP. Do not proceed with the revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ YES Go to Question 4.2.

☒ NO Go to Part V.

4.2 Has the original commitment been implemented?

☐ YES STOP, You have completed this evaluation. Revise the commitment and notify NRC of revised commitment in summary report.

☐ NO Go to Question 5.1.

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☒ YES Go to Question 5.2.

☐ NO STOP. You have completed this evaluation. Revise the commitment. No NRC notification required.

5.2 **Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?**

☐ **YES** **Revise the commitment and notify NRC of revised commitment in next annual/RFO interval summary report.**

☒ **NO** **Revise commitment: no NRC notification is required:**

REFERENCES

List documents (e.g., procedures, NRC submittals, etc.) affected by this change.

Doc. Number	Description
01-S-08-6	GGNS Radioactive Material Control Procedure. Assigned Rx. Engineering Responsibility for inventory.
17-S-02-300	GGNS SNM Movement and Inventory Control Procedure. Has Attachment used for the inventory.
17-S-02-301	GGNS NNM Movement and Inventory Control Procedure. Has Attachment used for the inventory.

GGNS Commitment Change Evaluation Number

CCE 2003-007

COMMITMENT CHANGE EVALUATION FORM

Commitment Number:	P 34691	Plant Licensing Tracking Number:	CCE 2003-0007
Source Document:	AECM 85/0201 RESPONSE 2, PARAGRAPH 3		

Commitment: Deletion? ☑ Revision? ☐

Has the original commitment been implemented? ☑ YES ☐ NO, Notify Plant Licensing

Original Commitment Description:

THIS COMMITMENT SPECIFIES THAT GGNS ADMINISTRATIVE PROCEDURE "DETERMINATION OF SAFETY/QUALITY CLASSIFICATIONS" WILL BE USED FOR GUIDANCE UNTIL THE MASTER EQUIPMENT LIST DATABASE IS DEVELOPED.

Revised Commitment Description:

Summary of Justification for Change or Deletion:

THIS COMMITMENT IS NO LONGER VALID. THE MASTER EQUIPMENT LIST, AND IT'S SUCCESSORS (CDB, EDB) HAVE SATISFIED THE REQUIREMENT THAT PROMPTED THE ISSUANCE OF THIS COMMITMENT.

(Attach additional sheets if necessary)
Refer to Attachment 9.4 for a flow diagram that outlines the commitment change evaluation process.

	Print Name/Signature	Date
Prepared By:	DANIEL E. JOHNSON Daniel E. Johnson	NOVEMBER 12, 2003
Management Approval:	Charles H Quick CHQuick	11-12-2003
Plant Licensing Management Concurrence:	C A Bottemiller [signature]	11-18-03

☑	QA RECORD
	RT- 19.57
	NON-QA RECORD
	INITIALS NVR
NUMBER of PAGES	4
DATE	11-18-03
RELATED DOCUMENT NUMBER	

REF: LI-110

PART I		
1.1	Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Program, Fire Protection Program, or Security Plan?	

☐ YES STOP. Do not proceed with this evaluation. Instead use appropriate codified process (e.g., 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☑ NO Go to Part II.

PART II

2.1 Could the change negatively impact the ability of a System, Structure, or Component (SSC) to perform its safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its intended safety function?

☐ YES Go to Question 2.2.

☑ NO Continue with Part III. Briefly describe rationale:

THIS COMMITMENT CONTAINS NO REQUIREMENTS OR CONTROLS THAT AFFECT OR MENTION ANY GGNS SSC'S.

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ YES ☐ NO

Basis:

N/A

Does the revised commitment create the possibility of a new or different kind of accident from any previously evaluated?

☐ YES ☐ NO

Basis:

N/A

Does the revised commitment involve a significant reduction in a margin of safety?

☐ YES ☐ NO

Basis:

N/A

If any of the above questions are answered Yes, STOP. Do not proceed with the revision , OR discuss change with NRC and obtain necessary approvals prior to implementation of the proposed change. If all three questions are answered NO, go to Part III.
(Attach additional sheets as necessary.)

PART III

3.1 **Was the original commitment (e.g., response to NOV, etc.) to restore an Obligation (i.e., rule, regulation, order or license condition)?**

☐ YES **Go to question 3.2.**

☑ NO **Go to Part IV.**

3.2 **Is the proposed revised commitment date necessary and justified?**

☐ YES **Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.**

Rationale:

N/A

☐ NO **STOP. Do not proceed with the revision, OR apply for appropriate regulatory relief.**

PART IV

4.1 **Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?**

☑ YES **Go to Question 4.2.**

☐ NO **Go to Part V.**

4.2 **Has the original commitment been implemented?**

☑ YES **STOP, You have completed this evaluation. Revise the commitment and notify NRC of revised commitment in summary report.**

☐ NO **Go to Question 5.1.**

PART V

5.1 **Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?**

☐ YES **Go to Question 5.2.**

☐ NO **STOP. You have completed this evaluation. Revise the commitment. No NRC notification required.**

5.2 **Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?**

☐ **YES** **Revise the commitment and notify NRC of revised commitment in next annual/RFO interval summary report.**

☐ **NO** **Revise commitment: no NRC notification is required:**

REFERENCES

List documents (e.g., procedures, NRC submittals, etc.) affected by this change.

Doc. Number	Description

GGNS Commitment Change Evaluation Number

CCE 2003-008

COMMITMENT CHANGE EVALUATION FORM

Commitment Number:	P-34723	**Plant Licensing Tracking Number:**	CCE 2003-0008
Source Document:	GNRO-2000/00011 .99-1503. PARA 3 Item 3		
Commitment:	**Deletion?** ☐	**Revision?** ☒	
Has the original commitment been implemented?	☐ **YES**	☐ **NO, Notify Plant Licensing**	

Original Commitment Description:

Revise Procedure 01-S-06-1 'Protective Tagging System' to more clearly Describe how information tags are to be used.

Revised Commitment Description:

Revise Procedure 02-S-01-38, "Protective Tagging" to more clearly describe how Caution tags are to be used

Summary of Justification for Change or Deletion:

01-S-06-1 has been replaced by OP-102 and 02-S-01-38. Information tags are now Caution tags. this added info is included in site specific procedure 02-S-01-38

(Attach additional sheets if necessary)
Refer to Attachment 9.4 for a flow diagram that outlines the commitment change evaluation process.

Prepared By:	[signature]	11/20/03
	Print Name/Signature	**Date**
Management Approval:	Stephen Humphries / [signature]	11/20/03
	Print Name/Signature	**Date**
Plant Licensing Management Concurrence:	[signature] C A Bottemiller	11-20-03
	Print Name/Signature	**Date**

☒	QA RECORD
	RT- 514.57
	NON-QA RECORD
	INITIALS WPM
NUMBER of PAGES	1 of
DATE	11-21-03
RELATED DOCUMENT NUMBER	

REF: LI-110

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Program, Fire Protection Program, or Security Plan?

☐ YES — STOP. Do not proceed with this evaluation. Instead use appropriate codified process (e.g., 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ NO — Go to Part II.

PART II

2.1 Could the change negatively impact the ability of a System, Structure, or Component (SSC) to perform its safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its intended safety function?

☐ YES — Go to Question 2.2.

☒ NO — Continue with Part III. Briefly describe rationale:

only nomenclature change

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ YES ☒ NO

Basis:

Nomenclature change only

Does the revised commitment create the possibility of a new or different kind of accident from any previously evaluated?

☐ YES ☒ NO

Basis:

Nomenclature change only

Does the revised commitment involve a significant reduction in a margin of safety?

☐ YES ☒ NO

Basis:

Nomenclature change only

If any of the above questions are answered Yes, STOP. Do not proceed with the revision , OR discuss change with NRC and obtain necessary approvals prior to implementation of the proposed change. If all three questions are answered NO, go to Part III.
(Attach additional sheets as necessary.)

PART III

3.1 Was the original commitment (e.g., response to NOV, etc.) to restore an Obligation (i.e., rule, regulation, order or license condition)?

☐ YES Go to question 3.2.

☒ NO Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ ~~YES~~ Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

Rationale:

☐ NO STOP. Do not proceed with the revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ YES Go to Question 4.2.

☒ NO Go to Part V.

4.2 Has the original commitment been implemented?

☒ YES STOP, You have completed this evaluation. Revise the commitment and notify NRC of revised commitment in summary report.

☐ NO Go to Question 5.1.

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ YES Go to Question 5.2.

☐ NO STOP. You have completed this evaluation. Revise the commitment. No NRC notification required.

5.2 **Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?**

☐ **YES** **Revise the commitment and notify NRC of revised commitment in next annual/RFO interval summary report.**

10/21/87

☒ **NO** **Revise commitment: no NRC notification is required:**

REFERENCES

List documents (e.g., procedures, NRC submittals, etc.) affected by this change.

Doc. Number	Description

GGNS Commitment Change Evaluation Number

CCE 2003-009

COMMITMENT CHANGE EVALUATION FORM

Commitment Number:	34432	**Plant Licensing Tracking Number:**	CCE 2003-0009
Source Document:	GNRI-99/00047		

Commitment: **Deletion?** ☒ **Revision?** ☐

Has the original commitment been implemented? ☒ **YES** ☐ **NO, Notify Plant Licensing**

Original Commitment Description:

Check valves E38F002A/B and E38F003A/B, have historically been disassembled and inspected on a sample basis per 06-ME-1E38-R-0001 to ensure Code compliance (ref. Valve relief request VRR-E38-01). The Closed safety position was tested on a Cold Shutdown frequency and the Open safety position was tested on a Refuel frequency.

Revised Commitment Description:

Check valves E38F002A/B and E38F003A/B will now be Full Stroke Opened and Closed on a Cold Shutdown Frequency (including refueling outages) per Ma-1988, Part 10 to ensure obturator travel to the Open and Closed safety positions.

Summary of Justification for Change or Deletion:

Check valves E38F002A/B and E38F003A/B, have historically been disassembled and inspected on a sample basis per 06-ME-1E38-R-0001 to ensure Code compliance as documented in CEP-IST-1, 2 and 3 (ref. Valve Relief Request VRR-E38-01). This relief request will be deleted and the valves will now be Full Stroke Opened and Closed on a Cold Shutdown Frequency (including refueling outages) per Ma-1988, Part 10 to ensure obturator travel to the safety positions. This meets all Regulatory Requirements as defined in OMa-1988, Part 10 per Telecon from Gary Young on 12/11/03.

(Attach additional sheets if necessary)

Refer to Attachment 9.4 for a flow diagram that outlines the commitment change evaluation process.

Prepared By:	Gary D. Young/ Gary D. Young	12/11/03
	Print Name/Signature	Date
Management Approval:	[illegible] for Mike Renfroe	12/11/03
	Print Name/Signature	Date
Plant Licensing Management Concurrence:	C A Bottemiller [signature]	12-15-03
	Print Name/Signature	Date

[illegible] 12/11/03

✓	QA RECORD
	RT- R6 B14.39
	NON-QA RECORD
	INITIALS [illegible]
NUMBER of PAGES	4
DATE	12-15-03
RELATED DOCUMENT NUMBER-	

REF: LI-110

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Program, Fire Protection Program, or Security Plan?

☐ YES STOP. Do not proceed with this evaluation. Instead use appropriate codified process (e.g., 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ NO Go to Part II.

PART II

2.1 Could the change negatively impact the ability of a System, Structure, or Component (SSC) to perform its safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its intended safety function?

☐ YES Go to Question 2.2.

☒ NO Continue with Part III. Briefly describe rationale:

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ YES ☐ NO

Basis:

Does the revised commitment create the possibility of a new or different kind of accident from any previously evaluated?

☐ YES ☐ NO

Basis:

Does the revised commitment involve a significant reduction in a margin of safety?

☐ YES ☐ NO

Basis:

If any of the above questions are answered Yes, STOP. Do not proceed with the revision , OR discuss change with NRC and obtain necessary approvals prior to implementation of the proposed change. If all three questions are answered NO, go to Part III.
(Attach additional sheets as necessary.)

PART III

3.1 Was the original commitment (e.g., response to NOV, etc.) to restore an Obligation (i.e., rule, regulation, order or license condition)?

☐ YES Go to question 3.2.

☒ NO Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ YES Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

Rationale:

☐ NO STOP. Do not proceed with the revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ YES Go to Question 4.2.

☒ NO Go to Part V.

4.2 Has the original commitment been implemented?

☐ YES STOP, You have completed this evaluation. Revise the commitment and notify NRC of revised commitment in summary report.

☐ NO Go to Question 5.1.

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ YES Go to Question 5.2.

☒ NO STOP. You have completed this evaluation. Revise the commitment. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ YES Revise the commitment and notify NRC of revised commitment in next annual/RFO interval summary report.

☐ NO Revise commitment: no NRC notification is required:

REFERENCES

List documents (e.g., procedures, NRC submittals, etc.) affected by this change.

Doc. Number	Description

GGNS Commitment Change Evaluation Number

CCE 2003-0010

COMMITMENT CHANGE EVALUATION FORM

Commitment Number:	A-35207	**Plant Licensing Tracking Number:**	CCE 2003-0010
Source Document:	GNRO-02/00054		

Commitment: **Deletion?** ☒ **Revision?** ☐

Has the original commitment been implemented? ☐ **YES** ☒ **NO, Notify Plant Licensing**

Original Commitment Description:

Prior to RF13, GGNS will perform a parametric study at the uprated conditions to quantify the impact of TPO (Thermal Power Optimization) on GGNS wear rates and update the CHECWORKS model as necessary.

Revised Commitment Description:

Summary of Justification for Change or Deletion:

See attached.

(Attach additional sheets if necessary)

Refer to Attachment 9.4 for a flow diagram that outlines the commitment change evaluation process.

Prepared By:	Brian Ricker / Brian Ricker **Print Name/Signature**	12-2-03 **Date**
Management Approval:	MOHSEN MIRI / Mohsen Miri for C.M. Renfroe **Print Name/Signature**	12/17/03 **Date**
Plant Licensing Management Concurrence:	CABottemiller / [signature] **Print Name/Signature**	12-18-03 **Date**

✓	QA RECORD
	RT. 519.37
	NON-QA RECORD
	INITIALS WBR
NUMBER of PAGES	5
DATE	12-18-03
RELATED DOCUMENT NUMBER	

PART I

1.1 **Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Program, Fire Protection Program, or Security Plan?**

☐ **YES** **STOP. Do not proceed with this evaluation. Instead use appropriate codified process (e.g., 10 CFR 50.71(c), 10 CFR 50.54) to evaluate commitment.**

☑ **NO** **Go to Part II.**

PART II

2.1 **Could the change negatively impact the ability of a System, Structure, or Component (SSC) to perform its safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its intended safety function?**

☐ **YES** **Go to Question 2.2.**

☑ **NO** **Continue with Part III. Briefly describe rationale:**

See attached for commitment deletion justification/rationale.

2.2 **Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:**

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ **YES** ☐ **NO**

Basis:

Does the revised commitment create the possibility of a new or different kind of accident from any previously evaluated?

☐ **YES** ☐ **NO**

Basis:

Does the revised commitment involve a significant reduction in a margin of safety?

☐ **YES** ☐ **NO**

Basis:

If any of the above questions are answered Yes, STOP. Do not proceed with the revision , OR discuss change with NRC and obtain necessary approvals prior to implementation of the proposed change. If all three questions are answered NO, go to Part III.
(Attach additional sheets as necessary.)

PART III

3.1 Was the original commitment (e.g., response to NOV, etc.) to restore an Obligation (i.e., rule, regulation, order or license condition)?

- ☐ YES — Go to question 3.2.
- ☑ NO — Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

- ☐ YES — Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

Rationale:

- ☐ NO — STOP. Do not proceed with the revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

- ☐ YES — Go to Question 4.2.
- ☑ NO — Go to Part V.

4.2 Has the original commitment been implemented?

- ☐ YES — STOP, You have completed this evaluation. Revise the commitment and notify NRC of revised commitment in summary report.
- ☐ NO — Go to Question 5.1.

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

- ☐ YES — Go to Question 5.2.
- ☑ NO — STOP. You have completed this evaluation. Revise the commitment. No NRC notification required.

5.2 **Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?**

☐ **YES** **Revise the commitment and notify NRC of revised commitment in next annual/RFO interval summary report.**

☐ **NO** **Revise commitment: no NRC notification is required:**

REFERENCES

List documents (e.g., procedures, NRC submittals, etc.) affected by this change.

Doc. Number	Description

Commitment A-35207 Deletion Justification

At the time commitment A-35207 was made, GGNS plans included implementing an extended power up-rate, in addition to the App. K up-rate, during 2003. The original intent was to revise the CHECWORKS model to account for both the extended and the App. K up-rates in one model revision. However, the extended up-rate project was postponed indefinitely in early 2003.

Per the commitment description, GGNS is to perform a parametric study to quantify the impact of the TPO on GGNS wear rates and update the CHECWORKS model, if necessary. The parametric study will in fact include a revision of the CHECWORKS model to account for the increase in power level due to the App. K up-rate. As a result of initial investigation into the affect of the TPO on wear rates, the inspection scope of RF12 was increased to include highly susceptible components and components with low margins that were due for inspection within one or two outages after RF12. The inspection results noted no significant wear rate increase in any of the components which could have been attributed to the TPO. Additionally, more data on other highly susceptible components will be collected during the upcoming refuel outage (RF13-2/22/04). The RF13 results are expected to show, similar to RF12 results, that no significant wear rate increases will be found which could be definitively attributed to the TPO. These components along with all the other modeled program components will continue to be inspected per the requirements of the FAC program. The quantification of the impact of the TPO on all modeled systems will be addressed as a result of normal FAC program maintenance and a commitment was not/is not required to track completion of this task. However, to track completion of this task and for planning purposes, WT 2003-0520 has been assigned to EP&C to revise the CHECWORKS model to the up-rated conditions in order to quantify the impact on model wear rates.

GGNS Commitment Change Evaluation Number

CCE 2003-0011

COMMITMENT CHANGE EVALUATION FORM

Commitment Number:	25105	**Plant Licensing Tracking Number:**	CCE 2003-0011
Source Document:	SIL 156		

Commitment: Deletion? ☒ Revision? ☐

Has the original commitment been implemented? ☒ YES ☐ NO, Notify Plant Licensing

Original Commitment Description:

Verify neutron monitoring instrument tubes are properly seated after maintenance activities are performed in the vicinity of the tubes except for normal fuel movement.

Revised Commitment Description:

No longer performing a visual check of instrument tube seating during refueling after maintenance activities.

Summary of Justification for Change or Deletion:

The instrument bosses were chamfered after initial core loading years ago where GGNS actually experienced unseating instrument strings during fuel movements. This design change greatly reduced the likelihood of the instrument string inadvertently being dislodged. GGNS OE since that time has been no further instances of this event through a dozen refueling outages with fuel movement and other cell maintenance activities. Hence, it is prudent based on OE and modifications to remove this vendor recommended practice.

(Attach additional sheets if necessary)
Refer to Attachment 9.4 for a flow diagram that outlines the commitment change evaluation process.

Prepared By:	Paul M. Different *Paul M. Different*	12/18/03
	Print Name/Signature	**Date**
Management Approval:	Ken L. Walker *Ken Walker Ken Walker*	12/18/03
	Print Name/Signature	**Date**
Plant Licensing Management Concurrence:	Charles A. Bottemiller	12-18-03
	Print Name/Signature	**Date**

QA RECORD	✓
RT-	519.39
NON-QA RECORD	
INITIALS	CWW
NUMBER of PAGES	4
DATE	12-18-03
RELATED DOCUMENT NUMBER	

QA RECORD	
RT-	
NON-QA RECORD	
INITIALS	
NUMBER of PAGES	
DATE	
RELATED DOCUMENT NUMBER	

cww 12-18-03

PART I

1.1 **Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Program, Fire Protection Program, or Security Plan?**

☐ **YES** **STOP. Do not proceed with this evaluation. Instead use appropriate codified process (e.g., 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.**

☒ **NO** **Go to Part II.**

PART II

2.1 **Could the change negatively impact the ability of a System, Structure, or Component (SSC) to perform its safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its intended safety function?**

☐ **YES** **Go to Question 2.2.**

☒ **NO** **Continue with Part III. Briefly describe rationale:**

Commercial item only. If a string were to be dislodged it would fall into open space where fuel bundle was removed. Subsequent attempts to load the intended fuel assembly would notice the dislodged string as the string would hinder or prevent the ability to load the fuel assembly. The string would then be either reseated or replaced if damaged.

2.2 **Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:**

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ **YES** ☐ **NO**

Basis:

Does the revised commitment create the possibility of a new or different kind of accident from any previously evaluated?

☐ **YES** ☐ **NO**

Basis:

Does the revised commitment involve a significant reduction in a margin of safety?

☐ **YES** ☐ **NO**

Basis:

If any of the above questions are answered Yes, STOP. Do not proceed with the revision , OR discuss change with NRC and obtain necessary approvals prior to implementation of the proposed change. If all three questions are answered NO, go to Part III.
(Attach additional sheets as necessary.)

PART III

3.1 Was the original commitment (e.g., response to NOV, etc.) to restore an Obligation (i.e., rule, regulation, order or license condition)?

☐ YES Go to question 3.2.

☒ NO Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ YES Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

Rationale:

☐ NO STOP. Do not proceed with the revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ YES Go to Question 4.2.

☒ NO Go to Part V.

4.2 Has the original commitment been implemented?

☐ YES STOP, You have completed this evaluation. Revise the commitment and notify NRC of revised commitment in summary report.

☐ NO Go to Question 5.1.

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ YES Go to Question 5.2.

☒ NO STOP. You have completed this evaluation. Revise the commitment. No NRC notification required.

5.2 **Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?**

☐ **YES** **Revise the commitment and notify NRC of revised commitment in next annual/RFO interval summary report.**

☐ **NO** **Revise commitment: no NRC notification is required:**

REFERENCES

List documents (e.g., procedures, NRC submittals, etc.) affected by this change.

Doc. Number	Description
03-1-01-5	Delete step 2.14 from Refueling IOI