Entergy

Entergy Operations , Inc.
7003 Bald Hill Road
P.O. Box 756
Port Gibson, Mississippi 39150
Tel: 601-437-6299

Christina L. Perino
Manager
Licensing

GNRO-2011/00077

September 20, 2011

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

SUBJECT: Report of 10CFR50.59 Evaluations and Commitment Changes – July 1, 2010 through June 30, 2011
Grand Gulf Nuclear Station, Unit No. 1
Docket No. 50-416
License No. NPF-29

Dear Sir or Madam:

Pursuant to 10CFR50.59(d)(2) Entergy Operations, Inc. hereby submits a description of 50.59 and commitment change evaluations for the period of July 1, 2010 through June 30, 2011 in Attachment 1. Attachment 2 contains copies of the full evaluations.

If you have any questions or require additional information, please contact Dennis Coulter at 601-437-6595.

This letter does not contain any commitments.

Sincerely,

CLP/DMC

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

cc: (See Next Page)

cc: NRC Senior Resident Inspector
Grand Gulf Nuclear Station
Port Gibson, MS 39150

U.S. Nuclear Regulatory Commission
ATTN: Mr. Elmo E. Collins, Jr. (w/2)
Regional Administrator, Region IV
612 East Lamar Blvd, Suite 400
Arlington, TX 76011-4125

U. S. Nuclear Regulatory Commission
ATTN: Mr. Alan Wang, NRR / DORL (w/2)
Mail Stop OWFN 8 B1
Washington, DC 20555-0001

Attachment 1 to GNRO 2011-00077

Table of Contents

Grand Gulf Nuclear Station
10CFR50.59 Evaluation and Commitment Change Evaluation Report
for the Period July 1, 2010 through June 30, 2011

10CFR50.59 Evaluations

Evaluation Number	Initiating Document	Description
SE 2011-0001-R00	Licensing Basis Document Change Request (LBDCR) 2011-031	Technical Specification Bases change to drywell bypass leakage test start pressure from > 3 psid to 3 +/- 0.05 psid.
SE 2011-0002-R00	Engineering Change (EC) 23898	Change the safety classification of the Dryer and Separator strong back from class 3 to "other" to allow use of a replacement strong back.
SE 2011-0003-R00	EC 23790 and EC 23791	Modifications to the fuel pool cooling and cleanup system following the station extended power uprate.

Commitment Change Evaluations

Commitment Number	Source Document	Description
CCE 2010-0009	AECM 84/0508	Procedures provide a method for cross referencing technical manuals. This commitment has been tracked more than two years and is well established in site processes. Tracking is no longer required.
CCE 2011-0001	CNRO-97/00004 – Entergy 180 Day Response to Generic Letter 96-05	Revises commitment to make a one-cycle one-time extension to the Static Test Frequency interval for the RCIC Minimum Flow to Suppression Pool Isolation Valve that will also correspond to a one-time 6 month extension to the maximum interval (10 years) between static tests.
CCE 2011-0002	CNRO-97/00004 – Entergy 180 Day Response to Generic Letter 96-05	Revises commitment to make a one-cycle one-time extension to the Static Test Frequency interval for the RCIC Turbine Exhaust Inboard Vacuum Breaker Valve that will also correspond to a one-time 4 month extension to the maximum interval (10 years) between static tests.

Attachment 2 to GNRO 2011-00077

10CFR50.59 Evaluations
and
Commitment Change Evaluations

GGNS 50.59 Evaluation Number

SE 2011-0001-R00

I. OVERVIEW / SIGNATURES[1]

SE 2011-0001-R00

Facility: Grand Gulf Nuclear Station **Evaluation # / Rev. #:**

Proposed Change / Document: LDC 2010-031 for Technical Specification Bases 3.6.5.1.1

Description of Change: Technical Specification Bases 3.6.5.1.1 describes the drywell bypass leakage test. Procedure 06-ME-1M10-O-0003 was changed based on calculation MC-Q1M24-08008 with EC 8900. The method of evaluation changed from a constant pressure makeup test to a pressure decay test. The Technical Specification Bases were not changed at that time. This LDC changes the Technical Specification Bases to allow a pressure decay test and allow nominal start pressure of 3 ± 0.05 psid per procedure 06-ME-1M10-O-0003.

Is the validity of this Evaluation dependent on any other change? ☐ Yes ☒ No

If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval? ☐ Yes ☒ No

Preparer: Robert W. Fuller / Robert W Fuller / EOI / DE - Mech-Civil / 1-21-11
Name (print) / Signature / Company / Department / Date

Reviewer: Al Evans / [signature] / EOI / DE-Mech-Civil / 1-21-11
Name (print) / Signature / Company / Department / Date

OSRC: Marty Richey / [signature] / 1/26/11
Chairman's Name (print) / Signature / Date

OSRC Meeting # 02-2011

QA RECORD
FILE 15.14.33
NON-QA RECORD
INITIALS
NUMBER OF PAGES 4
DATE 2-7-11
RELATED DOCUMENT NUMBER:

[1] Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

II. 50.59 EVALUATION

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 UFSAR?

☐ Yes
☒ No

BASIS: The drywell leakage test is not an accident initiator so there is no minimal increase in the frequency of occurrence of an accident.

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 UFSAR?

☐ Yes
☒ No

BASIS: The measurement of drywell leak tightness is not a contributor to the likelihood of occurrence of a malfunction of the drywell.

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

☐ Yes
☒ No

BASIS: The consequences of an accident in the drywell are evaluated in UFSAR Chapter 15. The accident assumes a 3000 cfm flow rate for drywell to containment leakage flow. This flow rate is based on a testing $\frac{A}{\sqrt{k}} = 0.09$ ft^2. The design $\frac{A}{\sqrt{k}} = 0.9\,\text{ft}^2$. The pressure associated with the $\frac{A}{\sqrt{k}}$ is 3 psid. This is the pressure to expose the vents.

In the past GGNS has performed a constant pressure makeup test. Under these conditions, the drywell is maintained at approximately 3 psid and air is added to the drywell to maintain the pressure differential. The amount of air added to the drywell to maintain the pressure differential is directly measured to determine the leakage rate. This is compared to the acceptable leakage rate.

The proposed test is a pressure decay test. This initializes the drywell at a pressure differential between containment not to exceed the pressure that could cause the vents to be exposed.

Changing from a constant pressure makeup test to a pressure decay test does not affect the overall system performance in a manner that could lead to an accident, the proposed change to the Technical Specification basis would not increase the overall consequences of an accident previously evaluated in the UFSAR.

Changing the test to start at less than 3 psid will not increase the consequences of an accident. The current margins for radiation dose will be maintained and will not affect the consequences of an accident previously evaluated in the FSAR.
The basis for the change is the pressure decay curve provided by MC-Q1M24-08008. The pressure decay curve demonstrates acceptability of the drywell bypass leakage test. The acceptance criteria is the drywell pressure decay is above the curve provided by the calculation.

The method still ensures that $\frac{A}{\sqrt{k}}$ limits are met.

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 UFSAR? ☐ Yes ☒ No

BASIS: The consequences of an accident in the drywell are evaluated in UFSAR Chapter 15. The accident assumes a 3000 cfm flow rate for drywell to containment leakage flow. This flow rate is based on MELCOR analysis that was explicitly reviewed and approved by the NRC. The thermal hydraulic analysis assumes a design bypass leakage limit of $\frac{A}{\sqrt{k}} = 0.9\,\text{ft}^2$. The analysis for the pressure drop test assumes a testing $\frac{A}{\sqrt{k}} = 0.09$ ft^2.

The pressure associated with the $\frac{A}{\sqrt{k}}$ is 3 psid. This is the pressure to expose the vents.

This pressure differential is not being changed for accident scenarios and consequences. The pressure being tested at will be changed to allow margin between the current Technical Specification allowable of 3 psid and the pressure to expose the vents at 3.02 psid. Currently 06-ME-1M10-O-0003 allows the pressure to vary as much as 2.95 psid. This pressure will be adequate to initiate the pressure decay test.

The capability of the drywell to containment pressure boundary is not affected. There will be no increase in the consequences to the structure of the drywell by changing the test methodology from a constant pressure test to a pressure decay test. Changing the starting pressure to a pressure less than the pressure to expose the drywell/containment vents does not affect the consequences of the accident.
The basis for the change is the pressure decay curve provided by MC-Q1M24-08008. The pressure decay curve demonstrates acceptability of the drywell bypass leakage test. The acceptance criteria is the drywell pressure decay is above the curve provided by the calculation.

If $\frac{A}{\sqrt{k}}$ is met then the drywell/containment is within design basis.

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

BASIS: This evaluation does not involve an initiator or failure not considered in the FSAR. The change proposes to change the drywell bypass leakage test from a constant pressure test to a pressure decay test. With this change in test, the starting pressure of the test will be allowed to go below 3 psid. The 3 psid is based on the drywell to containment pressure that will expose the vents. Preoperational testing determined that 3.02 psid between containment and drywell will cause air to pass through the vents. This design feature of the drywell to containment is not being changed. Changing the type of test does not create the 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 UFSAR? ☐ Yes ☒ No

BASIS: This evaluation does not lead to a failure mode with a different result previously evaluated in the FSAR. The change proposes to change the drywell bypass leakage test from a constant pressure test to a pressure decay test. With this change in test, the starting pressure of the test will be allowed to go below 3 psid. The 3 psid is based on the drywell to containment pressure that will expose the vents. Preoperational testing determined that 3.02 psid between containment and drywell will cause air to pass through the vents. This design feature of the drywell to containment is not being changed. Changing the type of test does not create the possibility for an accident of a different type than any previously evaluated in the FSAR. Since $\frac{A}{\sqrt{k}}$ is preserved, there is no different malfunction postulated.

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

BASIS: $\frac{A}{\sqrt{k}}$ is not changed. The proposed measurement method preserves the limit. No design basis limit or fission barrier is being compromised as a result of a change in the type of test from a constant pressure test to a pressure decay test.

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

BASIS: Changing from a measured leakage test to a drywell pressure drop test is essentially the same because calculation MC-Q1M24-08008 demonstrates that assuming the design drywell bypass leakage limit that the drywell would depressurize in 32.6 minutes. The results from the 2008 calculation is valid for all pressures between 0 psid and 3 psid drywell to containment pressure differential. The drywell bypass leakage assumed in the calculation is 3 psid for 3456 scfm. This leakage bounds the leak rates received in previous drywell bypass leakage tests. The pressure decay curve from RF16 was acceptable to the pressure decay curve in calculation MC-Q1M24-08008. The results of the RF16 pressure decay curve were above the calculations curve which is acceptable.

If any of the above questions is checked "Yes," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure EN-LI-103.

GGNS 50.59 Evaluation Number

SE 2011-0002-R00

I. OVERVIEW / SIGNATURES[1]

SE 2011-0002 R00

Facility: Grand Gulf Nuclear Station **Evaluation # / Rev. #:** 2011-002 Rev. # 0

Proposed Change / Document: Table 3.2-1 and 9.1-3

Description of Change: Change the Safety Classification of the Dryer and Separator Strong back from Class 3 to "other" for replacement Steam Dryer and Separator Strong-back. This change will allow the replacement of the current Steam Dryer and Separator strongback (special lifting device) with the replacement Steam Dryer and Separator strong back (special lifting device) that is non-safety related.

Is the validity of this Evaluation dependent on any other change? ☐ Yes ☒ No

If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

N/A

Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval? ☐ Yes ☒ No

Preparer: Vikram D. Patel/ SEE IAS / SHAW GROUP/ Design Engineering (Civil)) / 03-07-2011
Name (print) / Signature / Company / Department / Date

Reviewer: Michael Withrow/ SEE IAS / Entergy Corporation / Eng Mgmt /03/10/2011
Name (print) / Signature / Company / Department / Date

OSRC: M. Rickey / [signature] 3/30/11
Chairman's Name (print) / Signature / Date

OSRC Meeting # 2011 - 006

✓	QA RECORD
	RT 014.33
	NON-QA RECORD
	INITIALS
NUMBER of PAGES	4
DATE	4-4-11
RELATED DOCUMENT NUMBER	EC 23898

[1] Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

II. 50.59 EVALUATION

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 UFSAR?

☐ Yes
☒ No

BASIS: The only accident evaluated in the UFSAR that could be impacted by the safety classification change to the steam dryer and separator strongback is the design basis fuel handling accident described in UFSAR Section 15.7.4. The dryer strongback is not considered in this analysis since it cannot be used to handle fuel. However, in accordance with UFSAR Section 9.1.4.2.5.6, the dryer strongback is a special lifting device used to transport the steam dryer and the shroud head with the moisture steam separators between the reactor vessel and the storage pools. As such, the strongback is used to lift heavy loads (*i.e.*, loads in excess of 1,140 lbs. defined in the Technical Requirements Manual as the light load limit for postulated fuel handling accidents) as described in UFSAR section 9.1.4.2.2.5. The load handling requirements of UFSAR sections 15.7.4 and 9.1.4.2.2.5 are not dependent on the safety classification of the dryer strongback as long as the strongback still meets the requirements specified in UFSAR section 9.1.4.2.5.6 (*e.g.*, single failure proof design, structural safety factor margins, proof testing, and post-test weld inspections). In addition to meeting the single failure criteria, the new strongback is constructed to the requirements of ANSI N14.6-1978 which ensures compliance with all other requirements of UFSAR section 9.1.4.2.5.6, including stress design margins, load testing, and inspections. The failure frequency of the new strongback is thus not increased. Therefore, downgrading the safety classification of the dryer and separator strongback will not result in an increase in the frequency of occurrence of an accident previously evaluated in the UFSAR.

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 UFSAR?

☐ Yes
☒ No

BASIS: UFSAR Section 9.1.4.2.2.1 summarizes the analyses involving the dropping of the steam dryer assembly from its highest lift point into the open reactor vessel and the dropping of the shroud head/steam separator assembly from its highest lift point into the open reactor vessel. While the replacement steam dryer is heavier than the existing steam dryer, the weight of the new dryer is still less than that of the steam separator head which is the bounding weight considered in the design of the strongback and postulated load drop. UFSAR Section 9.1.4.2.2.5 documents compliance with NUREG 0612, Control of Heavy Loads at Nuclear Power Plants. UFSAR Appendix 9D states that NUREG-0612 provides a defense in depth approach for controlling the handling of heavy loads so that load handling events have a very low probability of occurrence. In addition, Section 5.1.1 of NUREG-0612 requires that special lifting devices satisfy the guidelines of ANSI N14.6-1978. This standard specifies design, fabrication, acceptance testing, maintenance, and continuing compliance requirements for special lifting devices used to move heavy loads. Neither NUREG 0612 nor ANSI N14.6 1978 requires a safety classification for special lifting devices. Since the new strongback continues to meet single failure requirements and complies with NUREG-0612 and ANSI N14.6-1978, the likelihood of a component failure during load handling is not increased. Thus, downgrading the safety classification of the dryer and separator strongback will not result in an increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR.

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

☐ Yes
☒ No

BASIS: The dryer and separator strongback is a passive special lifting device that does not have any defined safety function and performs no role in mitigating the consequences of an accident. In addition, the requirements of UFSAR Appendix 9D and NUREG-0612 that provide defense in depth associated with heavy load lifts for the protection of safe shutdown and accident mitigation equipment are not affected by the replacement dryer strongback. Therefore, this change will not result in an increase in the consequences of an accident previously evaluated 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 UFSAR? ☐ Yes ☒ No

BASIS: UFSAR Section 9.1.4.2.2.1 summarizes the analyses involving the dropping of the steam dryer assembly from its highest lift point into the open reactor vessel and the dropping of the shroud head/steam separator assembly from its highest lift point into the open reactor vessel. These analyses concluded that the postulated drop of either the dryer or the separator head would not affect the ability of the plant to remain in a safe shutdown condition or result in the release of significant amounts of radioactive materials. Changing the safety classification of the dryer strongback does not impact any input parameters or assumptions used in these analyses. The strongback also performs no accident mitigation function. Thus, this change will not result in an increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR.

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

BASIS: In accordance with UFSAR Section 9.1.4.2.5.6, the dryer and separator head strongback is a special lifting device that performs no safety function. Changing the safety classification of this equipment will not change its function nor would it introduce any new accident precursor. Therefore, this change does not create the possibility for an accident of a different type than any previously evaluated 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 UFSAR? ☐ Yes ☒ No

BASIS: UFSAR Section 9.1.4.2.2.1 summarizes the analyses involving the dropping of heavy loads from their highest lift point into the open reactor vessel. This section summarizes the most serious consequence resulting from the accidental drop of a heavy load as a severe plastic deformation of the vessel top flange. The analysis also concludes that the drop will not produce any vessel leaks or result in the release of significant amounts of radioactive material. The new strongback still meets the single failure proof design requirements as well as all of the requirements for design safety margins, materials of construction, tests, and inspections as the original strongback. The change in the safety classification does not change the form, fit, or function of the strongback. Thus, this change will not create a possibility for a malfunction of a structure, system, or component important to safety with a different result than previously evaluated.

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

BASIS: The dryer and separator strongback is a passive lifting device that does not have any defined safety function and is not part of any fission product barrier nor does it affect the design basis limit for any fission product barrier. Therefore, changing the safety classification of the dryer and separator strongback does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

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

BASIS: The proposed UFSAR revision changes the safety classification of the dryer and separator strongback from Class 3 to "other." The safety classification of this special lifting device is not considered in any UFSAR evaluation or in any method of evaluation described in the UFSAR. The codes and standards to which the new strongback is constructed are no different than those previously described in the UFSAR. Thus, this change will not result in a departure from a method of evaluation described in the UFSAR.

If any of the above questions is checked "Yes," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure EN-LI-103.

GGNS 50.59 Evaluation Number

SE 2011-0003-R00

I. OVERVIEW / SIGNATURES[1]

SE 2011-0003 R00

Facility: Grand Gulf Nuclear Station

Evaluation # / Rev. #: 2011-0003 / Rev. 00

Proposed Change / Document(s): EC# 23790 GGNS EPU Fuel Pool Cooling & Cleanup Hx Replacement Train A
EC# 23791 GGNS EPU Fuel Pool Cooling & Cleanup Hx Replacement Train B

Description of Change:

This change involves modifying the two redundant trains of the Fuel Pool Cooling and Cleanup (FPCCU) System and associated cooling water systems [Component Cooling Water (CCW) and Standby Service Water (SSW)] to increase the Spent Fuel Pool (SFP) decay heat removal capacity. The FPCCU system increased cooling capacity will be required following Extended Power Uprate (EPU) implementation in order to maintain SFP water temperatures below the temperature limits set by the SFP design criteria and licensing limits, for all limiting scenarios. More specifically, the proposed change includes the following:

- Replacement of the existing Shell & Tube heat exchangers in the FPCCU System with higher thermal capacity Alfa Laval Plate & Frame heat exchangers (ASME III, Division 1, Class 3).
- Modifying the supply and return piping to the FPCCU heat exchangers in FPCCU System to accommodate the new heat exchanger design and layout
- Modifying the common supply and return piping to the FPCCU heat exchangers in the CCW/SSW system(s) to accommodate the new heat exchanger design and layout
- Installation of Simplex Strainers (ASME III, Division 1, Class 3) on the cooling water inlet side of the FPCCU heat exchangers to protect the Plate & Frame FPCCU heat exchangers from iron oxide (rust) particles greater than 0.125 in. coming from the SSW system, which can block flow paths and degrade thermal performance.

This modification does not increase the system susceptibility to hydraulic transients such as water hammer, startup transients, etc. therefore hydraulic transients are not the focus of this evaluation.

A Process Applicability Determination (PAD for EC# 23790) was performed in regards to the proposed change that identified both a change to a Licensing Bases Documents (LBD), namely the UFSAR, as well as a potential adverse effect which is negligible in the current FPCCU System design. The adverse effect introduced by the proposed change is an aggravated failure mode with regards to flow blockage inherent in the smaller flow passages of a Plate & Frame heat exchanger type design. If the heat exchanger flow rate is reduced from blockage the thermal hydraulics of the system are adversely affected and the result is a reduction in the heat removal capability of the FPCCU System. The result is the potential of no longer being able to maintain the SFP water temperature within the required limits, either during normal operations or an accident. During normal operations, the cooling water is supplied from the Component Cooling Water (CCW) System, and during accident conditions the cooling water is supplied from the Standby Service Water (SSW) System.

The flow blockage of the Plate & Frame heat exchanger during normal operation, with CCW as the cooling fluid, is mitigated by existing water chemistry control. Periodic testing of the FPCC heat exchangers will be performed as part of the heat exchanger thermal performance monitoring program to comply with NRC Generic Letter 89-13. This testing will trend heat exchanger performance and fouling levels to determine when cleaning is required. The FPCCU system water is normally circulated through a filter/demineralizer to maintain water purity and is demineralized. The CCW is a closed loop, cooling system that uses chemically controlled water with the capability for chemical additions and further control. The susceptibility to thermal degradation due to flow blockage of the nature described above is low for the FPCCU water in a forced circulation mode of operation given the above preventative actions. In the case of an alignment to the SSW system during accident conditions or SSW flow balancing activities, the increase in potential FPCCU heat exchanger cooling water flow blockage is mitigated by the high fluid turbulence caused by the corrugated and parallel heat exchanger plates, thus creating a greater resistance to flow blockage, as well as by the use of proposed strainers on the cooling water side.

The SSW system is the open loop cooling water system which acts as the ultimate heat sink (UHS) for the plant in the case of a design basis accident. The SSW system provides cooling water to several safety related heat exchangers, including the FPCCU heat exchangers. The SSW water is cooled by two cooling towers, and is then collected in two concrete basins below the towers. The SSW pumps take suction from their respective basin, and provide flow back to the

[1] Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form. When using other manual methods, type/print Name, electronic method of signature, Company, and Department (and Date, if desired), where applicable.

✓	QA RECORD
	RT. B 14.33
	NON-QA RECORD
✓	INITIALS
NUMBER of PAGES	8
DATE	5-17-11
RELATED DOCUMENT NUMBER-	EC 23790 and 23791

piping network supplying the FPCCU and other heat exchangers. The SSW system is chemically treated to minimize corrosion and biological flow blockage of equipment. The SSW basin is subject to the intrusion of foreign material. In order to prevent clogging of the safety related SSW heat exchangers, the pump suction is protected by an existing debris screen consisting of stainless steel plates designed to remove particulate 1/8th inch or greater in size (the minimum heat exchanger plate gap is 0.157", which is greater than 1/8th inch). The heat exchanger plates are designed for a maximum particle size of 1/8th inch. An additional flow blockage mode to the aforementioned can occur that can adversely affect heat exchanger performance. The corrosion products in the SSW supply piping can be released to the FPCCU heat exchangers. Simplex strainers are being installed to address the flow blockage mode.

The strainers being installed are QA Category 1, Nuclear Safety Related, and Safety Class 3 Simplex Strainers. The flow blockage rate of the strainers is therefore the primary adverse effect requiring evaluation; however additional topics are evaluated herein for completeness. The design of the Simplex Strainers assumes the maximum predicted particulate loading from the total liftoff of metal oxides in the system piping. This liftoff is expected to accumulate in the strainers a short period of time after SSW system startup. This particulate loading is not continuous but rather the result of stagnant particulate accumulating in a burst after forced circulation is introduced. The maximum pressure drop from the resulting flow blockage in the strainers is accounted for in the hydraulic evaluations and the results shown to meet the existing design bases of the SSW system.

The overall scope of this 50.59 evaluation is to demonstrate that the introduction of the new FPCCU Plate & Frame heat exchangers and associated strainers do not result in an unresolved safety question since the new failure mode inherent in the proposed design can be effectively mitigated with a combination of existing water chemistry control, periodic maintenance, heat exchanger design and modifications to the common cooling water system supply by the addition of strainers. The complete modification, heat exchangers and strainers, are therefore shown to be adequate in meeting the requirements of both the existing design bases and the expected EPU conditions. The primary sources for demonstrating design adequacy of the proposed changes are the conformed specifications utilized in the construction and acquisition of both the new Plate & Frame heat exchangers (SPEC-10-00021-G) and the cooling water supply Simplex strainers (SPEC-11-00017-G).

Several items related to the proposed change have already been evaluated in the PAD and did not warrant further 50.59 reviews, therefore they are not included in the scope of this document. The items screened out are listed herein for completeness:

- FPCCU Piping and CCW/SSW Piping Qualification and Associated Supports
- Civil / Structural Qualification
- FPCCU Heat Exchanger Haul Path Evaluation
- Radiological Impact Review

Individual evaluations for the aforementioned excluded topics can be found either in the PAD and/or with independent supplemental evaluations as a function of the pertinent discipline.

The following acronyms and component numbers are utilized throughout this evaluation:

Acronyms:

FPCCU	Fuel Pool Cooling & Cleanup
CCW	Component Cooling Water
SSW	Standby Service Water
EPU	Extended Power Uprate
LBD	Licensing Bases Document
DBD	Design Bases Document
EC	Engineering Change
PAD	Process Applicability Determination
HXs	Heat Exchangers
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
UFSAR	Updated Final Safety Analysis Report
TS	Technical Specifications
OLM	Operating License Manual
TRM	Technical Requirements Manual
FHA	Fire Hazards Analysis

Component Numbers:

1G41B001A/B	Existing FPCCU HXs
1G41B005A/B	New FPCCU HXs
1P42D005A/B	Proposed Strainers
Q1P42-F028A/B	CCW Inlet Isolation Valves

An extensive review of existing GGNS operating, design, and licensing bases information was performed and compared to expected EPU conditions to support the proposed change. The review included the following:

- GGNS UFSAR Sections:
  - Chapter 1: 1.2.1, 1.2.2.8.1, 1.2.2.8.2, 1.2.2.8.20,
  - Chapter 3: 3.1.2.6.2, 3.2.4.2.1, Table 3.2-1, 3.9.3.1.1.1.22, 3.9.3.4.3, Table 3.9-2X, Table 3.9.3C, Table 3.9.33,
  - Chapter 7: 7.6.1.9, 7.6.1.10, 7.6.2.8, Table 7.6-10, Table 7.6-13
  - Chapter 9: Table 9.1-12, 9.1.3, 9.2.1, 9.2.2, Table 9.2-16, Table 9.2-17
  - Chapter 14: 14.2.12.1.22,
  - Chapter 15: 15A.6.2.3.14
- GGNS Technical Specifications and Basis:
  - 3.7, B 3.7,
  - 6.7.4, B 6.7.4
- GGNS Plant Operating Manual
  - 05-1-02-V-1 R021 Off-Normal Event Procedure "Loss of Component Cooling Water"
  - 05-1-02-V-11 R031 Off-Normal Event Procedure "Loss of Plant Service Water"
  - 06-OP-1G41-Q-0001 Surveillance Procedure "FPCCU Pump and Valve Operability Test"

Based on the review of the documents above the following licensing and design conditions exist with which to compare the impact of the proposed change:

- From the plant Technical Specifications LCO 6.7.4 requires that the SFP water temperature be maintained less than or equal to 140°F whenever there is irradiated fuel in the SFP.

The existing thermal and hydraulic calculations associated with the FPCC heat exchangers were revised to reflect the new configurations. The revisions demonstrate that with the installation of the new heat exchangers and the cooling water side strainers, the FPCCU System will be capable of:

- Dissipating the peak EPU fuel pool heat load of 27.4 MBtu/hr while simultaneously maintaining the SFP temperature below 140°F with either SSW or CCW cooling water.
- Maintaining the spent fuel pool temperature below 140°F with only one FPCCU train after approximately 70 days.
- Maintaining the spent fuel pool temperature below 150°F during a Loss of Coolant Accident (LOCA) concurrent with a Loss of Offsite Power (LOOP) and the failure of a diesel generator (Note: this cannot be accomplished until just after 15 days after a normal shutdown).

Therefore, from a thermal/hydraulic perspective the new heat exchangers with the associated strainers have the ability to meet the design function of the FPCCU System as described in the UFSAR.

The FPCCU System does not in itself mitigate the consequences of an accident as described in Chapter 15 of the UFSAR; however it is essential in maintaining the SFP water temperature within design and licensing limits even during a LOCA via the removal of decay heat from irradiated fuel stored in the spent fuel pool. The effects on several of the accidents discussed in Chapter 15 of the UFSAR were evaluated with respect the proposed change.

There are no safety functions being added, altered, or deleted by the proposed change. Nor are there any seemingly positive aspects of the proposed change which under unusual circumstances adversely affect safety.

Is the validity of this Evaluation dependent on any other change? ☐ Yes ☒ No

If "Yes," list the required changes/submittals. The changes covered by this 50.59 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

Note: Although the impetus for the proposed changes is to accommodate the increase in Spent Fuel Pool decay heat loads at Extended Power Uprate conditions, the license amendment approval for the EPU itself is not a prerequisite for implementation of the modification. The capabilities of the proposed change/modification exceed the existing system requirements and justify pre-EPU implementation if so desired. Therefore, NRC approval of the EPU license amendment request is not a pre-requisite for the proposed changes evaluated herein.

Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval? ☐ Yes ☒ No

Preparer: Santiago Velez /See IAS / Shaw Group Inc. / Thermal Performance Group/ 05-02-11
Name (print) / Signature / Company / Department / Date

Reviewer: G.E. Broadbent/ See IAS / Entergy / EPU Engineering / 05-02-11
Name (print) / Signature / Company / Department / Date

OSRC: Marty Richey / 5/3/11
Chairman's Name (print) / Signature / Date **[GGNS P-33633, P-34230, & P-34420; 3 P-151]**
008-2011
OSRC Meeting #

II. 50.59 EVALUATION [10 CFR 50.59(c)(2)]

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 UFSAR?

☐ Yes
☒ No

BASIS: A review of the UFSAR Chapter 15 was conducted to determine if the frequency of any UFSAR postulated accidents may be impacted by the modification of the FPCCU System and the addition of the associated cooling water (CCW/SSW) supply strainer.

UFSAR Section 15.7.4 discusses a fuel bundle drop accident onto stored irradiated fuel. Additionally, Section 15.7.5 discusses a Spent Fuel Cask Drop Accident. Both accidents do not involve the FPCCU System. Accidents involving the FPCCU System are discussed neither in this section nor in any other UFSAR Chapter 15 sections.

UFSAR Section 15.6.5 discusses a LOCA and the resulting requirement of adequate decay heat removal. In this context, the SSW system operability is essential in mitigating the consequences of a LOCA type accident via the removal of decay heat from the RHR system and from other emergency safeguard equipment. UFSAR Section 9.1.3 states that the SSW was analyzed for Spent fuel pool heat removal following a LOCA with LOOP to ensure adequate heat removal. The addition of a strainer in the CCW/SSW systems to the common supply of the FPCCU HX does not increase the frequency of occurrence of a LOCA as currently established. The overall performance and reliability of the SSW system during a LOCA is similarly not affected since the replacement FPCCU heat exchangers and the new strainer do not affect SSW system performance, which is flow balanced in advance to ensure adequate flow to other essential loads. The CCW system is not evaluated since it is considered out of service during an accident and is therefore not credited.

The proposed changes add only passive components that do not affect the electrical loadings in either the FPCCU or SSW systems. The proposed changes do affect the structural loadings with regards to floor loadings, seismic responses, etc. but the effect has been determined to be acceptable, therefore no new failure modes in regards to causing an accident are introduced. The replacement FPCCU heat exchangers and strainers are designed to remain intact and functional in the event of a safe shutdown earthquake.

While the majority of the work which will install the new heat exchangers, strainers, and piping will be performed while the plant is online, all work which requires dismantling an operating system or component, or breaches an existing system pressure boundary, will be performed while the plant is offline. Therefore, there is no increase in the frequency of occurrence of any accident discussed in the UFSAR.

The proposed change is similar in complexity and human factor considerations as the existing system, as the passive components being added do not add any additional operator burdens or challenges during an accident and in this sense can be considered equivalent to the existing system.

In summary, the FPCCU and associated cooling systems are credited for maintaining spent fuel pool temperatures during normal and postulated accidents; however, the FPCCU and associated cooling systems do not initiate any accidents evaluated in the UFSAR and accordingly do not change the frequency of any postulated accidents.

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 UFSAR? ☐ Yes ☒ No

BASIS: The FPCCU System is considered in the prevention and mitigation of an analyzed malfunction or accident, crediting the removal of decay heat from irradiated fuel stored in the spent fuel pool. The proposed changes, including the addition of new FPCCU heat exchangers and the addition of strainers on the cooling water supply have been evaluated from both a seismic and structural standpoint and have been found to be adequate in their proposed design and construction. No adjacent systems that are credited in UFSAR scenarios will be affected by the proposed change from this perspective.

The replacement of the existing Shell & Tube FPCCU heat exchangers with new Plate & Frame FPCCU heat exchangers will not affect the separation criteria between separate divisions of equipment and will retain the dual train configuration currently employed, including redundancy and cross-tie capabilities. The required hydraulic performance of the FPCCU pumps and associated diesel loading requirements are unaffected by the proposed change.

The proposed strainer design is sufficient in accommodating the effects of long term flow blockage that arise when the SSW system is in a stagnant configuration. More specifically, the pressure drop that arises from the periodic use of the SSW system across the strainer has been accounted for in the thermal hydraulic performance of the FPCCU System and shown to be adequate for the most limiting conditions, including during a LOCA concurrent with a LOOP. The strainers prevent the introduction of particles that would be large enough to block the cooling water paths in the FPCCU heat exchangers. Additionally, the pressure drop from flow blockage does not result in increasing the likelihood of occurrence of a malfunction of an SSC discussed in the UFSAR.

During a LOCA/LOOP scenario, the SSW system is assumed to no longer contribute additional sources of particulate fouling matter to the strainer beyond the initial burst. This assumption is reasonable since the long term cooling scenario is still short in comparison to the rate of particulate formation in the SSW system.

In summary, there is no increase in the likelihood of occurrence of a malfunction of the FPCC system and associated cooling systems due to the proposed modification. The new filter has been sized to capture particles which could clog the replacement plate and frame FPCC heat exchanger and the SSW system has been evaluated for the maximum predicted filter loading. The existing train redundancy of the FPCCU System ensures adequate performance under all modes of operation, even assuming single failure criteria. This condition remains unaffected by the introduction of the proposed changes and can be considered equivalent to the existing design.

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

BASIS: As discussed in the response to Question 1 above, the FPCCU system and the cooling of FPCCU by the SSW is only discussed in the LOCA accident scenario. Accident mitigation via the SSW system as the primary heat sink for the RHR system and other emergency safeguard equipment is unaffected by either the replacement of the FPCCU HXs or the addition of the cooling water Simplex Strainers since the SSW system is flow balanced prior to operation. The flow balancing ensures adequate cooling to all safety systems by the SSW system in the event of an accident even when the Simplex Strainers are at their maximum fouled condition. In addition, as discussed in the response to Question 2 above, the FPCCU System and its SSW cooling water maintain SFP water temperature within design limits even during a LOCA; hence the SFP fuel cladding will remain unaffected.

The modification does not affect the containment boundary or any safety systems that circulates the primary fluid outside the containment following an accident. Since the accident mitigation is not affected, including any leakage paths outside the containment, there is no increase in the consequences of an accident previously evaluated 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 UFSAR? ☐ Yes ☒ No

BASIS: As discussed in the response to Question 2 above, the malfunctions of both the FPCCU and SSW systems are already considered and addressed via train redundancy in the existing licensing and design bases of the plant. The addition of the proposed change with respect to the heat exchanger upgrades does not change the existing malfunctions and can be considered equivalent in nature to the existing system. With respect to the Simplex Strainers, the potential for flow obstruction as a hypothetical malfunction in the SSW system does not increase the consequences of a malfunction since the system is flow balanced and will be designed to account for the differential pressure across the strainer due to the maximum predicted particle loading.

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

BASIS: Both the FPCCU and SSW systems remain unchanged in their underlying design function. The systems will not cause any new accidents different than those already evaluated in the UFSAR. These systems will continue to perform their safety functions after this modification.

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 UFSAR? ☐ Yes ☒ No

BASIS: As discussed in the response to Question 2 above, the malfunctions postulated for FPCCU result in the loss of one of two redundant trains. The proposed change does not adversely affect any SSC that is important to safety. The most limiting malfunction involving both the FPCCU and SSW system has been shown via evaluation to remain unaffected. The FPCCU System will continue to meet its design bases (maintaining SFP temperature below limits) during a LOCA. The SSW system is flow balanced to ensure adequate flow is delivered to all essential loads even with the postulated loading on the strainers; therefore the system will provide the heat sink capability assumed in the various scenarios as a boundary condition (i.e. design input), thereby ensuring an adequate plant response predicted for all scenarios at both pre- and post-EPU conditions. The response to all scenarios at EPU is not a function of the proposed changes discussed herein and approval of said responses by the appropriate licensing amendment is independent of this evaluation.

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

BASIS: Fission product barriers include fuel rod cladding, the reactor coolant system, and the primary/secondary containment buildings. Design basis limits would be those limits such as temperature, pressure, enthalpy, and strain. Modification of the FPCCU and CCW/SSW systems does not affect any design basis limits for any of the fission product barriers. The FPCCU System is responsible for maintaining SFP water temperature within design limits of the SFP liner, racks, and associated criticality analyses. Calculations have demonstrated that the FPCCU System is capable of maintaining the SFP temperature below design limits for all operating conditions, including during a LOCA. The result is that the local fuel cladding of spent nuclear fuel is unaffected by the proposed change since the ambient cooling medium will remain within the limits required.

The SSW system operability for all required modes of operation is unaffected by the proposed change since the system is flow balanced to ensure essential loads receive adequate cooling water and can continue to support the removal of decay heat and the operation of the emergency safeguard equipment during an accident. Since the LOCA and other accident analyses are unaffected, the proposed changes to both the FPCCU HXs and the CCW/SSW strainers will not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

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

BASIS: A review of the UFSAR sections listed previously was performed to determine whether this change departs from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses. Although an industry-standard computer code is required for determining the heat rejection capability of the new plate-and frame heat exchangers, this method of evaluation is not described in the UFSAR 9.1.3 other than merely reporting the performance of the existing components. Likewise, the piping design methods are also unaffected. Therefore, this change does not constitute a change to any calculation or methodical framework used in the safety analyses as described in the UFSAR.

If any of the above questions is checked "Yes," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure EN-LI-103.

GGNS Commitment Change Evaluation Number

CCE 2010-0009

COMMITMENT CHANGE EVALUATION FORM

LRS/CMS I.D. P-24008 Site Licensing Tracking Number: CCE2010-0009

Source Document / Date: AECM84/0508

Commitment:
- ☐ Closure/Deletion
- ☒ Historic/Hist/Retired/NEI
- ☐ Revision
- ☐ Superseded

Has the original commitment been implemented? ☒ YES ☐ NO, notify site Licensing

Original Commitment Description:
Procedure provides a method for cross-ref. Tech Manuals with the applicable plant procedure (s).

Revised Commitment Description:

Summary of Justification for Change (attach additional sheets if necessary):
Per EN-LI-110:
As a result of changes in circumstances a commitment may be rendered irrelevant or no longer useful. This may include commitments that have been tracked for more than two years and are well established in work processes or otherwise embedded in site processes. In these situations, commitment may be removed from continued tracking (see Step 3.0[13] for status changes that may be used to indicate continuing tracking is no longer required).

Reference Procedures: EN-DC-132, Control of Engineering Documents and EN-DC-148,. Vendor Manuals and the Vendor Re-Contact Process. (Vendor manuals are received electronically and processed by Administrative Services) Reg 10/26/10

Prepared By: Patricia H. Reed Patricia H. Reed Date: 10/25/2010
Print Name / Signature

Management Approval: Angela S. Mosby Angela S. Mosby Date: 10/25/2010
Print Name / Signature

Forward completed form to site Licensing.

Site Licensing Management (or designee) Concurrence: Rita R. Jackson for CLP Date: 10/26/10
Print Name / Signature

✓	QA RECORD
	RT. B14.39
	NON-QA RECORD
SW	INITIALS
NUMBER of PAGES	4
DATE	11-15-10
RELATED DOCUMENT NUMBER:	

PART I

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

☐ YES STOP. Use appropriate codified process (e.g., 10 CFR 50.71(e), 10 CFR 50.54, 10 CFR 50.59, 10 CFR 50.55) to evaluate commitment and then continue to Part II, if appropriate. IF NRC approval obtained, THEN revise commitment; no further evaluation is required and no further NRC notification is required.

☒ 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 specified safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its specified 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 Describe basis below

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

☐ YES ☐ NO Describe basis below

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

☐ YES ☐ NO Describe basis below

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. IF NRC approval obtained, THEN revise commitment; no further evaluation is required and no further NRC notification is required. IF all three questions are answered NO, THEN 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 (e.g., rule, regulation, order or license condition)?

☐ YES Go to Question 3.2.

☒ NO Continue with Part IV.

3.2 Is the proposed revised commitment necessary and justified?

☐ YES Briefly describe rationale (attach additional sheets as necessary). IF commitment not yet implemented, THEN notify NRC of revised commitment prior to the original date in which the commitment was to be completed/satisfied.

☐ NO STOP. Do not proceed with the revision OR apply for appropriate regulatory relief. IF NRC approval obtained, revise commitment; THEN no further evaluation is required and no further NRC notification is required.

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 Continue with Part V.

4.2 Has the original commitment been implemented?

☒ YES Go to Question 4.3.

☐ NO STOP. Do not proceed with revision OR notify NRC of desired change. IF NRC approval obtained, THEN revise commitment; no further evaluation is required and no further NRC notification is required.

4.3 Is the proposed revised commitment necessary and justified?

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

☐ NO STOP. Do not proceed with revision OR notify NRC of desired change. IF NRC approval obtained, THEN revise commitment; no further evaluation is required and no further NRC notification is required.

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

Doc. Number / ID	Description

GGNS Commitment Change Evaluation Number

CCE 2011-0001

LRS/CMS I.D. P-32889 Site Licensing Tracking Number: CCE 2011-0001

Source Document / Date: CNRO-97/00004, dated 3/17/97, is the Entergy 180 Day Response to NRC Generic Letter 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves"

Commitment: ☐ Cancellation ☒ Revision
☐ Historic/Hist/Retired/NEI ☐ Superseded

Has the original commitment been implemented? ☒ YES ☐ NO, stop and notify site Licensing

Original Commitment Description:

Commitment P-32889 is addressed in the original GL 96-05 response per CNRO-96/00016, dated 11/15/96, and restated verbatim in the 180-day response per CNRO-97/00004. This commitment tracks the implementation of general program guidance and methodology at each EOI site in site specific programs. The 180-day response provides a summary of the "EOI Safety-Related Motor Operated Valve Periodic Verification Program" and includes the following:

> "The following matrix provides the results of the combined risk ranking and setup ratio used to determine a maximum bounding test frequency. In addition, the maximum interval between static tests is not to exceed 10 years.

STATIC TEST FREQUENCY MATRIX

H (3)	3 Cycles	3 Cycles	2 Cycles	2 Cycles
M (2)	6 Cycles	4 Cycles	4 Cycles	3 Cycles
L (1)	6 Cycles	6 Cycles	6 Cycles	4 Cycles
RISK ↑ / RATIO →	H-H (1)	H (2)	M (3)	L (4)

✓	QA RECORD
	RT. 614.37
	NON-QA RECORD
SW	INITIALS
NUMBER OF PAGES	5
DATE	5-2-11
RELATED DOCUMENT NUMBER	

Revised Commitment Description:

Make a one-cycle one-time extension to the Static Test Frequency interval for 1E51F019 (RCIC Minimum Flow to Suppression Pool Isolation Valve), that will also correspond to a one-time 6 month extension to the maximum interval (10 years) between static tests. The valve was last tested on 6/13/2001 and to meet the commitment, the valve must be tested by 6/13/2011. The valve will be tested during RF18 (Feb - April 2012) so GGNS is requesting a 11 month extension beyond the 10 year commitment.

Summary of Justification for Change (attach additional sheets if necessary):

1E51F019 was scheduled for static diagnostic testing during CYCLE 18 during the March 2011 RCIC system outage. The RCIC system outage was in the process of being moved to July of 2011 but because of the events to the nuclear plants in Japan in March of 2001, the GGNS management team decided to move all on-line RCIC work to refuel outage 18 to reduce any risk of having RCIC inoperable. To statically test the 1E51F019 valve requires that the breaker for the valve be tagged in the OPEN position while the test equipment is being connected, thus making RCIC inoperable. To reduce plant risk, the site (GGNS) is requesting a deferral of the late date for performing the diagnostic test on valve 1E51F019 until 5/13/2012 to accommodate diagnostic testing the valve in Refuel 18. To approve this deferral requires a one-time one-cycle test-interval-extension for this valve to be made. This is acceptable because this valve has a high safety margin (CLOSE: 24.4%, OPEN: 215.3%) and there have not been any failures or

4-29-11 RAG

~~have not been any failures~~ or stroke time anomalies noted in the past 10 years based on a review of the results of the quarterly stroke time test performed per 06-OP-1E51-Q-0002. The valve has a PM task that monitors the condition of the assembly every 36 months and no abnormalities have been identified. Also no failures of this MOV has been observed over the past 10 years based on a search of PCRS.

Prepared By: Daniel G. Herrod / _____ Date: 4-21-11
Print Name / Signature

Management Approval: Linda Patterson / _____ Date: 4/29/11
Print Name / Signature

Forward completed form to site Licensing.

Site Licensing Management (or designee) Concurrence: Christina Perino / _____ Date: 4-29-11
Print Name / Signature
RAG 4/29/11

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Program Manual, Fire Hazards Analysis, 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, 10 CFR 50.59, 10 CFR 50.55) 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 specified safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its specified 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 Describe basis below

This commitment is to perform diagnostic testing on 1E51F019 to monitor the performance of the MOV on a frequency not to exceed 10 years. The 1E51F019 has a safety function to OPEN automatically to provide minimum flow pump protection and to CLOSE automatically when pump flow is greater than minimum flow to provide maximum flow to the reactor. The MOV also has a safety function to CLOSE automatically to provide containment isolation when not required to provide minimum flow protection. The 1E51F019 is normally CLOSED during RCIC standby operation.

The 1E51F019 MOV has a PM task that monitors the condition of the assembly every 36 months and no abnormalities have been identified over the past 10 years. Based on a search of PCRS, no failures of this MOV has been observed over the past 10 years. There have also not been any failures or stroke time anomalies noted in the past 10 years based on a review of the results of the quarterly stroke time test performed per 06-OP-1E51-Q-0002.

Therefore based on the discussion above, a one time extension of the committed testing frequency would not cause a significant increase in the probability or consequence 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 Describe basis below

Allowing the 1E51F019 valve to be diagnostically tested at a one time frequency of 10 years and 11 months will not create the possibility of a new or different kind of accident from any previously evaluated since the diagnostic testing, performed now on a 4 year frequency, primarily monitors for valve and actuator degradation and test the actuator capability to produce enough thrust to close the valve against design basis accident pressures. Past testing has shown little to no changes in the output torque or thrust of this MOV. Based on the previous static testing and quarterly stroke history, exceeding the 10 static test frequency should in no way jeopardize the functionality of this MOV.

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

☐ YES ☒ NO Describe basis below

This MOV has a safety margin to CLOSE of 24.4% and a safety margin to OPEN of 215.3% based on the last diagnostic static test performed in 2001. Since testing began in 1991, this MOV has had little to no change in output thrust and torque, therefore this one time revision to the commitment will not involve a significant reduction in the 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. IF NRC approval obtained, THEN revise commitment; no further evaluation is required and no further NRC notification is required. IF all three questions are answered NO, THEN 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 (e.g., rule, regulation, order or license condition)?

☐ YES Go to Question 3.2.

☒ NO Continue with Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ YES Briefly describe rationale (attach additional sheets as necessary). IF commitment not yet implemented, THEN notify NRC of revised commitment date prior to the original date in which the commitment was to be completed/satisfied.

☐ NO STOP. Do not proceed with the revision, OR apply for appropriate regulatory relief. IF NRC approval obtained, THEN revise commitment; no further evaluation is required and no further NRC notification is required.

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 Continue with 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 CCEF 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 CCEF summary report.

☐ NO Revise commitment. No NRC notification is required.

REFERENCES

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

Doc. Number / ID	Description
CNRO-97/00004, dated 3/17/97	Entergy 180 Day Response to NRC Generic Letter 96-05

GGNS Commitment Change Evaluation Number

CCE 2011-0002

LRS/CMS I.D. P-32889 Site Licensing Tracking Number: CCE 2011-0002

Source Document / Date: CNRO-97/00004, dated 3/17/97, is the Entergy 180 Day Response to NRC Generic Letter 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves"

Commitment: ☐ Cancellation ☒ Revision
☐ Historic/Hist/Retired/NEI ☐ Superseded

Has the original commitment been implemented? ☒ YES ☐ NO, stop and notify site Licensing

Original Commitment Description:

Commitment P-32889 is addressed in the original GL 96-05 response per CNRO-96/00016, dated 11/15/96, and restated verbatim in the 180-day response per CNRO-97/00004. This commitment tracks the implementation of general program guidance and methodology at each EOI site in site specific programs. The 180-day response provides a summary of the "EOI Safety-Related Motor Operated Valve Periodic Verification Program" and includes the following:

> "The following matrix provides the results of the combined risk ranking and setup ratio used to determine a maximum bounding test frequency. In addition, the maximum interval between static tests is not to exceed 10 years.

STATIC TEST FREQUENCY MATRIX

H (3)	3 Cycles	3 Cycles	2 Cycles	2 Cycles
M (2)	6 Cycles	4 Cycles	4 Cycles	3 Cycles
L (1)	6 Cycles	6 Cycles	6 Cycles	4 Cycles
RISK ↑ / RATIO →	H-H (1)	H (2)	M (3)	L (4)

✓ QA RECORD
RT- 614.37
NON-QA RECORD
SW INITIALS
NUMBER of PAGES 5
DATE 5-2-11
RELATED DOCUMENT NUMBER:

Revised Commitment Description:

Make a one-cycle one-time extension to the Static Test Frequency interval for 1E51F078 (RCIC Turbine Exhaust Inboard Vacuum Breaker Valve), that will also correspond to a one-time 4 month extension to the maximum interval (10 years) between static tests. The valve was last tested on 11/17/2001 and to meet the commitment, the valve must be tested by 11/17/2011. The valve will be tested during RF18 (Feb-April 2012) so GGNS is requesting a 6 month extension beyond the 10 year commitment.

Summary of Justification for Change (attach additional sheets if necessary):

1E51F078 was scheduled for static diagnostic testing during CYCLE 18 during the March 2011 RCIC system outage. The RCIC system outage was in the process of being moved to July of 2011 but because of the events to the nuclear plants in Japan in March of 2001, the GGNS management team decided to move all on-line RCIC work to refuel outage 18 to reduce any risk of having RCIC inoperable. To statically test the 1E51F078 valve requires that the breaker for the valve be tagged in the OPEN position while the test equipment is being connected, thus possibly making RCIC inoperable. To reduce plant risk, the site (GGNS) is requesting a deferral of the late date for performing the diagnostic test on valve 1E51F078 until 5/17/2012 to accommodate diagnostic testing the valve in Refuel 18. To approve this deferral requires a one-time one-cycle test-interval-extension for this valve to be made. This is acceptable because this valve has a high safety margin (CLOSE: 1370.2%) and there have not been any failures or

stroke time anomalies noted in the past 10 years based on a review of the results of the quarterly stroke time test performed per 06-OP-1E51-Q-0002. The valve has a PM task that monitors the condition of the assembly every 36 months and no abnormalities have been identified. Also no failures of this MOV has been observed over the past 10 years based on a search of PCRS.

Prepared By: Daniel G. Herrod / _____ Date: 4-21-11
Print Name / Signature

Management Approval: Linda Patterson / _____ Date: 4/21/11
Print Name / Signature

Forward completed form to site Licensing.

Site Licensing Management (or designee) Concurrence: Christina Perino / _____ Date: 4-27-11
Print Name / Signature
RAJ 4/29/11

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Program Manual, Fire Hazards Analysis, 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, 10 CFR 50.59, 10 CFR 50.55) 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 specified safety function or negatively impact the ability of plant personnel to ensure the SSC is capable of performing its specified 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 Describe basis below

This commitment is to perform diagnostic testing on 1E51F078 to monitor the performance of the MOV on a frequency not to exceed 10 years. The 1E51F078 has a safety function to CLOSE automatically on a combination of high drywell pressure and low reactor pressure to provide containment isolation when RCIC is not functional during a LOCA. The 1E51F078 is normally CLOSED during RCIC standby operation per system operating instructions 04-1-01-E51-1.

The 1E51F078 MOV has a PM task that monitors the condition of the assembly every 36 months and no abnormalities have been identified over the past 10 years. Based on a search of PCRS, no failures of this MOV has been observed over the past 10 years. There have also not been any failures or stroke time anomalies noted in the past 10 years based on a review of the results of the quarterly stroke time test performed per 06-OP-1E51-Q-0002.

Therefore based on the discussion above, a one time extension of the committed testing frequency would not cause a significant increase in the probability or consequence 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 Describe basis below

Allowing the 1E51F078 valve to be diagnostically tested at a one time frequency of 10 years and 6 months will not create the possibility of a new or different kind of accident from any previously evaluated since the diagnostic testing, performed now on a 4 year frequency, primarily monitors for valve and actuator degradation and test the actuator capability to produce enough thrust to close the valve against design basis accident pressures. Past testing has shown very minor changes in the output torque or thrust of this MOV. Based on the previous static testing, safety margin and quarterly stroke history, exceeding the 10 static test frequency should in no way jeopardize the functionality of this MOV.

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

☐ YES ☒ NO Describe basis below

This MOV has a safety margin to CLOSE of 1370.2% based on the last diagnostic static test performed in 2001. Since testing began in 1991, this MOV has had very minor changes in output thrust and torque, therefore this one time revision to the commitment will not involve a significant reduction in the 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. IF NRC approval obtained, THEN revise commitment; no further evaluation is required and no further NRC notification is required. IF all three questions are answered NO, THEN 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 (e.g., rule, regulation, order or license condition)?

☐ YES Go to Question 3.2.

☒ NO Continue with Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ YES Briefly describe rationale (attach additional sheets as necessary). IF commitment not yet implemented, THEN notify NRC of revised commitment date prior to the original date in which the commitment was to be completed/satisfied.

☐ NO STOP. Do not proceed with the revision, OR apply for appropriate regulatory relief. IF NRC approval obtained, THEN revise commitment; no further evaluation is required and no further NRC notification is required.

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 Continue with 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 CCEF 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 CCEF summary report.

☐ NO Revise commitment. No NRC notification is required.

REFERENCES

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

Doc. Number / ID	Description
CNRO-97/00004, dated 3/17/97	Entergy 180 Day Response to NRC Generic Letter 96-05