Entergy Operations, Inc.
1340 Echelon Parkway
Jackson, MS 39213
Tel 601-368-5138

Ron Gaston
Director, Nuclear Licensing

10 CFR 50.90

GNRO-2020/00015

April 7, 2020

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Response to Request for Additional Information - License Amendment Request for One-Cycle Extension of Appendix J Type A Integrated Leakage Rate Test and Drywell Bypass Leakage Rate Test

Grand Gulf Nuclear Station, Unit 1
NRC Docket No. 50-416
Renewed Facility Operating License No. NPF-29

References:

1) Entergy Operations, Inc. (Entergy) letter to U. S. Nuclear Regulatory Commission (NRC), "License Amendment Request for One-Cycle Extension of Appendix J Type A Integrated Leakage Rate Test and Drywell Bypass Leakage Rate Test," (ADAMS Accession No. ML20091M363), dated March 31, 2020

2) NRC Electronic mail from S. P. Lingam (NRC) to J. Shaw (Entergy), Subject: "Grand Gulf - RAI for Exigent LAR Associated with One-Time Extension of Appendix J Type A Integrated Leakage Rate Test Frequencies from 11.5 Years to 13.5 Years (EPID L-2020-LLA-0060)," dated April 6, 2020

In Reference 1, Entergy Operations, Inc. (Entergy) submitted a request for a proposed amendment to Renewed Facility Operating License (FOL) NPF-29, Appendix A, "Technical Specifications" (TS) for Grand Gulf Nuclear Station, Unit 1 (GGNS). The proposed change would allow for a one-cycle extension of the interval to perform the GGNS Type A integrated leakage rate test (ILRT) and drywell bypass leakage rate test (DWBT) from 11.5 years to 13.5 years.

In Reference 2, the U. S. Nuclear Regulatory Commission (NRC) transmitted a request for additional information (RAI) concerning the proposed license amendment. The Enclosure to this letter provides a response to the NRC RAI.

In addition, Attachments 1 and 2 to the Enclosure provide updated versions of the existing TS pages marked up to show the proposed changes and revised (clean) TS pages.

This letter contains no new regulatory commitments.

Should you have any questions or require additional information, please contact Ron Gaston, Director, Nuclear Licensing at 601-368-5138.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), a copy of this application, with attachments, is being provided to the designated State Officials.

I declare under penalty of perjury, the foregoing is true and correct. Executed on April 7, 2020.

Respectfully,

Ron Gaston

RWG/jls

Enclosure: Response to Request for Additional Information

Attachments to Enclosure:
1. Markup of Technical Specification Pages
2. Retyped Technical Specification Pages

cc: NRC Region IV Regional Administrator
NRC Senior Resident Inspector – Grand Gulf Nuclear Station, Unit 1
State Health Officer, Mississippi Department of Health
NRC Project Manager - Grand Gulf Nuclear Station, Unit 1

Enclosure

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Response to Request for Additional Information

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

By letter dated March 31, 2020 (Reference 1), Entergy Operations, Inc. (Entergy), requested NRC approval of a proposed amendment to Renewed Facility Operating License (FOL) NPF-29, Appendix A, "Technical Specifications" (TS) for Grand Gulf Nuclear Station, Unit 1 (GGNS). The proposed change would allow for a one-cycle extension of the interval to perform the GGNS Type A integrated leakage rate test (ILRT) and drywell bypass leakage rate test (DWBT) from 11.5 years to 13.5 years.

By email dated April 6, 2020 (Reference 2), the NRC informed Entergy that additional information is needed to support the Staff's continued review of the application. The specific question presented in the Reference 2 request for additional information (RAI) is repeated below for ease of review. Entergy's response is provided thereafter.

RAI-1:

Based on the testing of Type B and Type C component testing performed to date in the current outage (RF22), provide a brief synopsis of the testing results. The information should include the performance testing of the above components that did not show acceptable performance in the previous two outages. In addition, include information on any other components that exceeded the administrative acceptance criteria limits in the current outage and the basis for acceptance of the test results (i.e., actions taken for correcting the leakage or acceptance in as is condition).

Entergy Response

As of April 6, 2020, GGNS has completed 92% of the as-found (AF) Type B and Type C Local Leakage Rate Tests (LLRTs) and 29% of the as-left (AL) LLRTs for Refueling Outage (RF) 22. Type B and C LLRT results indicate <33% of the $0.6L_a$ margin for Maximum Pathway Leakage and <15% of the $0.6L_a$ margin for Minimum Pathway Leakage. There have been four AF LLRT components which exceeded their administrative leakage limits during RF22, seven during RF21, and seven during RF20. Additional details are provided in Tables 1, 2, and 3, below.

Two of the four RF22 components which exceeded their local administrative limits (i.e., P53F001 and P45F062) were corrected by completing actuator rebuilds. The D23F593 and E12F044A AF leakage values were accepted, and no further work is planned for RF22. The leakage values were acceptable, as neither of the exceeded administrative limits challenged the overall margin to $0.6L_a$, and their respective penetrations were not inoperable.

E12F044A is the only component which exceeded its administrative limit in both RF21 and RF22. E12F044A exceeded its administrative limit in RF21 with an AF leakage of 1275 sccm. A valve internal inspection was completed during RF21 (Reference Table 2); However, the 1280 sccm AL leakage remained above the administrative limit of 1040 sccm. E12F044A exceeded its administrative limit in RF22 with an AF leakage of 1950 sccm. The resultant increase of 670 sccm from the RF21 AL value to the RF22 AF value represents 0.34% of the overall margin to $0.6L_a$. This did not challenge the overall margin to $0.6L_a$. Additional information is provided in Table 1.

G36F106, B21F016, and E12F028B exceeded their administrative leakage limits in both RF20 and RF21. These valves were repaired or replaced, as indicated in Table 2 below, and did not exceed their administrative leakage limit in RF22.

Table 1: RF22 Type B and C Components Exceeding Administrative Limits

Valve/Penetration	Leakage (sccm)	Administrative Leakage (sccm)	Condition Report Number	Resolution
P53F001 Instrument Air Outboard Containment Isolation Valve (CIV)	AF: 725 sccm AL: 60 sccm	650 sccm	CR-GGN-2020-2771	Work Order (WO) 52830932 for actuator rebuild preventative maintenance; Leakage issues resolved after rebuild.
P45F062 Drywell & Containment Floor Drain Sump Pump Discharge Outboard CIV	AF: 1800 sccm AL: 0 sccm	1580 sccm	CR-GGN-2020-3195	Actuator rebuild under the same WO (52830471-01A/B/C). Leakage issues resolved after rebuild.
D23F593 Drywell Fission Product Monitor Return Inboard CIV	AF: 1890 sccm	195 sccm	CR-GGN-2020-3313	No other work planned in RF22; WO 541006 moved to RF23. Leakage accepted. Did not challenge overall margin to $0.6L_a$. Outboard valve, D23F594, tested satisfactory in RF22 with a leakage of 20 sccm.

continued

Table 1: RF22 Type B and C Components Exceeding Administrative Limits (cont.)

Valve/Penetration	Leakage (sccm)	Administrative Leakage (sccm)	Condition Report Number	Resolution
E12F044A Residual Heat Removal (RHR) "A" to Low Pressure Coolant Injection (LPCI) Inboard CIV	AF: 1950 sccm	1040 sccm	CR-GGN-2020-4379 CR-GGN-2020-4631	Exceeded administrative limit in RF21 and RF22. No other work planned in RF22; Replacement valve purchased from manufacturer and expected to be delivered summer 2020 for RF23. An alternate option is based on discussion with Operations Engineering. This line is not used at the site; therefore, a request has been submitted for an modification in RF23 to cut and cap the line where E12F044A and its sister valve E12F044B are installed. For RF22, leakage was accepted as it does not challenge overall margin to $0.6L_a$. E12F044A is an inboard isolation valve in Penetration 20. Other inboard valves in this penetration include E12F042A, E12F028A, E12F025A, and E12F037A as well as E12F027A serving as the outboard valve. All the other components in this penetration tested below their administrative limit.

Table 2: RF21 Type B and C Components Exceeding Administrative Limits

Valve/Penetration	Leakage (sccm)	Administrative Leakage Limit (sccm)	Condition Report Number	Resolution
E12F028B RHR "B" to LPCI Inboard CIV	AF: 13800 sccm AL: 40 sccm	4680 sccm	CR-GGN-2018-3724	Engineering Change (EC) 78125 was implemented to accept valve testing in reverse direction, as valve is a containment isolation valve upstream of the containment spray header ring and testing is in direction that would assist in unseating valve.
G36F106 Reactor Water Cleanup (RWCU) Backwash Transfer Pump Discharge Inboard CIV	AF: 71000 sccm AL: 0 sccm	1040 sccm	CR-GGN-2018-3863	Valve seat replacement.
E51F064 Steam Supply to Reactor Core Isolation Coolant (RCIC) Turbine Outboard CIV	AF: 3380 sccm AL: 3570 sccm	2600 sccm	CR-GGN-2018-4166 CR-GGN-2018-5845	Actuator rebuild performed. Torque switch adjustments left leakage at 3570 sccm; In-body inspection performed to blue check and cleaned seat. Retested in RF22; as-found LLRT was satisfactory.
P53F002 Instrument Air Inboard CIV	AF: 800 sccm AL: 450 sccm	650 sccm	CR-GGN-2018-4236 CR-GGN-2018-6352	Valve replacement under WO-500229.
B21F016 Main Steam Line (MSL) Drain, (Main Steam Isolation Valve (MSIV)) Inboard CIV	AF: 18600 sccm AL: 70 sccm	780 sccm	CR-GGN-2018-4448	Valve replacement under WO 376662.
E30P120 Suppression Pool Level Instrument Tubing Outboard	AF: 145 sccm AL: 145 sccm	100 sccm	CR-GGN-2018-5190	Appendix J has no requirements beyond commitment to count the leakage against the running total. No operability concerns to address. Leakage Reduction Program has confirmed leakage within program limits.
E12F044A Residual Heat Removal (RHR) "A" to Low Pressure Coolant Injection (LPCI) Inboard CIV	AF: 1275 sccm AL: 1280 sccm	1040 sccm	CR-GGN-2018-6191	Issue with troubleshooting handwheel as threads were not engaged; Performed a valve inbody inspection of E12F044A, including blue check of the valve seat.

Table 3: RF20 Type B and C Components Exceeding Administrative Limits

Valve/Penetration	Leakage (sccm)	Administrative Leakage Limit (sccm)	Condition Report Number	Resolution
B21F028A MSIV for MSL "A" Outboard CIV	AF: 81000 sccm AL: 33400 sccm	47200 sccm	CR-GGN-2016-1398 CR-GGN-2016-1456	Repaired valve under WO 52568389.
B21F016 MSL Drain, MSIV Inboard CIV	AF: 9860 sccm AL: 9860 sccm	780 sccm	CR-GGN-2016-1577 CR-GGN-2016-1586	Accepted AF data as AL and replaced in RF21.
E12F028B RHR "B" to LPCI Inboard CIV	AF: 8600 sccm AL: 1400 sccm	4680 sccm	CR-GGN-2016-1587	WO 52564082 replaced wearable parts such as seal ring gasket and replaced the valve internals.
B33F125 Reactor Recirculation Post Accident Sampling Outboard CIV	AF: >20000 sccm AL: 2 sccm	260 sccm	CR-GGN-2016-1706	CR notes recommendation to adjust the torque/limit setting on B33F125 to reduce leakage. Repaired under WO 439571.
G36F106 Reactor Water Cleanup (RWCU) Backwash Transfer Pump Discharge Inboard CIV	AF: 37000 sccm AL: 28600 sccm	1040 sccm	CR-GGN-2016-1860	Recommended in CR to troubleshoot/rework valve; Repaired in RF21.
E22F004 High Pressure Core Spray (HPCS) Pump Discharge to Reactor Pressure Vessel (RPV)	AF: 40300 sccm AL: 2650 sccm	3120 sccm	CR-GGN-2016-2380	Exceeded leakage was due to boundary valve leakage. The valve was retested and passed its AL LLRT.

REFERENCES

1. Entergy Operations, Inc. (Entergy) letter to U. S. Nuclear Regulatory Commission (NRC), "License Amendment Request for One-Cycle Extension of Appendix J Type A Integrated Leakage Rate Test and Drywell Bypass Leakage Rate Test," (ADAMS Accession No. ML20091M363), dated March 31, 2020

2. NRC Electronic mail from S. P. Lingam (NRC) to J. Shaw (Entergy), Subject: "Grand Gulf - RAI for Exigent LAR Associated with One-Time Extension of Appendix J Type A Integrated Leakage Rate Test Frequencies from 11.5 Years to 13.5 Years (EPID L-2020-LLA-0060)," dated April 6, 2020Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," September 1995.

Attachment 1 to Enclosure

GNRO-2020/00015

Markup of Technical Specification Pages

<u>TS Page</u>
3.6-53
5.0-16

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1.1	Verify bypass leakage is less than or equal to the bypass leakage limit. However, during the first unit startup following drywell bypass leak rate testing performed in accordance with this SR, the acceptance criterion is leakage ≤ 10% of the bypass leakage limit.	24 months following two consecutive tests with bypass leakage greater than the bypass leakage limit until two consecutive tests are less than or equal to the bypass leakage limit <u>AND</u> 48 months following a test with bypass leakage greater than the bypass leakage limit <u>AND</u> ----------NOTE-------- SR 3.0.2 is not applicable for extensions > 12 months. ------------------------ In accordance with the Surveillance Frequency Control Program except next drywell bypass leak rate test performed after the October 19, 2008 test shall be performed no later than plant restart after the End of Cycle ~~22~~ 23 Refueling Outage

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5.5 Programs and Manuals (continued)

5.5.11 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the license; or

   2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that do not meet the criteria of either Specification 5.5.11.b.1 or Specification 5.5.11.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.12 10 CFR 50, Appendix J, Testing Program

23 This program establishes the leakage rate testing program of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be implemented in accordance with the Safety Evaluation issued by the Office of Nuclear Reactor Regulation dated April 26, 1995 (GNRI-95/00087) as modified by the Safety Evaluation issued for Amendment No. 135 to the Operating License, except that the next Type A test performed after the October 19, 2008 Type A test shall be performed no later than the plant restart after the End of Cycle 22 Refueling Outage. For Type B and Type C local leakage rate testing, this program shall be in accordance with the guidelines contained in NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012. Consistent with standard scheduling practices for Technical Specifications required surveillances, intervals for the recommended surveillance frequency for Type A testing may be extended by up to 25 percent of the test interval, not to exceed 15 months. The calculated peak containment internal pressure for the design basis loss of coolant accident, Pa, is 12.1 psig.

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Attachment 2 to Enclosure

GNRO-2020/00015

Retyped Technical Specification Pages

<u>TS Page</u>
3.6-53
5.0-16

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1.1	Verify bypass leakage is less than or equal to the bypass leakage limit. However, during the first unit startup following drywell bypass leak rate testing performed in accordance with this SR, the acceptance criterion is leakage $\leq$ 10% of the bypass leakage limit.	24 months following two consecutive tests with bypass leakage greater than the bypass leakage limit until two consecutive tests are less than or equal to the bypass leakage limit <u>AND</u> 48 months following a test with bypass leakage greater than the bypass leakage limit <u>AND</u> ----------NOTE-------- SR 3.0.2 is not applicable for extensions > 12 months. ------------------------ In accordance with the Surveillance Frequency Control Program except next drywell bypass leak rate test performed after the October 19, 2008 test shall be performed no later than plant restart after the End of Cycle 23 Refueling Outage

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5.5 Programs and Manuals (continued)

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This program establishes the leakage rate testing program of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be implemented in accordance with the Safety Evaluation issued by the Office of Nuclear Reactor Regulation dated April 26, 1995 (GNRI-95/00087) as modified by the Safety Evaluation issued for Amendment No. 135 to the Operating License, except that the next Type A test performed after the October 19, 2008 Type A test shall be performed no later than the plant restart after the End of Cycle 23 Refueling Outage. For Type B and Type C local leakage rate testing, this program shall be in accordance with the guidelines contained in NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012. Consistent with standard scheduling practices for Technical Specifications required surveillances, intervals for the recommended surveillance frequency for Type A testing may be extended by up to 25 percent of the test interval, not to exceed 15 months. The calculated peak containment internal pressure for the design basis loss of coolant accident, Pa, is 12.1 psig.

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