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GNRO-2011/00070

August 25, 2011

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

SUBJECT: Request for Additional Information Regarding
Extended Power Uprate
Grand Gulf Nuclear Station, Unit 1
Docket No. 50-416
License No. NPF-29

REFERENCES:
1. Email from A. Wang to F. Burford dated August 3, 2011, GGNS EPU Request for Additional Information Related to Mechanical and Civil Engineering Branch Review Excluding the Steam Dryer (ME4679) (NRC ADAMS Accession Number ML112160007)
2. License Amendment Request, Extended Power Uprate, dated September 8, 2010 (GNRO-2010/00056, NRC ADAMS Accession Number ML102660403)

Dear Sir or Madam:

The Nuclear Regulatory Commission (NRC) requested additional information (Reference 1) regarding certain aspects of the Grand Gulf Nuclear Station, Unit 1 (GGNS) Extended Power Uprate (EPU) License Amendment Request (LAR) (Reference 2). Attachment 1 provides responses to the additional information requested by the Mechanical and Civil Engineering Branch.

No change is needed to the no significant hazards consideration included in the initial LAR (Reference 2) as a result of the additional information provided. There are no new commitments included in this letter.

If you have any questions or require additional information, please contact Jerry Burford at 601-368-5755.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 25, 2011.

Sincerely,

M. A Krupa

MAK/FGB/dm

Attachments:

1. Response to Request for Additional Information, Mechanical and Civil Engineering Branch

cc: Mr. Elmo E. Collins, Jr.
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NRC Senior Resident Inspector
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Port Gibson, MS 39150

Attachment 1

GNRO-2011/00070

Grand Gulf Nuclear Station Extended Power Uprate

Response to Request for Additional Information

Mechanical and Civil Engineering Branch

Response to Request for Additional Information
Mechanical and Civil Engineering Branch

By letter dated September 8, 2010, Entergy Operations, Inc. (Entergy) submitted a license amendment request (LAR) for an Extended Power Uprate (EPU) for Grand Gulf Nuclear Station, Unit 1 (GGNS). By letters dated February 23, 2011 (NRC ADAMS Accession No. ML110540545) and June 15, 2011 (NRC ADAMS Accession No. ML111670059), Entergy submitted responses to requests for additional information (RAI) from the Mechanical and Civil Engineering Branch. Subsequently, the Mechanical and Civil Engineering Branch has determined that the following additional information is needed for the U.S. Nuclear Regulatory Commission (NRC) staff to complete their review of the amendment. Entergy's response to each item is also provided below.

RAI # 1

In its June 15, 2011, response to RAI 2 b), Entergy Operations, Inc. (Entergy or the licensee) stated that the 13.1 percent feedwater (FW) flow rate increase accompanying the proposed extended power uprate (EPU) implementation at Grand Gulf Nuclear Station (GGNS) does not affect the stresses in the FW piping system. The licensee stated that the design basis of the FW piping system does not include loads which are governed by flow rate, including water hammer and transient loads. As such, the FW pipe stress analysis of record (AOR) does not credit pipe stresses resulting from flow-induced transients. While the design basis of the FW piping system does not include stresses resulting from water hammer and other flow-induced transients, the NRC staff requests that the licensee provide a technical justification which demonstrates that the FW piping and supports will maintain their structural integrity following transients whose load consequences would be amplified by the proposed flow increase accompanying EPU implementation at GGNS. Transients which should be considered include, but are not limited to, a FW pump trip, FW regulating valve closure and FW isolation valve closure. This technical justification should quantitatively demonstrate that the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code allowable values used in the original design of the FW piping system will continue to be satisfied following these events, upon implementation of the proposed EPU at GGNS.

Response

Background

The GGNS Feedwater (FW) piping system and components are comprised of large bore piping, valves, heaters, pumps, and various sizes of branch piping located in the turbine, auxiliary, and containment buildings. The FW system has the capability to automatically provide the required flow rate of water to the reactor pressure vessel (RPV) during startup, shutdown, at the power levels up to and including rated load and during the plant design transients, without interruption of operation or damage to equipment. The FW piping system design basis did not consider flow induced transients as a credible event for the current design nor for the increased flow velocity associated with EPU operation.

System transients such as pump trips, relief valve operation, or other valve manipulations may be included in the design set of loading combinations if these operational events result in significant loading conditions. For GGNS, the determination of the design loadings and combinations for the feedwater system did not include any system transients, such as pump trip

or rapid valve closure, due to the system operating parameters and equipment design features. However, the GGNS feedwater piping design stresses did include various safety relief valve (SRV) lift scenarios in the normal and upset load cases that envelope other types of transient events.

Evaluation of Feedwater Hydraulic Transients

A comprehensive transient hydraulic evaluation of the GGNS condensate and feedwater systems was developed for the current plant design (CLTP conditions). The steady state and transient behavior within the piping system was simulated using numerical methods incorporated in the SPS® (Stoner Pipeline Simulator) computer package. The numerical methods simultaneously solved the differential equations which govern fluid flow in spatial and time domains. The numerical model simulated the fluid properties, piping, pumps, valves, controls, and the physical elevation profile of the system. Pressure drop due to friction within the piping was computed by means of the Darcy-Weisbach Equation. The computer simulations applied the appropriate differential equations and boundary conditions pertaining to fluid flow in each device and at boundary conditions to determine the pressure and flow at each point in the system at every time step of the simulation.

This evaluation considered various system operating scenarios and conditions and demonstrated that normal operating transients do not challenge the piping design pressure limits. The worst case scenario (in terms of system pressure increases) for the feedwater and condensate piping (including the safety related portion) resulted from the sudden trip of both reactor feedwater pumps. The GGNS feedwater pumps are driven by steam turbines. During pump trips, the FW flow decreases at a rate based on coast down inertia of the turbine and pump impeller. The results from the transient hydraulic evaluation were used to determine the transient piping loads from the limiting scenario applicable to each section of piping. The pressure profiles, extrapolated to EPU flow rates, along with the acoustic velocity of the fluid and piping geometry, were then combined to determine the forcing functions for calculating the transient loading on the piping system at EPU conditions. These loads were then compared to the normal/upset loads used in the feedwater stress analysis to assess whether the loads were significant enough to include in the piping design.

For the FW piping (including the safety related portion), the most limiting transient loading due to the pressure wave acting on the piping system was a maximum of 420 lbf (see the attached Figure “RFP Case 1: Force at FW Header 1”). The peak force resulted in a “g” loading on the piping system of 0.016964 g. The FW piping design included loads from thermal, dead weight, seismic and annulus pressurization anchor movements, safe shutdown earthquake (SSE), and SRV actuations. The transient hydraulic loads considered in the piping design from SRV actuations applied a “g” loading of approximately 2 g’s. When the load from the limiting system operation hydraulic transient (trip of two FW pumps) is combined with the SRV load using the standard SRSS (square root of the sum of squares) method, the resulting load would increase from 2.0 g’s to 2.000072 g’s. The increase in the total piping design load from the limiting hydraulic transient is thus negligible.

For the condensate piping (all of which is non-safety related), the maximum load on the piping system due to a trip of both FW pumps was determined to be 2,212.1 lbf (see the attached Figure “Case 1: Condensate Piping Segment”). While this force is greater than that for the FW piping, the resulting loading is only 0.0747 g due to the larger mass of the condensate piping

segment. Therefore, the equivalent “g” loading is of a similar magnitude as the feedwater piping and is likewise negligible when compared to the piping design loads.

System Design Features

The design of the GGNS FW system precludes water hammer transients. There are no in-line component or flow conditions in the system that would cause a sudden interruption of flow. The only system valves, other than check valves, are FW system isolation valves, FW pump suction isolation, and FW pump discharge isolation valves. The isolation times associated with these valves vary from 32 to 100 seconds which are stroke times that are much longer than is required to generate a substantial transient pipe load. Feedwater flow is automatically controlled by the reactor feed pump speed set by the Feedwater controller. The normal feed pump operating configuration uses two adjustable speed turbine-driven feed pumps, which do not use control valves to regulate FW flow during normal operation. The discharge flow from the Reactor Feed Pumps is controlled by adjustable speed turbine drives. Use of adjustable speed turbines eliminates the types of flow transients associated with rapid opening or closing of high differential pressure, in-line FW flow control valves. Additionally, there has been no relevant experience at GGNS that indicates transients of this nature have taken place. As such, valve open/closure transients were not included in the GGNS feedwater design loading bases since the system design precludes valve operation from creating unacceptable pressure surges.

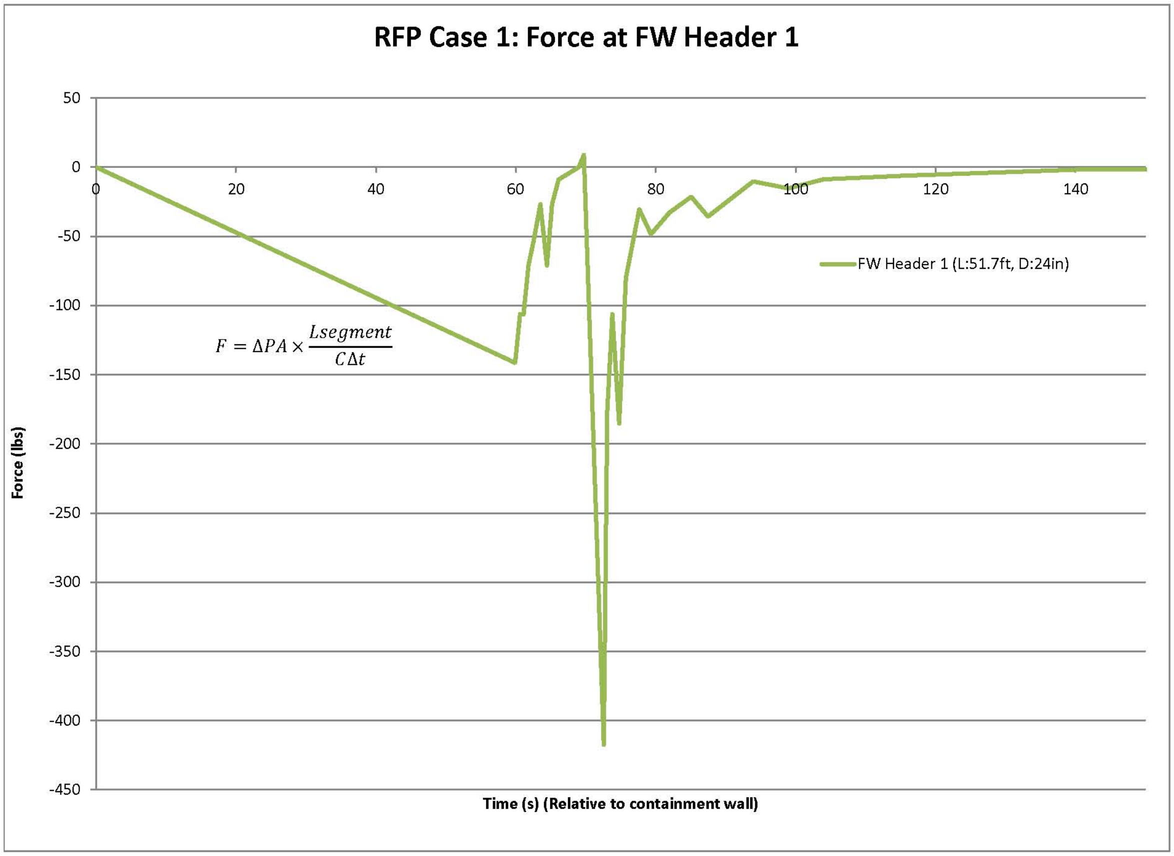
For the GGNS FW piping design (*i.e.*, pipe lengths, sizes, and configurations), the relevant time for flow stopping or starting must be smaller than one tenth of a second (0.1 seconds) in order for any significant loading to result. That is, for equipment related transient initiators such as pump trip or valve closure, the time for the action to occur should be at most on the order of magnitude of 0.1 second for that event to be considered significant. Certain physical transients, such as a cavity collapse, can be considered to occur almost instantaneously, so they are considered significant. The system geometry and operating conditions determine whether these types of transients are reasonable to include in the design loading conditions.

Cavity generation and collapse can occur in fluid piping systems when pumps trip and then restart. This phenomenon typically occurs in systems where the piping has large changes in elevation and where the system operates at relatively low pressures once the pump is tripped. The GGNS feedwater system normally remains pressurized to at least reactor system pressure following a pump trip. This pressure is well above the vapor pressure of the fluid contained in the piping system so no voiding is expected to occur. The piping does have a drop in elevation of approximately 20 feet near the reactor vessel. However, since this piping is directly connected to the vessel and remains pressurized, there is no reasonable scenario that would generate cavity formation and collapse during system operation. Additionally, there has been no relevant experience at GGNS that indicates transients of this nature have taken place. Thus, cavity collapse transients were not included in the feedwater design loading bases.

Conclusions

An evaluation of hydraulic transients determined that the piping system loads resulting from the worst case normal operational event (sudden trip of both FW pumps) were negligible and well bounded by the loads associated with SRV actuations and other parameters included in the piping design (*e.g.*, loads from thermal, dead weight, seismic and annulus pressurization anchor movements, SSE, etc.). In addition, the design of the system precludes the likelihood of severe

hydraulic transients. The more than 25 years of GGNS operating history of the feedwater system, including events such as pump trips, has confirmed that transients have not resulted in damaging forces to the piping system. Thus, loads due to flow-induced transients are negligible and need not be included in the feedwater system piping design.



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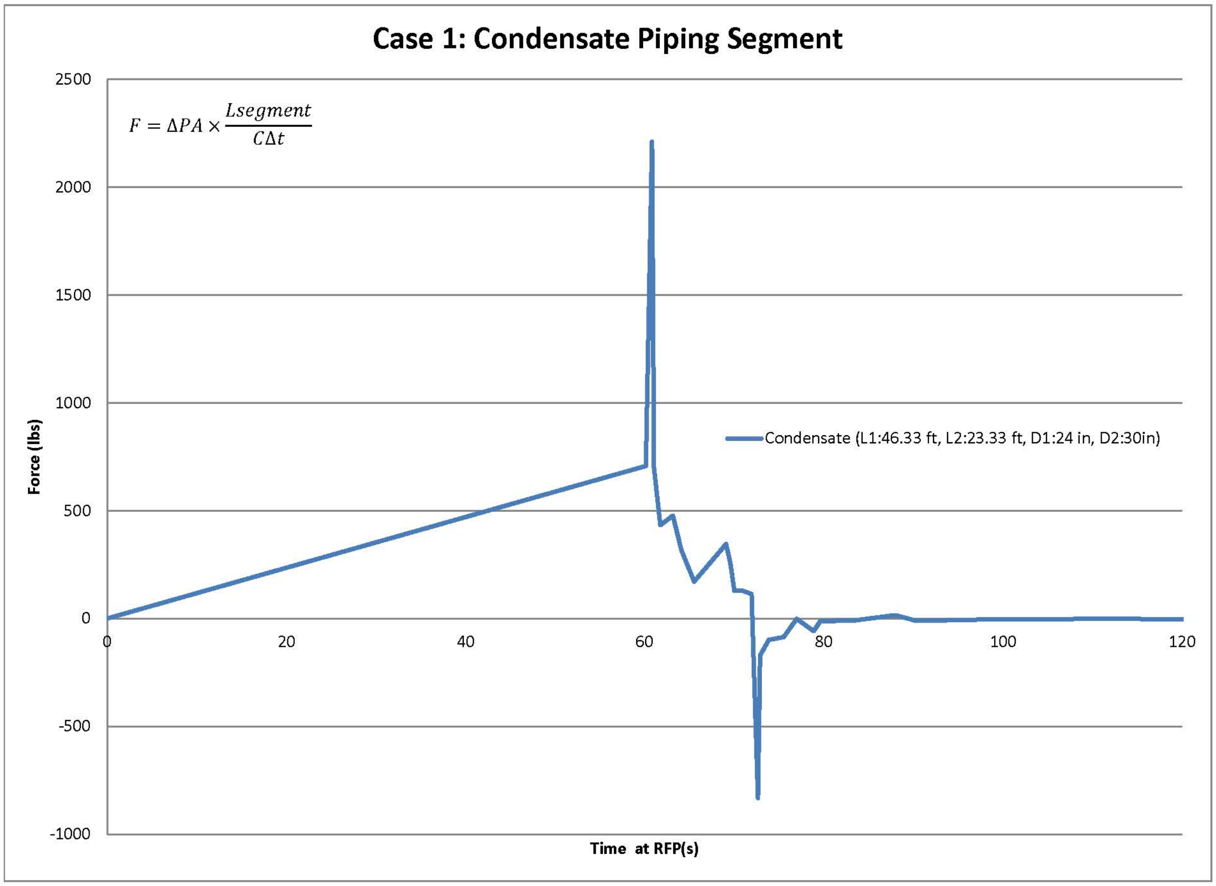

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RAI # 2

Note 11 of Table 2.2-7 of the Power Uprate Safety Analysis Report (PUSAR), submitted as part of the GGNS EPU license amendment request (LAR) on September 8, 2010, indicates that the low alloy steel forging portion of the FW nozzle at GGNS can be qualified for acceptable fatigue behavior using the methods of NUREG-0619, “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking,” or Revision 1 of GE-NE-523-A71-0594-A, “Alternate BWR Feedwater Nozzle Inspection Requirements.” Please state the specific inspection provisions of either reference which will be used to qualify the component for fatigue usage following EPU implementation. Additionally, please provide additional information regarding the proposed implementation schedule for these inspection provisions.

Response

NUREG-0619 was issued by the NRC in 1980 describing a cracking phenomenon of reactor pressure vessel feedwater nozzle and control rod drive nozzle inner radii. Examinations were performed at GGNS during its first 10-year In-Service Inspection (ISI) interval in strict

compliance with the recommended guidelines of the NUREG. However, as a result of enhanced technology and more sophisticated techniques for fatigue analysis, examination requirements for the feedwater nozzle blend radius were altered during the second interval based on BWR Owners' Group (BWROG) Licensing Topical Report GE-NE-523-A71-0594. This alternative approach continues to be implemented during the current third 10-year interval.

As noted in the NUREG, the objective of this inspection program is to ensure that even if feedwater nozzle thermal fatigue cracks are initiated, their growth will be limited to avoid violation of the applicable ASME Code or a threat to the integrity of the reactor vessel. This inspection program, as implemented by licensees, will assure continued reactor safety while improved nondestructive examination methods are being developed.

The inspection consists of an augmented examination program that includes all six feedwater nozzle inner radii. The subject nozzle inner radii are ultrasonically examined. Procedures, personnel and acceptance criteria used in the performance of the nozzle examinations satisfy the requirements of ASME Section XI.

This augmented inspection has been implemented at GGNS. The nozzles were last inspected during RF15 (Spring 2007) and are to be next inspected during RF19 (Spring 2014). That will be the first refueling outage following the implementation of the EPU. The inspection interval is consistent with recommendations in the BWROG Topical Report. The examination requirements augment those required by ASME Section XI for Examination Category B-D. Flaws detected during examination would be evaluated by comparing the examination results to the acceptance standards established in ASME Section XI, IWB-3512.

While the inspection process would ensure that any flaws are detected and evaluated to ensure Code requirements are met, the fact the calculated cumulative usage factor (CUF) is greater than 1.0 indicates the nozzle is not acceptable for use for the full original 40-year life of the plant. The CUF due to rapid cycling contributes to the total CUF exceeding 1.0. Entergy has now completed a re-evaluation of the fatigue usage of these nozzles based on observed corrosion rates of the nozzle; this evaluation has resulted in the total CUF for the nozzle of 0.5802. On this basis the nozzles are qualified for the entire 40-year plant life.