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GNRO-2011/00052

June 23, 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 June 22, 2011, GG EPU Grand Gulf Extended Power Uprate Containment and Ventilation Branch Request for Information (ME4679)
2. License Amendment Request, Extended Power Uprate, dated September 8, 2010 (GNRO-2010/00056, NRC ADAMS Accession No. 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 Containment and Ventilation 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 June 23, 2011.

Sincerely,

MAK/FGB/dm

Attachments:

1. Response to Request for Additional Information, Containment and Ventilation Branch

cc: Mr. Elmo E. Collins, Jr.
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U. S. Nuclear Regulatory Commission
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NRC Senior Resident Inspector
Grand Gulf Nuclear Station
Port Gibson, MS 39150

Attachment 1

GNRO-2011/00052

Grand Gulf Nuclear Station Extended Power Uprate

Response to Request for Additional Information

Containment and Ventilation Branch

Response to Request for Additional Information
Containment and Ventilation Branch

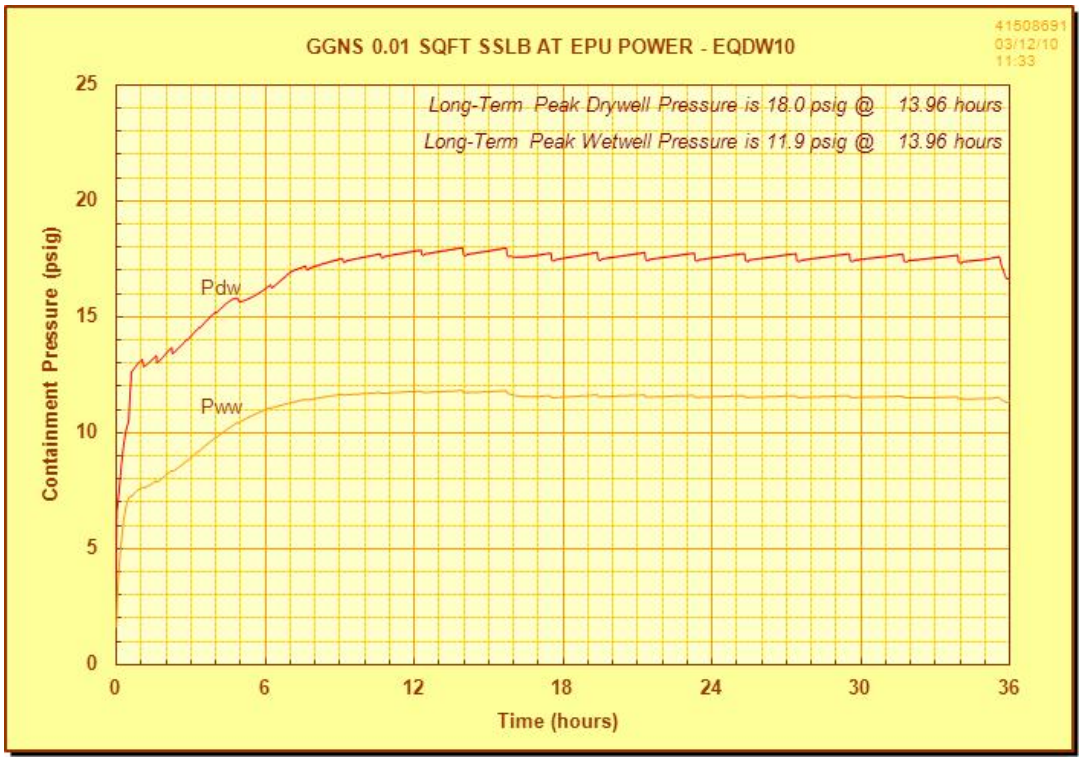
By letter dated September 8, 2010, Entergy Operations, Inc. (Entergy) submitted a license amendment request (LAR) for an Extended Power Uprate (EPU) for Grand Gulf Nuclear Station, Unit 1 (GGNS). By letters dated March 31, 2011 (NRC ADAMS Accession No. ML110900586) and June 8, 2011 (NRC ADAMS Accession No. ML111590836), Entergy provided responses to the questions from the Containment and Ventilation Branch. Subsequently, the U.S. Nuclear Regulatory Commission (NRC) staff has determined that the following additional information requested by the Containment and Ventilation Branch is needed for the NRC staff to complete their review of the amendment. Entergy's response to each item is provided below.

RAI # 1

Refer to Entergy's response to RAI # 25 in letter dated March 31, 2011 (ML110900586 and ML110900593). Provide a pressure and temperature profile for the containment analysis that resulted in the limiting peak pressure of 11.9 psig given in PUSAR Table 2.6-1 that Entergy proposes to use as **'P_a'** for the 10 CFR 50 Appendix J integrated leak rate testing (ILRT). Provide a detailed discussion explaining why the short term peak pressure of 14.8 psig (given in PUSAR Table 2.6-1) should not be used as **'P_a'.** In the discussion, include similar precedent Mark III containment examples that have been approved by NRC. Please note that in the current licensing basis the long term peak pressure of 11.5 psig (PUSAR Table 6.2-1) being greater than the short term peak pressure of 9.9 psig approximately (extracted from UFSAR Figure 6.2-10) justifies the current TS value of **'P_a'** as 11.5 psig. Also note that the definition of **'P_a'** given in 10 CFR Appendix J under "II Explanation of Terms" item I is: "Pa (p.s.i.g.) means the calculated peak containment internal pressure related to the design basis accident and specified either in the technical specification or associated bases."

Response

The two figures on the following page show the containment pressure and temperature profiles for the small steam line break event that results in the identified peak pressure of 11.9 psig. The times in the figures continue to 36 hours after initiation of the event, which captures the peak pressures and temperatures. The trend in these parameters continues to decrease as the containment cools. This analysis conservatively does not credit use of containment sprays, in order to capture a maximized drywell temperature profile. Use of containment sprays would rapidly reduce containment and drywell pressure.


GGNS 0.01 SQFT SSLB AT EPU POWER - EQDW10
41508691
03/12/10
11:33
Long-Term Peak Drywell Pressure is 18.0 psig @ 13.96 hours
Long-Term Peak Wetwell Pressure is 11.9 psig @ 13.96 hours
Pdw
Pww
Containment Pressure (psig)
Time (hours)
0
5
10
15
20
25
0
6
12
18
24
30
36


GGNS 0.01 SQFT SSLB AT EPU POWER - EQDW10
41508691
03/12/10
11:33
Tdw
Tww
Long-Term Peak Drywell Temperature is 320°F @ 1.04 hours
Long-Term Peak Wetwell Temperature is 149°F @ 9.12 hours
Temperature (°F)
Time (hours)
0
50
100
150
200
250
300
350
0
6
12
18
24
30
36

Regulatory Background

10 CFR 50, Appendix A, Criterion 16, Containment Design, requires:

> *Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.*

Appendix J to 10 CFR 50 describes the testing requirements to verify the leak-tight integrity of the primary containment. Three types of tests are required.

- *"Type A Tests" intended to measure the primary reactor containment overall integrated leakage rate at periodic intervals.*
- *"Type B Tests" means tests intended to detect local leaks and to measure leakage across each pressure-containing or leakage-limiting boundary for certain types of primary reactor containment penetrations.*
- *"Type C Tests" intended to measure containment isolation valve leakage rates.*

These tests are designed to ensure the containment barrier is essentially leak-tight to mitigate the release of radioactivity to the environment. In general, these tests are performed at the peak containment pressure (Pa), which is defined in Appendix J as:

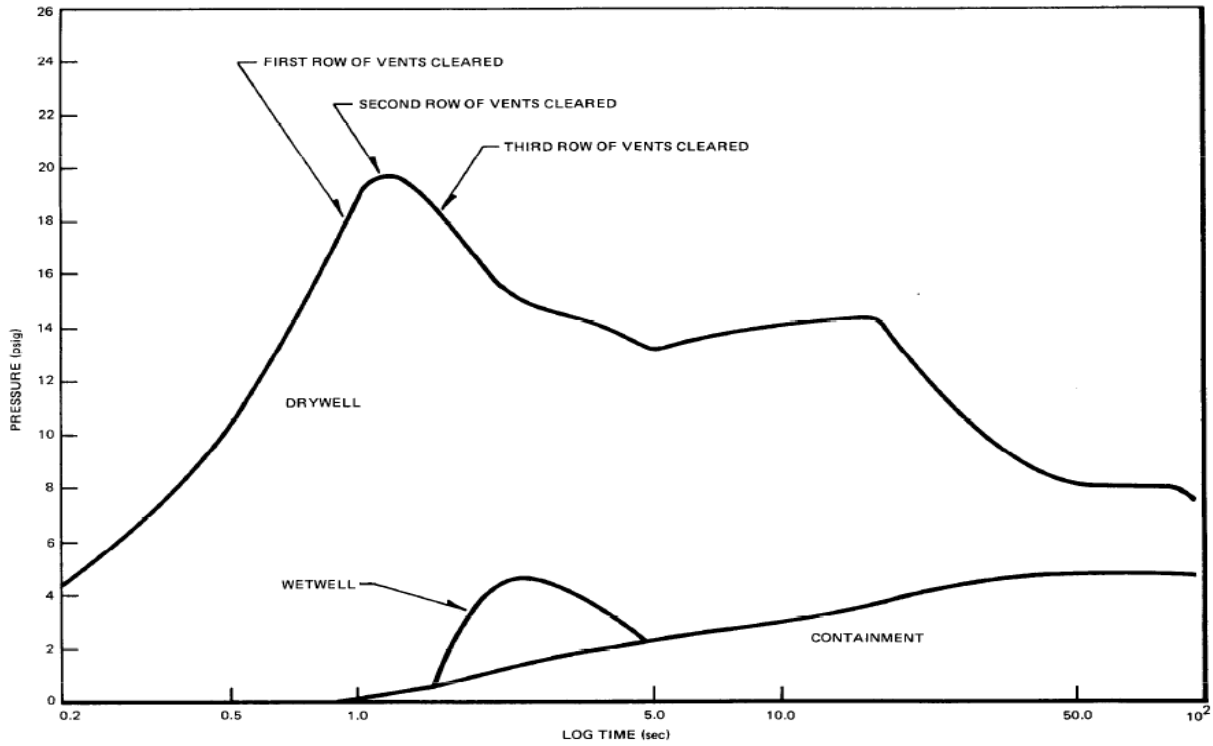
> *the calculated peak containment internal pressure related to the design basis accident and specified either in the technical specification or associated bases.*

At GGNS, Pa is not reported in the Technical Specifications although it is indirectly defined in the definition of the maximum allowable primary containment leakage rate (La). This definition indicates that this leakage rate is based on “the calculated peak containment pressure (Pa)”.

Current Licensing Basis

The GGNS containment analyses are described in Section 6.2 of the Update Final Safety Analysis Report (UFSAR) and address the recirculation line break and the main steam line break (MSLB). The short-term pressure results in UFSAR Figures 6.2-2 and 6.2-10 (provided below) both indicate a short-term pressurization of the area under the Hydraulic Control Unit (HCU) floor defined in the figures as the “wetwell.” This short pressurization appears to be worse for the MSLB where it reaches a pressure of approximately 9 psig before equilibrating back to the bulk containment pressure within the first 5 seconds of the accident. In all cases, this short-term pressurization is less than the long-term peak pressure of 11.5 psig which occurs in about 3.5 hours with minimum emergency core cooling system (ECCS) per UFSAR Figure 6.2-5 (provided below). Long-term peak pressure includes feedwater mass addition per UFSAR

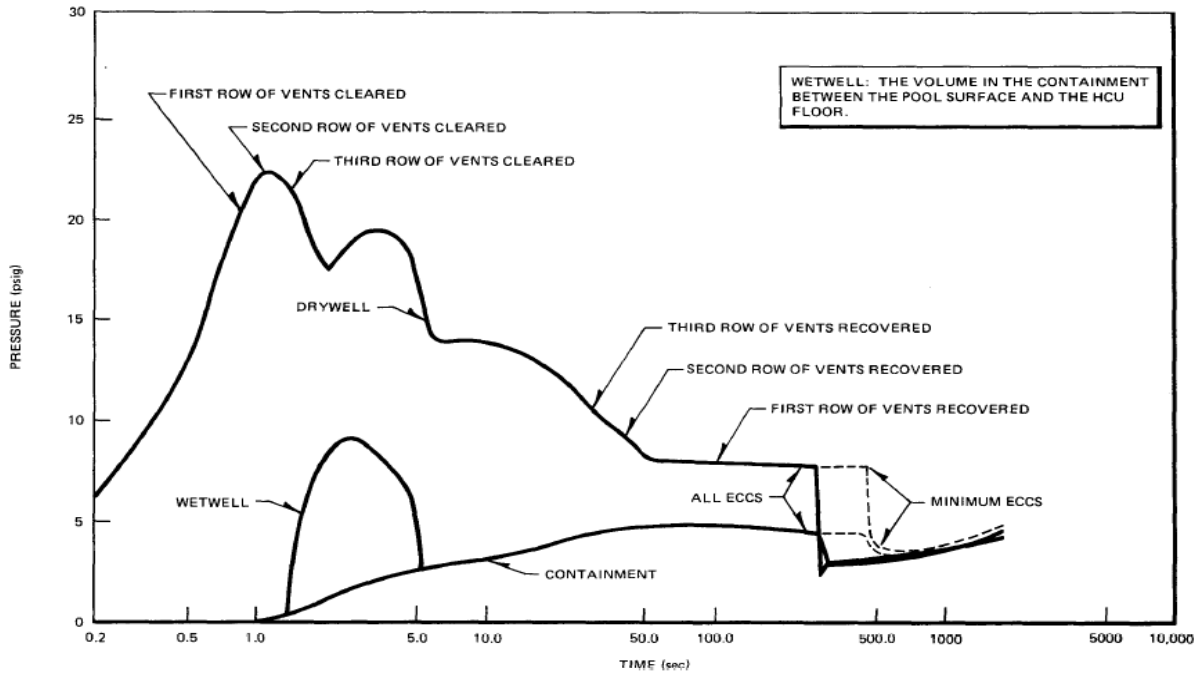
Table 6.2-13. Thus, the long-term peak is the highest containment pressure calculated for the 30-day evaluation period.



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	PRESSURE RESPONSE FOR RECIRCULATION LINE BREAK FIGURE 6.2-2

Rev. 0

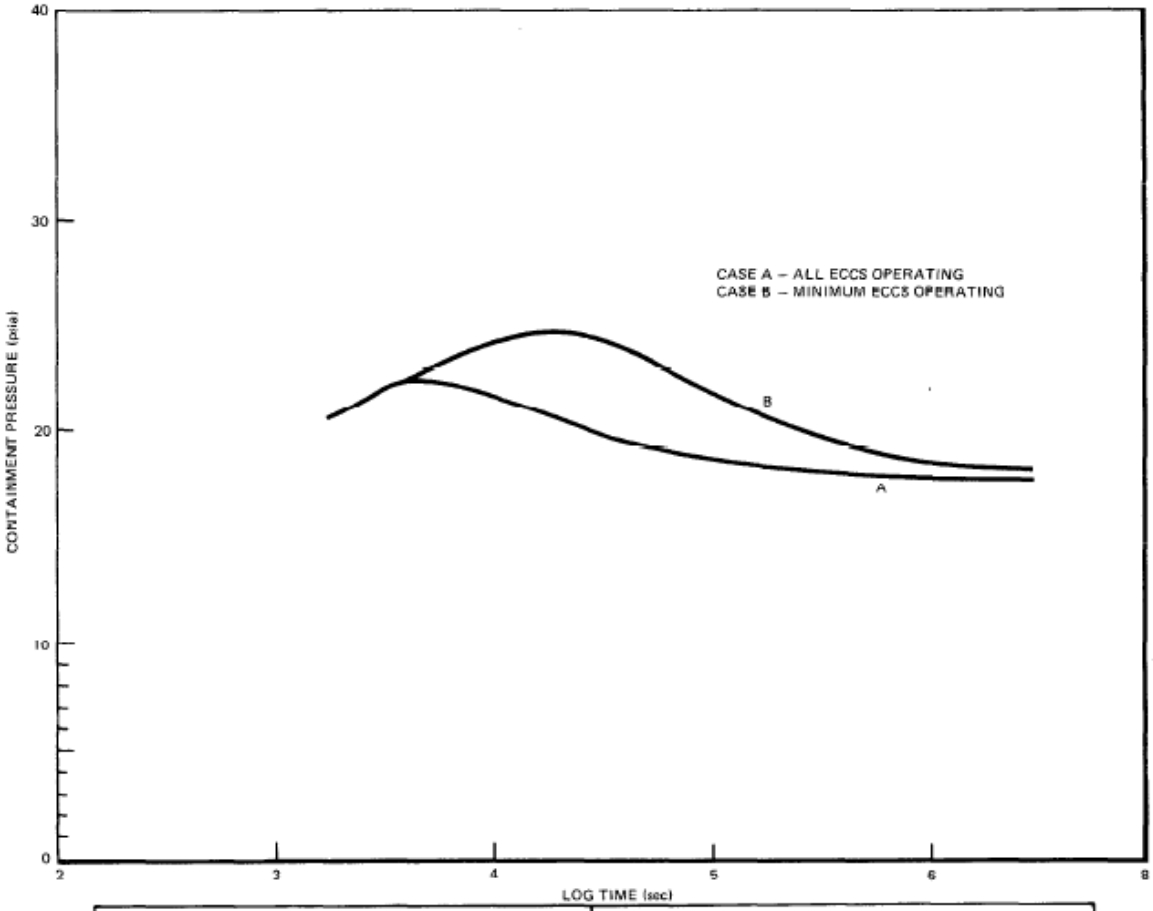
GGNS Current Licensed Thermal Power (CLTP)
Recirculation Line Break (Short-Term)



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	PRESSURE RESPONSE TO A STEAM LINE BREAK FIGURE 6.2-10

REV. 0

GGNS CLTP Main Steam Line Break (Short-Term)



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	CONTAINMENT PRESSURE RESPONSE FIGURE 6.2-5

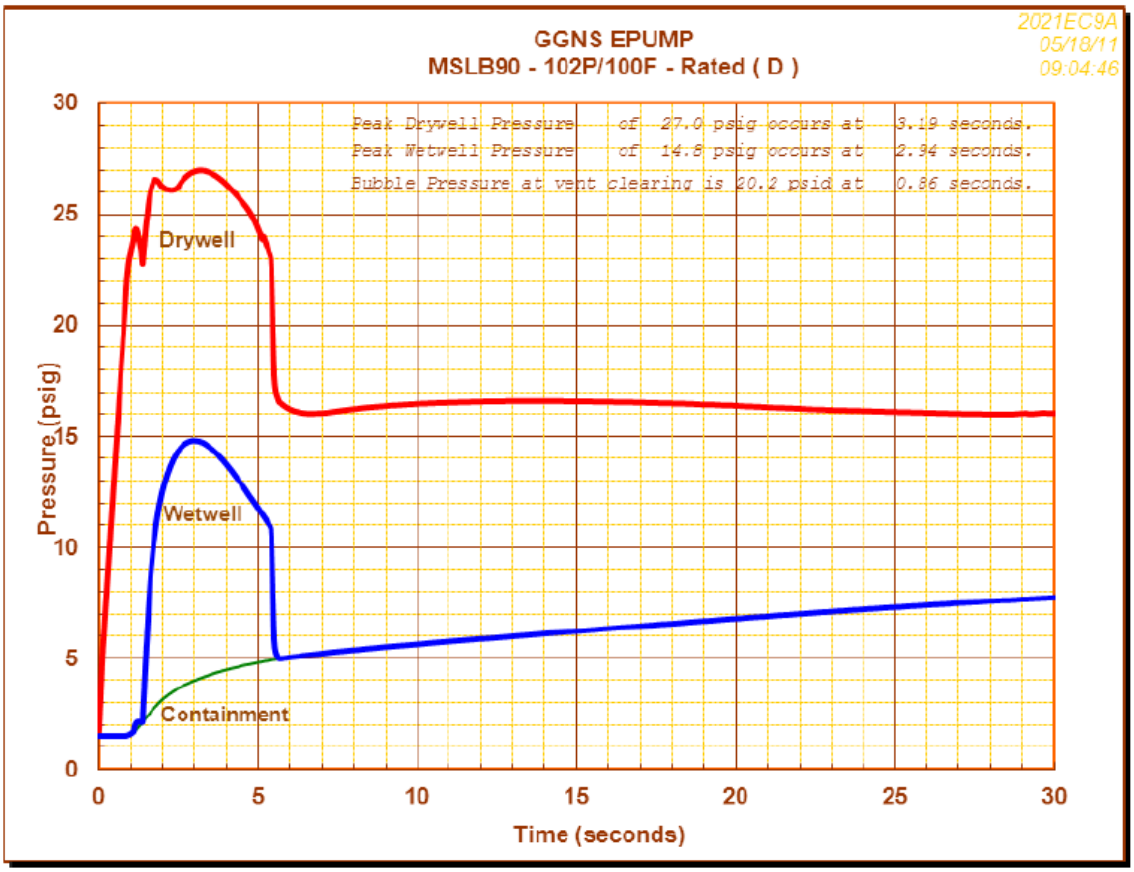
REV. 0

GGNS CLTP Containment Pressure Response (Long-Term)

Extended Power Uprate Licensing Basis

Due to changes in the GEH modeling methodology, the long-term peak does not bound the short-term peak for the EPU. The EPU analyses show trends similar to the CLTP analyses with the highest wetwell pressurization being associated with the MSLB. The analyses indicate that this short-term EPU pressure peak can reach 14.8 psig with timing similar to that associated with the CLTP case (i.e., < 6 seconds). The long-term pressure peak is driven by the steam line breaks resulting in a peak containment pressure of 11.9 psig occurring at about14 hours. The following revised Figure2.6-4, which is updated based on information provided in Entergy's June 8, 2011 letter to the NRC (NRC ADAMS Accession No. ML111590836), depicts the short-term peak for EPU conditions.

Figure 2.6-4 Short-Term DBA LOCA MSLB Pressure Response at EPU



The calculated EPU peak containment pressure is a very short pressure spike associated with compression of the drywell air emerging from the suppression pool in the localized region under

the HCU floor. This localized short-term peak need not be considered as part of Pa for the following three reasons.

1. Timing

This pressure spike is terminated within 6 seconds after a MSLB and before any significant source term inventory has been released from the reactor core. As described in Table 4 to Regulatory Guide 1.183, Rev. 0, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, the boiling water reactor (BWR) core source terms do not begin to be released from the reactor vessel until 2 minutes after a loss of coolant accident (LOCA). The only radioactivity released from the reactor during the first 6 seconds would be that associated with the reactor coolant which is very small and would be significantly scrubbed by the suppression pool before being released into the region between the pool and the HCU floor. Considering the primary containment function is to mitigate radioactivity leakage, the impact of any additional leakage rate associated with this early period would be negligible due to its low source term content.

2. Duration

This pressure spike lasts less than 6 seconds. Even if the leakage rate were increased by 24% based on the ratio of the pressures (14.8/11.9=1.24), the short period associated with increased leakage rate would result in a negligible increase in the amount of additional containment leakage.

3. Localized Impact

This pressure spike is limited to the region above the suppression pool and below the HCU floor (El. 135'4") and main steam tunnel (El. 140'). Based on a review of drawings, there are not many containment penetrations in this area and of these many are associated with ECCS return lines to the pool. The largest and most significant penetration in this area is the personnel hatch (penetration #3).

On these bases, the amount of radioactivity that may leak past the containment during this short period due to this additional pressure is considered to be insignificant.

Regulatory Precedence

By letter dated July 30, 1999, River Bend Nuclear Station (RBS) submitted a stretch power uprate license amendment request in which the short term and peak containment pressure responses were similar to the GGNS response. The RBS analysis indicated the short term peak wetwell pressure would reach 9.3 psig, based on the MSLB and the long term peak containment pressure would reach 3.6 psig, based on the MSLB and the recirculation line break. The value for P_a used for containment testing at RBS at the time of the submittal was

7.6 psig, which bounded the peak containment pressure calculated for the uprate. No change was proposed to the peak containment pressure (see the response to RAI 5 in NRC ADAMS Accession No. ML003734839). The NRC Safety Evaluation for the approval of the application was dated October 6, 2000 (NRC ADAMS Accession No. ML003762072).