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August 23, 2012

RBG-47278

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-001

Subject: Request for Exemption to 10 CFR 50 Appendix J
River Bend Station – Unit 1
License No. NPF-47
Docket No. 050-048

RBF1-12-0124

Dear Sir or Madam,

Pursuant to 10 CFR 50.12, Entergy Operations, Inc. hereby requests an exemption to certain requirements of 10 CFR 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." The proposed exemption relates solely to the definition of "Pa" in 10 CFR 50 Appendix J.

There are no new commitments in this letter. If you have any questions or require additional information, please contact Mr. Joseph Clark at 225-381-4177.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 23, 2012.

Respectfully,

Eric W. Olson

EWO/dhw

Attachment 1 – Technical Evaluation
Attachment 2 – Updated Safety Analysis Report Figures

A017
NRR

cc: U. S. Nuclear Regulatory Commission
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NRC Sr. Resident Inspector
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U.S. Nuclear Regulatory Commission
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Attachment 1
RBG-47278

Technical Evaluation

1.0 BACKGROUND

10 CFR 50 Appendix J details the test requirements that provide for preoperational and periodic verification by tests of the leak-tight integrity of the primary reactor containment, and systems and components which penetrate containment of water-cooled power reactors, and establish the acceptance criteria for these tests. The purposes of the tests are to assure that (a) leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values assumed in the accident analysis, and, (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment.

10 CFR 50 Appendix J defines "Pa" as the "calculated peak containment internal pressure related to the design basis accident and specified either in the technical specification or associated bases."

River Bend Station (RBS) Technical Specifications Section 5.5.13, "Primary Containment Leakage Rate Testing Program," applies the definition of Pa as follows: "The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 7.6 psig."

In July 1999, RBS submitted a license amendment request to increase the licensed thermal power of the station by 5% from 2894 MWth to 3039 MWth (Reference 1). As part of the application, the containment analysis for RBS was re-performed and the results summarized in the license amendment request. In response to a request for additional information regarding the containment analysis results (Reference 2), the analysis results were amplified.

The containment pressure response is shown in RBS Updated Safety Analysis Report (USAR) Figures 6.2-4 for main steam line breaks and 6.2-5 for reactor coolant recirculation line breaks. These curves (Attachment 2) show the pressure rise in containment and the wetwell. The containment is the air space above the hydraulic control unit (HCU) floor while the wetwell is the region between the HCU floor and the surface of the suppression pool.

As can be seen from USAR figures 6.2-4 and 6.2-5, the wetwell pressure is initially the same as the containment pressure. At about 2 seconds, the wetwell pressure increases rapidly while the containment pressure increases gradually. At about 5 to 6 seconds, the wetwell pressure decreases to the same pressure as the containment pressure. After this point, both the wetwell and containment pressure increase in unison to an eventual peak of 3.6 psig. All of the calculated pressures are within the design limits for containment pressure.

At that time, to avoid the large number of procedure changes which would be required if the value was changed, RBS elected to maintain Pa at the original (pre-uprate) value of 7.6 psig, which is conservative to the long term peak value calculated, 3.6 psig. In October 2000, RBS obtained approval of an amendment to the station operating license that raised the maximum authorized reactor power by approximately five percent (Reference 3).

In October 2011, RBS was contacted by NRC concerning the station's use of the Appendix J definition of Pa. A potential conflict between RBS' interpretation of that definition and a literal reading of the definition was raised considering the short term localized pressure spike in the wetwell. As discussed in Section 2.0 of this request, RBS does not believe that the short term localized pressure spike in wetwell pressure (9.3 psig considering Technical Specification allowable initial conditions) is appropriate to use for Pa considering Pa is used to determine the long term potential leakage for the containment as a whole.

2.0 TECHNICAL EVALUATION of ACCEPTABILITY

The current TS limit of 7.6 psig meets the underlying intent of the regulations and it is not appropriate to consider the short duration spike in wetwell peak pressure as Pa for the following reasons:

Timing

As described in Section 1 above, this pressure spike is terminated within 6 seconds after a main steam line break (MSLB) and before any significant source term inventory has been released from the reactor core, and before many containment isolation valves would have reached the closed position. 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). Therefore, this short term localized pressure spike has terminated before any significant source term inventory has been released from the reactor core and the purpose of 10 CFR 50 Appendix J testing to assure that leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values assumed in the accident analysis will continue to be met.

Localized Impact

This pressure spike is limited to the region above the suppression pool and below the HCU floor (El. 114'-0"). This represents a free volume of 128,160 ft^3 versus the net free volume for all of containment of 1,191,590 ft^3. Thus, this short duration peak pressure is applicable to the wetwell only and not to the containment in general. A review of plant drawings indicates that there are only a few containment penetrations in the wetwell region with the bulk of the containment penetrations above the wetwell. Therefore this pressure spike is not indicative of overall containment pressure, and a large majority of the valves will not be affected by this short term spike. Thus, the purpose of 10 CFR 50 Appendix J testing to assure that leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values assumed in the accident analysis will continue to be met.

Conclusion

Only a few containment penetrations are exposed to the wetwell pressure spike, which is expected to occur before any release of radionuclides from the reactor core. Therefore, use of the peak wetwell pressure for Pa is not appropriate. Continued use of the current Pa (7.6 psig) is appropriate as it bounds the peak bulk containment pressure of 3.6 psig and the purpose of 10 CFR 50 Appendix J testing to assure that leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values assumed in the accident analysis will continue to be met.

3.0 JUSTIFICATION and SPECIAL CIRCUMSTANCES

10 CFR 50.12, Specific exemptions, states that the Nuclear Regulatory Commission may grant exemptions from the requirements of the regulations of this part provided three conditions are met. The three conditions are: 1) the exemption is authorized by law, 2) the exemption will not present an undue risk to the health and safety of the public, 3) the exemption is consistent with the common defense and security. In addition, the Commission will not consider granting an exemption unless special circumstances are present.

3.1 This exemption is authorized by law.

Granting of the proposed exemption will not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations. As required by 10 CFR 50.12(a)(1), this requested exemption is "authorized by law". The NRC has the authority under the Atomic Energy Act and 10CFR50.12 to grant exemptions from the requirements of Part 50 upon showing proper justification. Entergy believes that the technical analysis presented above provides sufficient justification to support the proposed implementation of the requirements of Appendix J as they apply to RBS.

3.2 This exemption will not present an undue risk to public health and safety

The technical evaluation for this exemption request adequately justifies that use of the current Pa value for the site's Appendix J test program is consistent with the overall design function of the primary containment and the underlying purpose of the regulation, as well as the station's licensing basis as documented in the USAR.

3.3 This exemption is consistent with common defense and security

As noted above, the exemption request is only to apply, for the purposes of the Appendix J testing program, the long-term value of post-accident containment pressure rather than observe an unnecessarily literal definition of Pa contained in Appendix J. The request change does not affect any aspect of the security program and, as a result, is consistent with the common defense and security.

3.4 Special circumstances support the issuance of an exemption

10 CFR 50.12(a)(2) states that the NRC will not consider granting an exemption to the regulations unless special circumstances are present. The requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii) which states in that, "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

In this particular circumstance, a literal application of the subject regulation (i.e., the definition of "Pa") is not necessary to achieve the underlying purpose of the regulation. The short-term peak pressure in the wetwell area is not the appropriate value to be applied to the Appendix J test program. The phenomenon leading to that value is not indicative of the primary containment parameters that would exist in the long term following a design basis accident.

In order to reconcile an apparent disparity in the application of the requirements of Appendix J to the RBS primary containment test program (specifically, the definition of "Pa"), an exemption from the requirements of 10 CFR 50, Appendix J is requested. As required by 10 CFR 50.12, the requested exemption is authorized by law, presents no undue risk to public health and safety, and is consistent with common defense and security. Therefore, granting approval of this exemption request does not violate the underlying purpose of the rule and special circumstances exist to justify the approval of an exemption from the subject requirements.

As noted on Section 1.0 above, RBS Technical Specifications defines the value of Pa currently in use. Approval of this exemption request will therefore require no amendment to the station's operating license.

5.0 REGULATORY COMMITMENTS

None

6.0 REFERENCES

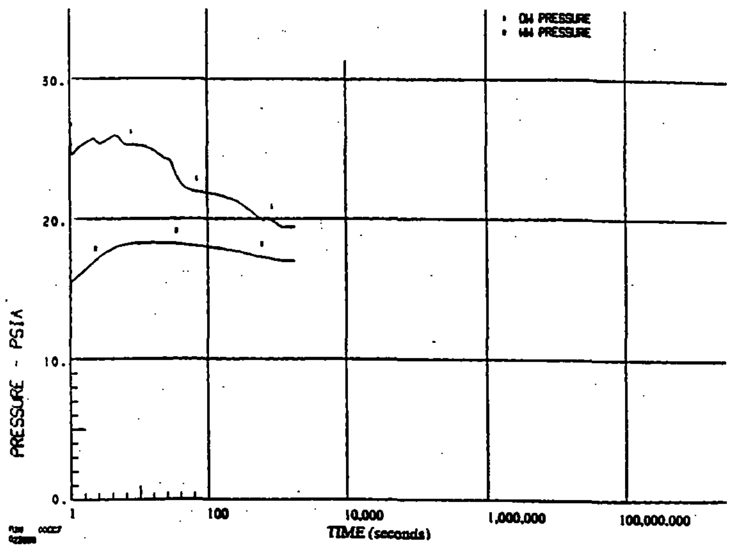
6.1 Letter from Randall K. Edington, EOI, to NRC, "License Amendment Request (LAR) 99-16, Change to Technical Specifications for Power Uprate of River Bend Station," July 30, 1999 (ML003746428).

6.2 Letter from Rick J. King, EOI, to NRC, "Additional Information Related to License Amendment Request (LAR) 99-15, Changes to Technical Specifications for Power Uprate of River Bend Station," July 18, 2000 (ML003734849).

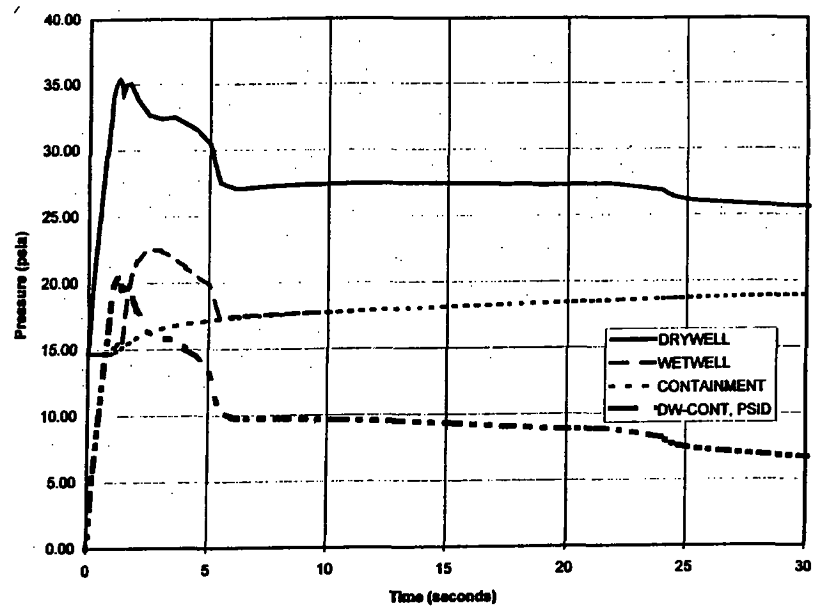
6.3 Letter from Jeffery F. Harold, NRC, to EOI, "River Bend Station, Unit 1 – Issuance of Amendment re: Increase in Maximum Allowable Thermal Power to 3039 Megawatts Thermal," October 6, 2000 (ML003762072)

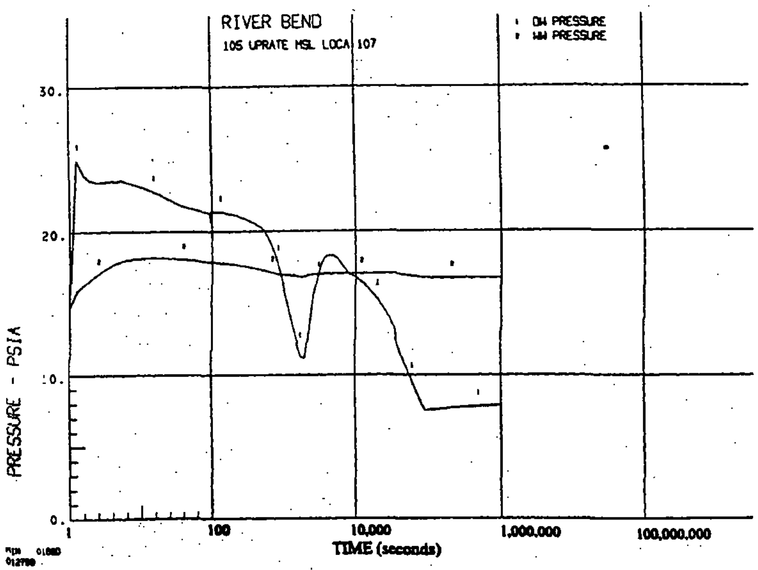
Attachment 2
RBG-47278

Updated Safety Analysis Report
Figures 6.2-4 and 6.2-5



DBA-LOCA Long-Term Drywell and Containment Airspace Pressure. MSLB 102% Uprated Power/ 100% Rated Core Flow, 0-1800 sec, (SHEX-04V)





DBA-LOCA Long-Term Drywell and Containment Airspace Pressure. MSLB 102% Uprated Power/ 100% Rated Core Flow, t >1800 sec, (SHEX-04V)

NOTES:

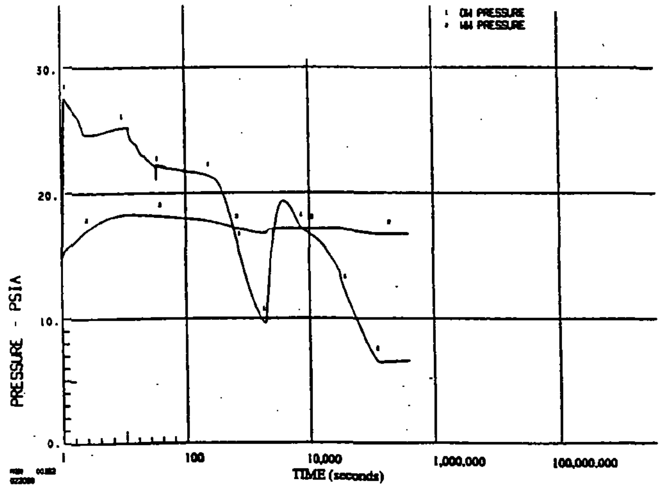
1) SHORT TERM RESPONSE IS BASED ON INITIAL HIGH SUPPRESSION POOL LEVEL AND ZERO FEEDWATER ADDITION
2) LONG TERM RESPONSE IS BASED ON INITIAL LOW SUPPRESSION POOL LEVEL AND ADDITION OF ALL HOT FEEDWATER
3) LONG TERM WETWELL RESPONSE FOLLOWS THAT OF THE CONTAINMENT

Figure 6.2-4

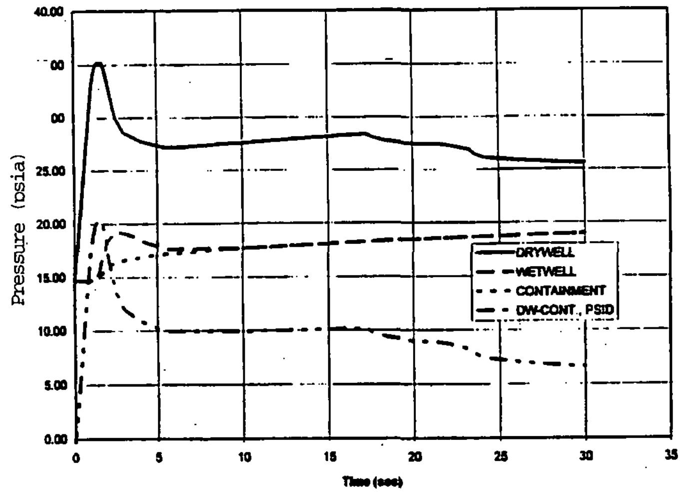
CONTAINMENT PRESSURE RESPONSE FOR MAIN STEAM LINE BREAK

RIVER BEND STATION

UPDATED SAFETY ANALYSIS REPORT



DBA-LOCA Long-Term Drywell and Containment Airspace Pressure. RCLB 102% Uprated Power/ 100% Rated Core Flow, (SHEX-04V)



DBA-LOCA Short-Term Drywell, Wetwell and Containment Airspace Pressure. RCLB 102% Uprated Power/ 100% Rated Core Flow, 0-30 sec, (M3CPT05V)

NOTES:

1) SHORT TERM RESPONSE IS BASED ON INITIAL HIGH SUPPRESSION POOL LEVEL AND ZERO FEEDWATER ADDITION
2) LONG TERM RESPONSE IS BASED ON INITIAL LOW SUPPRESSION POOL LEVEL AND ADDITION OF ALL HOT FEEDWATER
3) LONG TERM WETWELL RESPONSE FOLLOWS THAT OF THE CONTAINMENT

FIGURE 6.2-5

CONTAINMENT PRESSURE RESPONSE FOR RECIRCULATION LINE BREAK

RIVER BEND STATION

UPDATED SAFETY ANALYSIS REPORT