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1CAN080903

August 31, 2009

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

SUBJECT: Updated Information Regarding Technical Specification Changes and Analyses Relating to Use of Alternate Source Term In Reference to NRC/PWROG Meeting Held June 25, 2009
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

REFERENCES:

1. Entergy letter dated October 22, 2007, "License Amendment Request: Technical Specification Changes and Analyses Relating to Use of Alternate Source Term" (1CAN100703) (TAC No: MD7178)
2. Entergy letter dated March 13, 2008, "Supplement to Amendment Request: Technical Specification Changes and Analyses Relating to Use of Alternate Source Term" (1CAN030803) (TAC No: MD7178)
3. Entergy letter dated April 3, 2008, "Supplement to Amendment Request: Technical Specification Changes and Analyses Relating to Use of Alternate Source Term" (1CAN040802) (TAC No: MD7178)
4. NRC letter dated August 13, 2008, "Arkansas Nuclear One, Unit 1 – Re: Request for Additional Information Regarding Application of the Alternative Source Term" (1CNA080804) (TAC No. MD7178)
5. Entergy letter dated August 14, 2008, "Response to Request for Additional Information Regarding Technical Specification Changes and Analyses Relating to Use of Alternate Source Term" (1CAN080801) (TAC No. MD7178)
6. Entergy letter dated September 18, 2008, "Response to Request for Additional Information Regarding Technical Specification Changes and Analyses Relating to Use of Alternate Source Term" (1CAN090802) (TAC No. MD7178)
7. NRC letter dated July 31, 2009, "Arkansas Nuclear One, Unit No. 1 – Individual Plant Actions Re: Pressurized-Water Reactor Owners Group Topical Report BAW-2374, Revision 2, "Risk-Informed Steam Generator Tube Thermal Loads due to Breaks in Reactor Coolant System Upper Hot Let Large-Bore Piping"" (1CNA070901) (TAC No. MD7178)

Dear Sir or Madam:

By letter (Reference 1), Entergy Operations, Inc. (Entergy) proposed a change to the Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TSs) to support adoption and use of Alternate Source Term (AST) in the ANO-1 Safety Analyses.

By letter dated July 31, 2009 (Reference 7), the NRC requested Babcock & Wilcox (B&W) licensees provide information relating to once-through Steam Generator (SG) tube loads under conditions resulting from postulated breaks in reactor coolant system (RCS) upper hot leg large-bore piping. The letter contained six questions and also requested further dialogue with Arkansas Nuclear One, Unit 1 (ANO-1) to determine impact, if any, on the ANO-1 request to adopt an Alternate Source Term (AST), currently under NRC review. Applicable NRC and ANO-1 personnel participated in a conference call on August 6, 2009, to discuss appropriate responses and the AST submittal. As a result of the call, Entergy is providing a response to the questions presented in the NRC July 31, 2009 letter (Reference 7) under separate cover and is also providing a revised response to Question 3 of the Response to Additional Information (RAI) as previously depicted in Entergy letter dated August 14, 2008 (Reference 5), in Attachment 1 to this letter.

During the aforementioned conference call, the NRC also requested that the implementation of AST, if approved, be delayed until an analysis of tube loading under large break Loss of Coolant Accident (LBLOCA) conditions (e.g., a break in the U-bend or “candy-cane” region of the RCS hot leg) is complete. Therefore, Attachment 2 of this letter establishes an Operating License condition to ensure the relationship between this analysis and AST implementation is maintained.

Entergy requests the NRC use the revised response to Question 3 of the RAI in Attachment 1 of this letter discussed above in place of the response provided in Entergy letter dated August 14, 2008 (Reference 5).

In addition to the above, Entergy received a RAI from the NRC’s electrical branch via email on August 27, 2009. The questions and Entergy’s responses are included in Attachment 3 of this letter. A draft of the response was discussed with the NRC in conference call on August 25, 2009.

There are no technical changes proposed that impact the original no significant hazards consideration included in Reference 1. There are no new commitments contained in this letter. However, an Operating License condition is proposed as discussed above.

If you have any questions or require additional information, please contact David Bice at 479-858-5338.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 31, 2009.

Sincerely,

Original signed by K. T. Walsh

KTW/dbb

Attachments:
1. Updated Information Regarding Technical Specification Changes and Analyses Relating to Use of Alternate Source Term Related to June 25, 2009 NRC/PWROG Meeting
2. Markup of Affected Operating License Page
3. Response to Request for Additional Information – Electrical Branch

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Attachment 1 to

1CAN080903

Updated Information Regarding Technical Specification Changes and Analyses Relating to Use of Alternate Source Term in Reference to the NRC/PWROG Meeting Held June 25, 2009

Updated Information Regarding Technical Specification Changes and Analyses Relating to Use of Alternate Source Term in Reference to the NRC/PWROG Meeting Held June 25, 2009

Based on a meeting held between the NRC and the Pressured Water Reactor Owners Group (PWROG) on June 25, 2009, and subsequent NRC letter dated July 31, 2009 (Reference 1), Entergy Operations, Inc. (Entergy) has revised the response to Question 3 of the previous NRC Request for Additional Information (RAI) in NRC letter dated August 13, 2008 (Reference 2). Entergy's original response was provided in letter dated August 14, 2008 (Reference 3). Entergy requests the NRC use the below revised response to Question 3 of the RAI in place of the response provided in Entergy letter dated August 14, 2008 (Reference 3).

C. DOSE CACULATIONAL METHODOLOGY

3. *The ANO-1 SAR section 14.2.2.6, "Maximum Hypothetical Accident" states that: "In order to demonstrate that the operation of ANO-1 does not produce undue risk to the public under any accident conditions, the dose that would be received at the exclusion distance and the low population zone from a release of radioactivity larger than any which could actually occur is calculated."*

   This ANO-1 SAR description conforms to the postulated accident described in footnote 1 of 10 CFR 50.67, "Accident Source Term", which states, "The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products."

   The worst credible accident for dose consequence purposes is not necessarily defined as the design-basis accident for meeting 10 CFR 50.46 criteria for exclusion area boundary systems. For dose consequence purposes, the deterministic source term is postulated from the worst credible event used to test a facility's engineered safety features and are intentionally conservative to compensate for uncertainties in accident progression, fission product transport, and atmospheric dispersion.

   Please describe the ANO-1 accident that is considered the Maximum Hypothetical Accident for the AST analysis including the evaluation and any associated reanalysis that was done in order to ensure that ANO-1 has identified the most limiting accident under AST conditions. Also, since ANO-1 is a member of the PWR Owners Group that has requested formal review of BAW-2374 Revision 2 topical report, the NRC staff requests that ANO-1 clearly describe its AST analysis of the leakage pathway through failed steam generator tubes during a worst case dose consequence accident. If ANO-1 is susceptible to a consequential failure of the steam generator tubes during a large break LOCA, the NRC staff requests that the licensee describe how it meets the 10 CFR 50.67 regulatory criteria for protecting public health and safety during this credible design-basis accident. Furthermore, the NRC staff requests that the licensee describe the changes being made to its design and licensing basis and Technical Specifications as a result of this analysis.

Response:

The current Arkansas Nuclear One, Unit 1 (ANO-1) Maximum Hypothetical Accident (MHA) analysis reported in the Safety Analysis Report (SAR) is a Loss-of-Coolant-Accident (LOCA) that assumes the TID-14844 source term. The ANO-1 Alternate Source Term (AST) LOCA analysis is proposed to replace this analysis. The current ANO-1 LOCA analysis reported in the SAR assumes more realistic conditions than the MHA analysis, including a significantly lower fission product inventory, and is therefore bounded by the current ANO-1 MHA analysis. Due to the bounding nature of the Regulatory Guide (RG) 1.183 LOCA assumptions, a separate "realistic" LOCA analysis using the alternate source terms would not provide any meaningful insights and thus is considered unnecessary. Therefore, Entergy proposes to replace the separate MHA and realistic LOCA dose analyses currently in the ANO-1 SAR with a single, bounding AST LOCA analysis. As shown in Table 3-1 of the March 13, 2008 supplement (Reference 4), the worst credible ANO-1 AST accidents for dose consequence purposes are the LOCA for absolute Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) doses and the Control Rod Ejection Accident (CREA) (secondary release case) for control room dose. With AST, ANO-1 is proposing to delete use of the antiquated "Maximum Hypothetical Accident" terminology in the interest of maintaining a clear design basis per the guidance of RG 1.183, Regulatory Position 1.6.

Note the intent of any analysis is to ensure the "most limiting" accidents or conditions are used to verify accident consequences remain acceptable. The ANO-1 AST submittal and proposed SAR changes have appropriately identified these most limiting accidents and use of the antiquated MHA term is, therefore, being retired.

ANO-1 replaced its steam generators in the fall of 2005 with Areva's enhanced once-through steam generators (EOTSG) containing Inconel 690 tubes, which are not known to be susceptible to circumferential cracking. The tubes are hydraulically expanded the full depth of the 24" thick tube sheet and welded to form a leak-tight barrier. These and other improvements in the steam generator design greatly reduces the potential for failure of steam generator tubes and the tube-tubesheet interface following a hot leg U-bend LOCA. ANO-1 intends to add the hot leg U-bend LOCA to its design basis for the EOTSGs (i.e. qualify the EOTSGs to this event) and develop tube integrity inspection criteria that will ensure that any degraded tubes left in service will withstand the event should it occur. This work is scheduled to be complete by early Spring 2010, and will demonstrate that ANO-1 is not susceptible to a consequential failure of the steam generator tubes during a large break LOCA (LBLOCA). No changes to the ANO-1 current licensing basis (CLB) or Technical Specifications, due to inclusion of the event in the EOTSG design basis, have yet been identified. As stated above, the ANO-1 AST submittal and proposed SAR changes have appropriately identified and analyzed the most limiting accidents from a dose perspective. LBLOCA is currently considered in the operational assessments for all flaws including wear. The current circumferential extent of wear is very low. It should be noted that the PWROG does not intend to pursue further review and approval by the NRC of BAW-2374, Revision 2 (Reference 5).

Attachment 2 contains an Operating License condition requiring the aforementioned analysis to demonstrate ANO-1 EOTSG tube integrity during a LBLOCA, including a LOCA in the hot leg U-bend region.

REFERENCES

1. NRC letter dated July 31, 2009, "Arkansas Nuclear One, Unit No. 1 – Individual Plant Actions Re: Pressurized-Water Reactor Owners Group Topical Report BAW-2374, Revision 2, "Risk-Informed Steam Generator Tube Thermal Loads due to Breaks in Reactor Coolant System Upper Hot Let Large-Bore Piping"" (1CNA070901) (TAC No. MD7178)

2. NRC letter dated August 13, 2008, "Arkansas Nuclear One, Unit 1 – Re: Request for Additional Information Regarding Application of the Alternative Source Term" (1CNA080804) (TAC No. MD7178)

3. Entergy letter dated August 14, 2008, "Response to Request for Additional Information Regarding Technical Specification Changes and Analyses Relating to Use of Alternate Source Term" (1CAN080801) (TAC No. MD7178)

4. Entergy letter dated March 13, 2008, "Supplement to Amendment Request: Technical Specification Changes and Analyses Relating to Use of Alternate Source Term" (1CAN030803) (TAC No: MD7178)

5. BAW-2374, Revision 2, "Risk-informed Assessment of Once-Through Steam Generator Tube Thermal Loads due to Breaks in Reactor Coolant System Upper Hot Leg Large-Bore Piping"

Attachment 2 to

1CAN080903

Markup of Affected Operating License Page

(5) <u>Implementation of the Improved Technical Specifications (ITS)</u>

The licensee is authorized to relocate certain Technical Specification requirements previously included in Appendix A to licensee controlled documents, as described in Table R, Relocated Specifications, and Table LA, Removal of Details, attached to the Safety Evaluation for Amendment No. 215. These requirements shall be relocated to the appropriate documents as part of the implementation of the ITS.

The schedule for performing Surveillance Requirements (SRs) that are new or revised in Amendment No. 215 shall be as follows:

1. For SRs that are new in this amendment, the first performance shall be due at the end of the first surveillance interval, which begins on the date of implementation of this amendment.

2. For SRs that existed prior to this amendment whose intervals of performance are being reduced, the first reduced surveillance interval shall begin upon completion of the first surveillance performed after implementation of this amendment.

3. For SRs that existed prior to this amendment that contained modified acceptance criteria, the performance shall be due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment.

4. For SRs that existed prior to this amendment whose interval of performance are being extended, the first extended surveillance interval shall begin upon completion of the last surveillance performed prior to the implementation of this amendment.

(6) ~~Deleted~~<u>In accordance with Technical Specification Amendment xxx, the 60-day implementation period associated with the adoption of Alternate Source Term shall be delayed pending verification that the Steam Generators are designed to withstand the loading associated with the worst-case large break Loss of Coolant Accident and that tube integrity will be maintained for this LBLOCA as verified through implementation of Technical Specification 6.5.9, Steam Generator Program. This verification shall be completed no later than April 1, 2010, and the results of the verification shall be provided to the NRC within 30 days thereafter.</u>

(7) Deleted

Attachment 3 to

1CAN080903

Response to Request for Additional Information – Electrical Branch

Response to Request for Additional Information – Electrical Branch

The following are additional questions received from the NRC's electrical branch via email on August 27, 2009. Entergy Operations, Inc. (Entergy) has provided a response to each question below. These responses were discussed with the NRC in teleconference held on August 25, 2009.

1. *Are there any electrical non-safety related systems and components credited in the alternate source term analyses?*

   If so,

   a) *Describe how this system will be electrically separated from the safety-related system (i.e., provide a detailed discussion on how a fault on the non-Class 1E electrical circuit will not propagate to the Class 1E electrical circuit).*

   b) *Describe the independence (e.g., electrical and physical separation) and redundancy of these systems.*

   c) *Describe how these systems meet the single failure criterion.*

   d) *Describe how the operators will be notified in the event that these systems and components would become inoperable (e.g., control room annunciators).*

   e) *Describe any impacts on seismic qualifications of these systems and components.*

   Response:

   The Arkansas Nuclear One, Unit 1 (ANO-1) Alternate Source Term (AST) does not rely on any non-safety components that require electrical input.

2. *Are any loads being added to the ANO-1 emergency diesel generators (EDGs)? If so, describe how the loads being added to the EDGs affect the capability and capacity of the EDGs (e.g., describe the impact of the proposed change on the EDG ratings) and provide the loading sequence changes for each EDG.*

   Response:

   No loads are added to the ANO-1 EDGs as a result of AST adoption.

3. *Provide a list and description of components being added to your Title 10 to the Code of Federal Regulations Section 50.49 (10 CFR 50.49) program due to this LAR. Confirm that these components are qualified for the environmental conditions they are expected to be exposed to.*

   Response:

   No components are added to the ANO-1 10 CFR 50.49 program as a result of AST adoption.

4. *Are there any changes in the chemical composition of the chemical spray solution as a result of this LAR? If so, provide the chemical composition and provide a detailed evaluation to show the components are qualified for the environmental conditions they are expected to be exposed to. Also, describe, if any, changes in the operation of the chemical spray system and its impact on the environment.*

   Response:

   The chemical composition of the ANO-1 chemical spray solution and operation of the ANO-1 chemical spray system are not changed as a result of AST adoption.