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Richard L. Anderson
ANO Site Vice President

10 CFR 50.71(e)

1CAN111802

November 12, 2018

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

SUBJECT: ANO Unit 1 SAR Amendment 28, TRM, TS Bases, 10 CFR 50.59 Report, and Commitment Change Summary Report
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

Dear Sir or Madam:

In accordance with 10 CFR 50.71(e) and 10 CFR 50.4(b)(6), enclosed is an electronic copy of Amendment 28 to the Arkansas Nuclear One, Unit 1 (ANO-1) Safety Analysis Report (SAR). Included with this update is an electronic copy of the current ANO-1 Technical Requirements Manual (TRM) and the current ANO-1 Technical Specification (TS) Bases. The TS Bases file also includes the Table of Contents which outlines the contents of both the TSs and the TS Bases, since the Table of Contents is revised by the licensee in accordance with 10 CFR 50.59. Pursuant to 10 CFR 50.71(e)(4), these documents are being submitted within six months following the previous ANO-1 refueling outage (1R27) which ended May 22, 2018. Summaries of changes to the ANO-1 TRM and TS Bases are included in Attachments 1 and 2 of this letter, respectively. The SAR, TS Bases, and TRM changes enclosed are for the period beginning June 8, 2017, and ending November 12, 2018.

In accordance with NEI 98-03, Appendix A, Section A6, a list and short description of information removed from the SAR should be included with each SAR update submittal. For this reporting period, information was not removed from the SAR meeting the criteria of either Appendix A, Sections A4 or A5, of NEI 98-03, that would require reporting in accordance with NEI 98-03, Appendix A, Section A6.

Associated in part with post September 11, 2001, response related to security sensitive information, Entergy has reviewed the ANO-2 SAR and determined that the following items contain information required to be withheld from public disclosure with respect to NRC Regulatory Issue Summary (RIS) 2015-17, “Review and Submission of Updates to Final Safety Analysis Reports, Emergency Preparedness Documents, and Fire Protection Documents.”

SAR Section 2.4.4.1, “Maximum Probable Flood”

SAR Section 2.4.4.2, “Failure of Upstream Dams”

SAR Section 2.4.4.3, “Design Flood Elevation”

The above is consistent with currently redacted information from the ANO-1 SAR (reference ML17297B948). Entergy requests the aforementioned information be withheld from public disclosure in accordance with 10 CFR 2.390. Accordingly, a complete version and a redacted version of the ANO-1 SAR are included on the enclosed compact disc (CD).

In accordance with 10 CFR 54.37(b), after a renewed license is issued, the SAR update required by 10 CFR 50.71(e) must include any systems, structures, and components (SSCs) newly identified that would have been subject to an aging management review or evaluation of time-limited aging analyses in accordance with 10 CFR 54.21. The SAR update must describe how the effects of aging will be managed such that the intended function(s) in 10 CFR 54.4(b) will be effectively maintained during the period of extended operation. No SAR changes were required with respect to 10 CFR 50.37(b) during this reporting period.

A summary of ANO-1 10 CFR 50.59 evaluations and those evaluations common between ANO-1 and ANO Unit 2 (ANO-2) associated with changes to Licensing Basis Documents over the reporting period is provided in Attachment 3. Attachment 4 contains a copy of each evaluation.

Attachment 5 contains a summary of changes to regulatory commitments which have occurred over the reporting period.

Attachment 6 includes a list of SAR pages that were updated during the period.

If you have any questions or require additional information, please contact Stephenie Pyle at 479-858-4704.

I hereby certify that to the best of my knowledge and belief, the information contained in the above Licensing Basis Documents accurately reflects changes made since the previous submittal. The changes to these documents reflect information and analyses submitted to the Commission, prepared pursuant to Commission requirements, or made under the provisions of 10 CFR 50.59. Executed on November 12, 2018.

Sincerely,

ORIGINAL SIGNED BY RICHARD L. ANDERSON

RLA/dbb

Attachments:

1. Summary of ANO-1 TRM Changes
2. Summary of ANO-1 TS Bases Changes
3. Summary of ANO-1 and ANO-Common 10 CFR 50.59 Evaluations
4. 10 CFR 50.59 Evaluations – June 8, 2017, through November 12, 2018
5. ANO-1 and ANO-2 Commitment Change Summary Report
6. List of Affected SAR Pages

Enclosures (compact disc):

1. ANO-1 SAR Amendment 28 – Un-redacted Version (CD Rom)
2. ANO-1 SAR Amendment 28 – Redacted Version (CD Rom)
3. ANO-1 TRM
4. ANO-1 TS Table of Contents and TS Bases

cc: Mr. Kriss M. Kennedy
Regional Administrator
U. S. Nuclear Regulatory Commission
RGN-IV
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Arlington, TX 76011-4511

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Slot #30
Little Rock, AR 72205

Attachment 1 to

1CAN111802

Summary of ANO-1 TRM Changes

Summary of ANO-1 TRM Changes

The following changes to the Arkansas Nuclear One, Unit 1 (ANO-1) Technical Requirements Manual (TRM) were implemented in accordance with the provisions of 10 CFR 50.59. Because these changes were implemented without prior NRC approval, a description is provided below:

Revision #	TRM Section	Description of Change
61	TRO 3.7.8 TRO 3.7.12 B 3.3.6 B 3.7.8 B 3.7.12	Condition Reports CR-ANO-2-2015-2511, "Clarification of Inoperable Detector Actions for Fire Suppression Systems Non-Functionalities" and CR-ANO-C-2017-3030, "Clarify TRO Note Testing Exception"
62	TRO 3.3.7 TRM 5.5.1 B 3.5.1	Licensing Basis Document Change LBDC 17-062, "Correct MET Tower Condition B Wording", Licensing Basis Change LBDC-17-063, "Revise the Code of Record for ANO-1 Snubber Program the 5th 10 year IST interval"
63	TRO 3.4.11 Table 3.7.12-2	Engineering Change EC-73815, "ANO-1 Void Area Grease Cap Inspections" and Licensing Basis Document Change LBDC 18-013, "Delete Redundant DHR Relief Valve Maintenance"
64	TRO 3.7.12 Table 3.7.12-1 TRO 3.7.13 B 3.7.13	Licensing Basis Document Change LBDC 18-016, "Transition to NFPA 805" and Engineering Change EC-73886, "Fire Protection Engineering Evaluation Updates"
65	TRO 3.7.12 TR 3.7.12.1 TR 3.7.12.2 B 3.7.12	Licensing Basis Document Change LBDC 18-016, "Transition to NFPA 805 – Fire Wraps"

<u>List of Undefined Acronyms</u>

DHR	Decay Heat Removal
MET	Meteorological Tower
NFPA	National Fire Protection Association
TR	Technical Requirement
TRO	Technical Requirement for Operation

Attachment 2 to

1CAN111802

Summary of ANO-1 TS Bases Changes

Summary of ANO-1 TS Bases Changes

The following changes to the Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specification (TS) Bases were implemented in accordance with the provisions of 10 CFR 50.59 and the Bases Control Program of ANO-1 TS 5.5.14. Because these changes were implemented without prior NRC approval, a description is provided below:

Revision #	TS Bases Section	Description of Change
59	B 3.4.16	TS Amendment 258, "TSTF-510 SG Tube Integrity Program"
60	B 3.4.10 B 3.4.14 B 3.5.2 B 3.6.3 B 3.6.5 B 3.7.1 B 3.7.2 B 3.7.3 B 3.7.5	Licensing Basis Document Change LBDC 17-063, "Revise the Code of Record for the ANO-1 Snubber Program 5th 10-year interval" and Licensing Basis Document Change LBDC 17-058, "Revise TS Bases to Match TS 3.7.5, Action D.1 Note"
61	B 3.0.1 B 3.0.9	TS Amendment 259, "TSTF-427 Barrier Degradation"
62	B 3.7.5	TS Amendment 260, "TSTF-412 One Inoperable EFW Steam Supply"
63	B 3.3.15	Licensing Basis Document Change LBDC 18-040, "Adopt TSTF-539-T, Correction of PAM Bases"
64	B 3.7.5	TS Amendment 261, "Apply 7-Day Completion Time to EFW Steam Supply DC-Powered MOVs"

<u>List of Undefined Acronyms</u>

DC	Direct Current
EFW	Emergency Feedwater
MOV	Motor Operated Valve
PAM	Post Accident Monitoring
SG	Steam Generator
TSTF	Technical Specification Task Force

Attachment 3 to

1CAN111802

Summary of ANO-1 and ANO-Common 10 CFR 50.59 Evaluations

Summary of ANO-1 and ANO-Common 10 CFR 50.59 Evaluations

50.59 #	**50.59 Summary**
2018-001	Engineering Change EC-69811, "Cycle 28 Reload, Core Operating Limits Report (COLR) Refueling Boron (RFB) Concentration Limit Change and Reanalysis of the Moderator Dilution Accident (MDA) Event during Refueling Conditions"

Attachment 4 to

1CAN111802

10 CFR 50.59 Evaluations – June 8, 2017, and ending November 12, 2018

ANO 50.59 Evaluation Number

18-001

I. OVERVIEW / SIGNATURES[1]

Facility: Arkansas Nuclear One, Unit 1 **Evaluation #** FFN-2018-001 **/ Rev. #:** 0

Proposed Change / Document: EC 69811 Cycle 28 Reload, Core Operating Limits Report (COLR) Refueling Boron (RFB) concentration limit change and reanalysis of the Moderator Dilution Accident (MDA) event during refueling conditions

Since the MDA during refueling was required to be re-run to demonstrate that all required safety functions and design requirements are met, the change is considered to be adverse and must be screened in.

Description of Change:

EC 69811 Cycle 28 Reload Process Applicability Determination (PAD) identified an adverse change. The Cycle 28 reload report and reload technical document identified that the Analysis of Record (AOR) for the MDA event during refueling conditions was reanalyzed based on the Cycle 28 specific RFB concentration that is provided in the COLR. The guidance provided in CR-HQN-2015-00684 CA 4 and Revision 1 to NEI-96-07 which states: "*If the effect of a change is such that existing safety analyses would no longer be bounding and therefore UFSAR safety analyses must be re-run to demonstrate that all required safety functions and design requirements are met, the change is considered to be adverse and must be screened in*", requires the change to the COLR RFB concentration limit and the MDA event during refueling be evaluated under the 10 CFR 50.59 process. This evaluation does not address the entire Cycle 28 reload, it will only address the COLR change to the Cycle 28 specific RFB concentration limit and the MDA event during refueling change.

Summary of Evaluation:

EC 69811, ANO-1 Cycle 28 PAD identified an adverse change. The adverse change is associated with the change in the RFB concentration limit reported in the COLR and the Cycle 28 reload reanalysis of the MDA event during refueling conditions based on the Cycle 28 COLR RFB concentration limit. The limit on the RFB concentration ensures the reactor remains subcritical during refueling (Mode 6). The RFB concentration limit specified in the COLR ensures an overall core reactivity of $K_{eff} \leq 0.99$ during fuel handling, with all control rods out (ARO) and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedure. The criteria for reactor protection for the MDA event during refueling is the core shall remain subcritical.

Boron, in the form of boric acid in the reactor coolant, controls excess reactivity. During refueling or maintenance operations when the reactor closure head has been removed (Mode 6), the Reactor Coolant System (RCS) boron concentration is procedurally controlled to assure a minimum Shutdown Margin (SDM) that is greater than the change in reactivity that would result from a dilution event. In these conditions, the sources of dilution water to the makeup tank and therefore to the RCS are isolated and the makeup pumps are not operating. To ensure the ability of the reactor to tolerate a moderator dilution during refueling, the consequences of accidentally filling the makeup tank with dilution water and starting the makeup pumps are evaluated. The results of this evaluation are used to demonstrate the COLR required RFB concentration limit is sufficient to prevent criticality following a dilution event.

[1] The printed name, company, department, and date must be included on the form. Signatures may be obtained via electronic processes (e.g., PCRS, ER processes), manual methods (e.g., ink signature), e-mail, or telecommunication. If using an e-mail or telecommunication, attach it to this form.

The evaluation of the dilution during a refueling accident demonstrates that the COLR required RFB concentration limit is sufficient to prevent criticality following a dilution event. This evaluation is performed for each new fuel cycle. The COLR RFB concentration limit is the boron concentration required to maintain the reactor subcritical by at least 1% Δk/k with all control rods removed from the core. A dilution event from the RFB concentration results in a reduced boron concentration. This reduced boron concentration is required to remain higher than the critical boron concentration for the refueled core configuration with the two most reactive control rods withdrawn.

The refueling evaluation assumes a conservatively small volume of RCS water will be diluted by the injection of a makeup tank full of deborated water. The volume of water assumed to be diluted corresponds to the minimum reactor vessel level allowed for maintenance activities with the fuel in the core, plus the volume of the smaller of the two decay heat removal loops (one of the loops must be in operation to allow the dilution water to mix with the vessel water). Water in the refueling canal is conservatively assumed not to be diluted. The change in concentration caused by the dilution is independent of the rate at which the dilution occurs.

Reference 3 specifies the COLR required RFB concentration limit to be used for Cycle 28 and indicates that this RFB concentration is sufficient to maintain the core subcritical by at least 1 %Δk/k with ARO. The MDA during refueling evaluation is performed for each new cycle. For Cycle 28 this evaluation, as documented in the Reference 1 Reload Report, the Reference 2 Reload Technical Document, and the Reference 3 Core Load Plan, verified that the Cycle 28 specific COLR required RFB concentration is sufficient to protect from a dilution event. As previously stated, Reference 3 indicates that the Cycle 28 COLR required RFB concentration limit is sufficient to maintain the core subcritical by at least 1 %Δk/k with ARO and also reports that the core will remain subcritical by at least 1 %Δk/k in the event of a MDA during refueling.

Throughout this evaluation, any reference to MDA analysis specifically refers to the MDA analysis during refueling conditions (Mode 6).

<u>References</u>

1. Letter FS1-0035832-2.0, “ARKANSAS NUCLEAR ONE, UNIT 1, Cycle 28 Revised Reload Report”, dated 3/2/2018 from Russell Cox to Bret Hawes.
2. Letter FS1-0035802-2.0, “ARKANSAS NUCLEAR ONE, UNIT 1, Cycle 28 Revised Reload Technical Document”, dated 3/2/2018 from Russell Cox to Bret Hawes.
3. Letter FS1-0036363-1.0, “Arkansas Nuclear One, Unit 1, Cycle 28 Core Load Plan (CLP)”, dated 2/22/2018 from Russell Cox to Bret Hawes.

Is the validity of this Evaluation dependent on any other change? ☐ **Yes** ☒ **No**

If “Yes,” list the required changes/submittals. The changes covered by this 50.59 Evaluation cannot be implemented without approval of the other identified changes (e.g., license amendment request). Establish an appropriate notification mechanism to ensure this action is completed.

Based on the results of this 50.59 Evaluation, does the proposed change require prior NRC approval? ☐ **Yes** ☒ **No**

Preparer[2]: Bret A. Hawes / see EC 69811 / Entergy / PWR Fuels / 3-3-2018
Name (print) / Signature / Company / Department / Date

Reviewer[2]: Ben Harvey / see EC 69811 / Entergy / PWR Fuels / 3-23-2018
Name (print) / Signature / Company / Department / Date

Independent Review[3]: N/A
Name (print) / Signature / Company / Department / Date

OSRC: Stephanie L. Pyle / **ORIGINAL SIGNED BY STEPHENIE L. PYLE** / 3-29-2018
Chairman's Name (print) / Signature / Date [GGNS P-33633, P-34230, & P-34420; W3 P-151]

OSRC-2018-006
OSRC Meeting #

[2] Either the Preparer or Reviewer will be a current Entergy employee.

[3] If required by Section 5.1[3].

II. 50.59 EVALUATION [10 CFR 50.59(c)(2)]

Does the proposed Change being evaluated represent a change to a method of evaluation ONLY? If "Yes," Questions 1 – 7 are not applicable; answer only Question 8. If "No," answer all questions below.

☐ Yes
☒ No

Does the proposed Change:

1. Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the SAR?

☐ Yes
☒ No

BASIS:

The Cycle 28 reload safety analysis required a Cycle 28 specific analysis of the MDA event during refueling conditions. This Cycle 28 specific MDA analysis during refueling conditions is the new reload AOR and was performed based on the change to the COLR RFB concentration limit.

The MDA event during refueling conditions relates to the Safety Analysis Report (SAR) Section 14.1.2.4 analysis. SAR Section 14.1.2.4 assumes the dilution accident occurs. The change in the Cycle 28 COLR RFB concentration limit and the Cycle 28 MDA analysis based on the Cycle 28 COLR RFB concentration limit does not impact the occurrence of the dilution accident but is relevant to the accident results. The revised MDA analysis evaluates the impact of Cycle 28 specific reload related parameters on the severity of the accident to ensure the results remain within required limits. The Cycle 28 COLR RFB concentration limit and MDA analysis do not affect the accident initiators. The Cycle 28 MDA during refueling analysis confirms the COLR required RFB concentration is sufficient to protect from a dilution event during refueling conditions. The results of the analysis verify the core remains subcritical by at least 1 %Δk/k. The change does not create any new system interactions that could cause an accident.

The change in the Cycle 28 COLR RFB concentration limit and the Cycle 28 MDA analysis during refueling based on the COLR RFB concentration limit do not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the SAR.

2. Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the SAR?

☐ Yes
☒ No

BASIS:

The Cycle 28 COLR RFB concentration limit confirms the core remains subcritical by at least 1 %Δk/k with ARO during Mode 6. The Cycle 28 MDA analysis based on the COLR RFB concentration confirms the core remains subcritical by at least 1 %Δk/k in the event of a dilution accident. Therefore, there is no increase in the probability of fuel failure. No changes to the plant equipment are required due to the Cycle 28 COLR RFB concentration limit or MDA analysis. The Cycle 28 COLR RFB concentration limit and MDA analysis do not require any equipment important to safety to be operated in a different manner or at a higher duty. The Cycle 28 COLR RFB concentration limit and the MDA analysis do not degrade the performance of any safety systems assumed to function in the safety analysis. Instrumentation accuracy and response characteristics are not impacted. The MDA analysis and COLR RFB concentration limit do not increase the probability of a malfunction of equipment important to safety.

The change in the Cycle 28 COLR RFB concentration limit and the Cycle 28 MDA analysis during refueling based on the COLR RFB concentration limit do not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the SAR.

3. Result in more than a minimal increase in the consequences of an accident previously evaluated in the SAR?

☐ Yes
☒ No

BASIS:

The COLR RFB concentration limit and MDA event during refueling conditions were analyzed for Cycle 28 using NRC approved analysis methods (BAW-10179P-A, “Safety Criteria and Methodology for Acceptable Cycle Reload Analysis”) under approved quality assurance programs. The analytical method used for Cycle 28 is the same as was used in previous cycles. The consequence of the dilution event is a decrease in shutdown margin (SDM). The Cycle 28 MDA analysis confirms that the COLR RFB concentration limit is sufficient to maintain the core subcritical by at least 1 %Δk/k in the event of a MDA during refueling conditions. There are no increases in the radiological dose consequences as no fuel failure is caused by the event.

The change in the Cycle 28 COLR RFB concentration limit and the Cycle 28 MDA analysis during refueling based on the COLR RFB concentration limit do not result in more than a minimal increase in the consequences of an accident previously evaluated in the SAR.

4. Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the SAR?

☐ Yes
☒ No

BASIS:

The COLR required RFB concentration limit was confirmed to bound the MDA event during refueling conditions for Cycle 28. This confirms the Cycle 28 core can be operated safely and can be expected to meet license requirements for accident response. The function and duty of SSCs important to safety as assumed in safety analysis are not altered. The change to the Cycle 28 COLR RFB concentration limit and the MDA during refueling analysis do not place greater reliance on any specific plant SSC to perform a safety function. No changes in the assumptions concerning equipment availability or failure modes have been made and none are necessary for the change to the Cycle 28 COLR RFB concentration limit and the MDA during refueling analysis.

The change in the Cycle 28 COLR RFB concentration limit and the Cycle 28 MDA during refueling analysis based on the COLR RFB concentration limit do not result in an increase in the consequences of a malfunction of a SSC important to safety previously evaluated in the SAR.

5. Create a possibility for an accident of a different type than any previously evaluated in the SAR? ☐ Yes ☒ No

BASIS:

The change in the COLR RFB concentration limit and the MDA during refueling analysis for Cycle 28 do not introduce any new operating conditions, plant configurations, or failure modes that could lead to an accident of a different type than any previously evaluated in the SAR. No accident initiator is affected by the change in the COLR RFB concentration limit or the Cycle 28 MDA during refueling analysis. The MDA during refueling analysis for Cycle 28 verifies the COLR required RFB concentration limit is sufficient to maintain the core subcritical by at least 1 %Δk/k in the event of a MDA during refueling conditions.

The change in the Cycle 28 COLR RFB concentration limit and the Cycle 28 MDA during refueling analysis based on the COLR RFB concentration limit do not create a possibility for an accident of a different type than any previously evaluated in the SAR.

6. Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any previously evaluated in the SAR? ☐ Yes ☒ No

BASIS:

The change in the COLR RFB concentration limit and the Cycle 28 MDA during refueling analysis do not modify the design or operation of SSCs important to safety. The COLR RFB concentration limit and the Cycle 28 MDA during refueling analysis do not require any SSC important to safety to be operated in a different manner or with a higher duty. SSCs important to safety will function in the same manner as the previous cycle. The COLR RFB concentration limit and the Cycle 28 MDA during refueling analysis do not change any parameter that would affect the function of a SSC important to safety. The COLR RFB concentration limit and the Cycle 28 MDA during refueling analysis do not assume any changes in the failure modes of equipment important to safety.

The change in the Cycle 28 COLR RFB concentration limit and the Cycle 28 MDA during refueling analysis based on the COLR RFB concentration limit do not create a possibility for a malfunction of a SSC important to safety with a different result than any previously evaluated in the SAR.

7. Result in a design basis limit for a fission product barrier as described in the SAR being exceeded or altered? ☐ Yes ☒ No

BASIS:

The MDA during refueling analysis is part of the reload safety analyses for Cycle 28 that are performed to demonstrate compliance with design basis limits for fuel cladding, RCS pressure boundary, and containment fission product barriers. The Cycle 28 COLR RFB concentration limit was confirmed to maintain the core subcritical by at least 1%Δk/k in the event of a moderator dilution accident during refueling conditions. Therefore, the COLR RFB concentration limit and the Cycle 28 MDA during refueling analysis do not affect the ability of the fuel cladding to maintain its integrity as a fission product barrier.

The change in the Cycle 28 COLR RFB concentration limit and the Cycle 28 MDA during refueling analysis based on the COLR RFB concentration limit do not result in a design basis limit for a fission product barrier as described in the SAR being exceeded or altered.

8. Result in a departure from a method of evaluation described in the SAR used in establishing the design bases or in the safety analyses?

☐ Yes
☒ No

BASIS:

The COLR was changed to reflect the Cycle 28 specific RFB concentration limit. The Cycle 28 reload safety analysis required a Cycle 28 specific analysis of the MDA event during refueling conditions. This Cycle 28 specific MDA during refueling analysis is the new AOR. The MDA during refueling analysis evaluates the impact of Cycle 28 specific reload related parameters on the severity of the accident to ensure the results remain within required limits. Both the RFB concentration and the MDA during refueling analysis use the same NRC approved method (BAW-10179P-A) of evaluation as previous cycles under an approved quality assurance program. The methods are described in SAR Section 14.1.2.4.3. No new methods were required to calculate the COLR RFB concentration or for the MDA during refueling analysis.

The change in the Cycle 28 COLR RFB concentration limit and the Cycle 28 MDA during refueling analysis based on the COLR RFB concentration limit do not result in a departure from a method of evaluation described in the SAR used in establishing the design bases or in the safety analyses.

If any of the above questions is checked "Yes," obtain NRC approval prior to implementing the change by initiating a change to the Operating License in accordance with NMM Procedure EN-LI-103.

Attachment 5 to

1CAN111802

ANO-1 and ANO-2 Commitment Change Summary Report

ANO-1 and ANO-2 Commitment Change Summary Report

Number	Commitment Date	Changed Date	Short Title	Original Commitment	Justification of Change
18448 / 18449	10/04/2005	06/30/2017	Containment Sump Performance	Entergy will ensure that as part of the modification process, insulation materials that are introduced to containment are identified and evaluated to determine if they could affect sump performance or lead to downstream equipment degradation.	These commitments are closed since they have been incorporated in ANO processes for over 10 years and are now being incorporated into industry standard design processes. The nuclear industry has adopted industry procedure IP-ENG 007 for performing engineering modifications per the standard design process. Entergy procedure EN-DC-775 Rev. 21 endorses the use of the new industry procedure for the standard design process, IP-ENG-007, for Entergy.
18852	12/02/2008	03/20/2018	Communications Security	Implement procedures that describe where and when the Privatel devices can be used, how the identity and access authorization of the Privatel users will be verified, how to confirm the Privatel device is providing a secure conversation, and actions to be taken if the security or encoding of the conversation is suspected to be lost or compromised.	The Privatel device is no longer in use at any Entergy site. Entergy is canceling the devices' implementing procedure, EN-NS-2018 because the National Institute of Standards and Technology (NIST) no longer allows for its use. In the interim, Entergy has opted to not allow safeguards information discussions via any phone system until such a time that a new NIST-approved device is devised. EN-NS-204 is currently undergoing a revision to remove all reference to EN-NS-2018 due to the above. This commitment is not going to be implemented in any fleet or site procedure and, therefore, is deleted.
17917	12/02/2003	05/16/2018	Aging Management	Maintain the Fire Water System Program	Rather than manage selective leaching through specific component inspections as outlined in the Fire Water System Program, loss of material due to selective leaching will be managed by the Selective Leaching Program per commitment P-20017. The program is described in new Safety Analysis Report (SAR) Section 18.1.35.

Number	Commitment Date	Changed Date	Short Title	Original Commitment	Justification of Change
17925 / 20085	12/02/2003	05/16/2018	Aging Management	Modify and maintain the Periodic Surveillance and Preventive Maintenance (PSPM) Program	Rather than manage selective leaching through specific component inspections as outlined in the PSPM Program, loss of material due to selective leaching will be managed by the Selective Leaching Program. The program is described in new SAR Section 18.1.35. Both fouling and loss of material are adequately managed by the Service Water (SW) integrity program and the oil analysis programs, so further inspection under the PSPM program for 2P-89A, 2P-89B, and 2P-89C are not required to manage aging effects of the High Pressure Safety Injection pump bearing cooling units. During development of a repetitive activity for the Emergency Diesel Generator (EDG) and Alternate AC Diesel Generator (AACDG) expansion joints to perform nondestructive examination (NDE) ultrasonic thickness (UT) readings on the expansion joints, it was determined that UT readings of the metal expansion joints was not possible based on the closeness of the convolutions and size of the joints. Based on the inability to perform reliable, repeatable UT on the expansion joints, visual examination of the external surfaces of the expansion joints will be performed in accordance with the PSPM program frequency. Dye penetrant testing will be performed if defects are identified. Expansion joints are examined concurrently with other related EDG inspections, and the frequency of inspection for the expansion joints is in accordance with the PSPM program. The 2C-7 Atlas COPCO model LT-20-30 twin cylinder reciprocating starting air unit and the 2M-10 heatless regenerative desiccant dryer system were replaced with an air compressor/dryer system which utilizes a Sauer model WP65L compressor and air products membrane dehydrator. An air dryer with dew point measurement is not available on the new unit. The new unit is equivalent to the existing compressor/dryer (2C-7A). Preventative maintenance (PM) is performed on each unit to ensure significant moisture is not entrained in the system; however, dew point on the AACDG starting air dryer will not be monitored.

Number	Commitment Date	Changed Date	Short Title	Original Commitment	Justification of Change
17929	12/02/2003	05/16/2018	Aging Management	Maintain the Reactor Vessel Internals (RVI) Cast Austenitic Stainless Steel (CASS) Program	The only RVI CASS component is the control element assembly shroud tube. The reactor vessel internals stainless steel plates, forgings, welds and bolting program per MRP-227-A specifically addresses RVI components fabricated from CASS, martensitic stainless steel, or precipitation hardened stainless steel materials to ensure their functionality is maintained during the period of extended operation considering the potential loss of fracture toughness due to thermal and irradiation embrittlement. Consequently, the specific commitment as outlined in the license renewal application (LRA) for RVI CASS is no longer necessary and is deleted.
17931	12/02/2003	05/16/2018	Aging Management	Maintain the SW Integrity Program	Rather than manage selective leaching through specific component inspections as outlined in the SW Integrity Program, loss of material due to selective leaching will be managed by the Selective Leaching Program per commitment P-20017. The program is described in new SAR Section 18.1.35.
17932	12/02/2003	05/16/2018	Aging Management	Maintain the Steam Generator (SG) Integrity Program	The visual inspection of the SG lower internals is intended to quantify sludge deposition, identify and remove loose parts, and assess corrosion or damage in the accessible regions of the lower tube bundle. During this inspection, the specific components listed in letter 2CAN070404, request for additional information (RAI) responses for LRA, dated July 1, 2004, RAI 3.1.2.5-1 (anti-vibration bar end caps, U-bend peripheral retaining ring, U-shaped retainer bars, stay rods, stay rod hex nuts, spacer pipes, peripheral backup bars, wrapper, and wrapper jacking screws) are not visually inspected. Inspection of these components is not required by the SG vendor manual, NEI 97-06, *Steam Generator Program Guidelines*, or the Electric Power Research Institute, *Steam Generator Management Program Guidelines*.

Number	Commitment Date	Changed Date	Short Title	Original Commitment	Justification of Change
17936	12/02/2003	05/16/2018	Aging Management	Maintain the Wall Thinning Monitoring Program	As part of the Wall Thinning Monitoring Program, specific activity details require revision as follows. During development of a repetitive activity to perform NDE UT readings on the expansion joints, it was determined that UT readings of the metal expansion joints was not possible based on the closeness of the convolutions and size of the joints. Based on the inability to perform reliable, repeatable UT on the expansion joints, visual examination of the external surfaces of the expansion joints will be performed in accordance with the PSPM program frequency. There is a provision to perform dye penetrant testing if defects are identified.
17940	12/02/2003	05/16/2018	Aging Management	Implement Environmentally Assisted Fatigue Option Program	The change clarifies that the stainless steel charging nozzle and safety injection nozzle usage factors with environmental correction factors are 12.012 and 5.782, respectively.
18175	10/18/2004	05/16/2018	Aging Management	Perform a one-time inspection of selected 10 CFR 54.4(a)(2) components that will determine whether degradation, as a result of loss of intended function of the components, will be maintained during the extended period of operation (RAI-3.3.2.4.1 1-1).	Per letter 0CNA080005, dated August 17, 2000, *Elimination of PASS Requirements*, the NRC issued Amendment No. 218 to facility operating license NPF-6 for ANO-2. The amendment consisted of changes to the ANO-2 technical specifications, deleting requirements to maintain PASS. Subsequent to NRC approval for PASS elimination, PASS components were isolated; therefore, inspections of PASS system components are not performed.

Number	Commitment Date	Changed Date	Short Title	Original Commitment	Justification of Change
17927	12/02/2003	06/27/2018	Aging Management	Maintain the Reactor Vessel Head (RVH) Penetration Program	RVH Penetration Program (ANO-2 LRA, 2CAN100302, dated October 14, 2003, Appendix B, Section B.1.20) outlines requirements consistent with NRC Order EA-03-009, *Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors*. This commitment was deleted by letter 2CAN041801. Subsequent review has determined that it would have been more appropriate to clarify the commitment rather than delete it; therefore, the commitment is being reinstated as clarified below. Clarification: The ANO-2 RVH Penetration Program was based on NRC Order EA-03- 009. Since program inception, the NRC has promulgated 10 CFR 50.55a, introducing a rule that all pressurized water reactor licensees include the requirements of American Society of Mechanical Engineers Code Case N-729, *Alternative Examination Requirements for PWR Vessel Upper Heads with Nozzles having Pressure-Retaining Partial-Penetration Welds*, in the Inservice Inspection (ISI) Program. Entergy has augmented the ISI program with N-729 requirements as required by 10 CFR 50.55a(g)(6)(ii)(D)(1) through (4), thereby superseding the requirements of EA-03-009. Consequently, since the inspections required by the RVH Penetration Program have been superseded by 10 CFR 50.55a, the specific commitment as outlined in the LRA is being clarified to meet the ASME Code Case N-729 instead of the NRC Order EA-03-009.

Number	Commitment Date	Changed Date	Short Title	Original Commitment	Justification of Change
18889	05/01/2009	09/17/2018	Inservice Testing	Perform a sample test plan leak rate on one of the two valves each refueling outage on a rotating basis (2CV-1541-1 and 2CV-1560-2 ECP returns). If leak rate test fails, both valves must be tested.	The subject commitment was related to a relief request extending the frequency of testing from every 2 years to every 3 years (to match refueling outage frequency) where one valve is tested each refueling outage. The Operations and Maintenance code dictates required testing, and code requirements are captured in the ANO Inservice Test (IST) program; therefore, it is not necessary to track the test itself in the commitment management system (CMS) (i.e., if testing was not performed consistent with the correspondence, the default would be to go back to the two-year frequency). Code requirements also dictate test expansion upon failures. Since there is only two valves in this particular group, any expansion would automatically require testing of the redundant valve. Because the IST program is required to capture code requirements and be maintained up to date, it is not necessary to track this commitment in CMS. In accordance with NEI 99-04, *Guidelines for Managing NRC Commitment Changes*, it is not necessary to duplicate tracking of commitments:

Number	Commitment Date	Changed Date	Short Title	Original Commitment	Justification of Change
18833 / 18834	09/17/2008	10/01/2018		Entergy committed to the measurement of latent debris quantities every third refueling outage to confirm that latent debris quantities used in strainer testing and downstream effects analysis remain bounding. If subsequent inspections reveal that housekeeping and cleanliness measures continue to maintain latent debris loading below the tested/evaluated values with sufficient margin, then the inspection frequency could be extended to a maximum interval of every sixth outage (not to exceed ten years). If inspection results reveal an adverse trend in latent debris quantities such that latent debris margin for the tested and analyzed conditions are unacceptably reduced, then the inspection frequency will be shortened and the scope increased as appropriate to ensure adequate margin is maintained.	This commitment is closed as the sampling frequencies have been completed with satisfactory results and the current frequency has moved out to every 6th refueling outage as permitted by the commitment. The program also has steps to ensure the frequency is reduced in the future if results become unsatisfactory (150 lbs). CALC-ANO1-ME-09-00005, *ANO-1 Latent Debris Determination*, documents the results of the latest latent debris survey for ANO-1 that was performed in 1R23. The latent debris quantity from CALC-ANOI-ME-09-00005 is subsequently documented in CALC-ANO1-ME-09-00003, *ANO-1 Ctmt Sump Debris Margins*. CALC-ANO1- ME-09-00003 provides the programmatic guidance for adjusting the latent debris survey interval based upon the survey results. Similarly for ANO-2, CALC-ANO2-ME-09-00003, *ANO-2 Latent Debris Determination*, documents the results of the latest debris survey for ANO-2 that was performed in 2R23. The latent debris quantity from CALC-ANO2-ME-09-00003 is subsequently documented in CALC-ANO2-ME-09-00004, *ANO-2 Ctmt Sump Debris Margins*. CALC-ANO2-ME-09-00004 provides the programmatic guidance for adjusting the latent debris survey interval based upon the survey results. Because this analysis has been in place for nearly 10 years and proper controls are well established, it is no longer necessary to track the performance of this analysis via CMS.
19794	06/28/2016	11/07/2018	95003 Confirmatory Action Letter (CAL)	PM-9 Develop Metrics for the Number of Open Craft Work Order (WO) Feedback Requests	EN-WM-105, Section 5.9[3], requires that PM WO feedback be monitored and incorporated within 90 days, or evaluated and the PM model WO placed in the “plan” status within 90 days with a hold pending incorporation of the PM feedback. Therefore, there is no need to maintain a metric for open Craft WO Feedback Requests that are greater than 90 days of age. This commitment is retired.

Attachment 6 to

1CAN111802

List of Affected SAR Pages

List of Affected SAR Pages

The following is a list of Safety Analysis Report (SAR) pages revised in Amendment 28 to support corrections, modifications, implementation of licensing basis changes, etc., as described in the Table of Contents of each SAR chapter (reference Enclosure 1 of this letter). Information relocated from one page to another in support of the aforementioned revisions is not considered a change; therefore, these pages are not included in the following list. In addition, pages associated with the individual Table of Contents are not listed below as related revisions are administrative only changes.

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