May 11, 2015

MEMORANDUM TO: John Segala, Chief
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Division of New Reactor Licensing
Office of New Reactors

FROM: William (Billy) Gleaves, Sr. Project Manager ***/RA/***
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SUBJECT: AUDIT REPORT FOR REVIEW SELECTED AREAS OF U.S. EPR DESIGN CERTIFICATION RELATED TO INCORE TRIP SETPOINT AND TRANSIENT METHODOLOGY TOPICAL REPORT ANP-10287P

By letter dated December 11, 2007, as supplemented by letters dated February 7, 2008, and February 20, 2008, AREVA NP Inc. (AREVA), submitted an application to the U.S. Nuclear Regulatory Commission (NRC) for a standard design certification (DC) of the U.S. EPR, pursuant to Title 10 of the *Code of Federal Regulation* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." As part of that application, AREVA has submitted Topical Report (TR) ANP-10287P, "Incore Trip Setpoint and Transient Methodology for U.S. Evolutionary Power Reactor," Revision 2, (a public version is available in the Agencywide Documents and Management System (ADAMS) at Accession No. ML14220A191) addressing, in part, accident scenarios such as undetected self-powered neutron detector (SPND) single-failures, referenced reactor volumetric flow rates, and departure from nucleate boiling ratio (DNBR) algorithms for three-loop reactor coolant system (RCS) operation.

As part of the continuing technical review of the application, the NRC staff conducted an audit of this TR. The on-site audit was conducted at the AREVA NP, Inc., office located in Lynchburg, VA on November 3-6, 2014, by three NRC staff and an NRC consultant. The audit team reviewed the documents related to the determination of the reference volumetric flow rate, DNBR algorithm regarding the effects of reduced coolant flow, reverse flow, and penalties associated with non-uniform flow when one of the four RCS pumps is tripped or in a reduced flow capacity during plant operations and set point adjustments associated with an undetected single failure of an SPND.

Docket No.: 52-020

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The audit team identified no findings. Enclosed is the detailed audit report documenting the audit activities.

Enclosure: As stated

cc: See next page

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Enclosure: As stated

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AUDIT REPORT

1.0 INTRODUCTION

A 4 day audit was conducted by the U.S. Nuclear Regulatory Commission (NRC) staff at the AREVA office in Lynchburg, VA on November 3-6, 2014, with the primary focus on the treatment of an undetected single failure of a self-powered neutron detector (SPND) during a transient event as described in Request for Additional Information (RAI) 505 Question 07.01-33 and three loop reactor coolant system (RCS) operation described in RAI 544, Questions 04.04-67 and 04.04-68 regarding the determination of the reference volumetric flow rate and departure from nucleate boiling ratio (DNBR) algorithm relating to the effects of reduced coolant flow, reverse flow, and penalties associated with non-uniform flow.

This audit included review of 20 condition reports (CR), a root-cause analysis report, calculation summary sheets (CSS), design report (DR), engineering information records (EIR), and miscellaneous documents related to the model and calculations to address the concerns related to SPND failure and three loop RCS operation.

AUDIT MEMBERS

Project Team

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Technical Team

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Jose March-Leuba (Oak Ridge National Laboratory – Consultant)

2.0 DETERMINATION OF THE REFERENCE VOLUMETRIC FLOW RATE AND DEPARTURE FROM NUCLEATE BOILING RATIO (DNBR) ALGORITHM FOR THREE LOOP RCS OPERATION

2.1. INTRODUCTION

Based on RAI 544, Questions 04.04-67 and 04.04-68 regarding three loop RCS operation, the NRC audit team reviewed the documents related to the determination of the reference volumetric flow rate (Qref) and DNBR algorithm regarding the effects of reduced coolant flow, reverse flow, and penalties associated with non-uniform flow when one of the four RCS pumps is tripped or in a reduced flow capacity during plant operations.

2.2. BACKGROUND

Reference: March 15, 2012, letter from J. Sam Armijo, Chairman, Advisory Committee on Reactor Safeguards (ACRS), "Chapters 6, 7, 11, 13, 15, 16, and 18 of the Safety Evaluation Report (SER) with Open Items associated with the U.S. Evolutionary Power Reactor Design Certification Application."

2.3. RAI 544, Question 04.04-67

The response to RAI 16 on ANP-10287, "Incore Trip Setpoint and Transient Methodology for U.S. EPR," states that when a global three loop flow signal is received, the DNBR algorithm replaces the reference volumetric flow rate with a constant lower volumetric flow rate consistent with three pump operation. Does this lower flow rate capture the effects of reduced coolant flow, reverse flow, and penalties associated with non-uniform flow?

2.4. RAI 544, Question 04.04-68

The transient in Section 15.4.4, "Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature" does not capture all potential operating points described in technical specification (TS) 3.4.4. Provide an analysis demonstrating DNBR margin is maintained for the most limiting operation allowed by this TS, including operation just below the P3 permissive. This analysis should justify the statement in Final Safety Analysis Report (FSAR) Section 7.2.1.3.2 that the P3 permissive setpoint value corresponds to the value below which loss of one reactor coolant pump does not lead to risk of departure from nucleate boiling (DNB).

2.5. Documents Reviewed During Audit

The following documents were reviewed during the audit (electronic reading Room (ERR) and on-site):

2.5.1 Thermal-Hydraulic Assessment of RAI No. 544 (32-7013658-000)

- Overview and assessment of the DNBR algorithm for full/reduced RCS flow.
    - Effects of reduced coolant flow (three pump operation) as well as reduced coolant back flow (reverse flow) in the idle loop is identified through a conservative low volumetric flow rate (Qref) in ANP-10287P-002, "Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report."

- Confirms that the incore trip, DNBR algorithm, conservatively includes the effect of non-uniform flow, under full flow operation through the trip uncertainty terms.
- Confirms that the effects due to reduce flow, non-uniform flow, and reduced flow beyond those for full flow operation, such as for partial pump operation are included in the "reference flow rate" (Qref) term of the trip.

2.5.2 Juliette Tests Interpretation Flow Rate Distribution and Characterization of the Head Losses and Pressure Fields (38-9020405-000)

- Determination of core flow distribution in a non-uniform flow regime.
  - Juliette test facility is approximately 1/5th scale model for simulating the reactor core, inlet nozzle, core baffle, and inlet region.
  - Hydraulic testing was performed to determine the core flow distribution in a non-uniform flow regime.
  - Hydraulic testing was performed with three and four pump flow configuration under various flow rates to determine head losses and pressure fields including two tests to simulate three pump operation and the effect of reverse flow (versus no flow) through the idle pump.

2.5.3 U.S. EPR RCS Hydraulic Analysis (32-9011635-002)

- Reduced coolant flow and reverse flow analysis.
  - Three pump system analysis was benchmarked against a detailed plant hydraulic analysis discussed in the document.
  - Basis of the reversed flow through the idle pump was based upon head loss specification data for the actual pumps.
  - Resulting values from the analysis were { } gallons per minute (gpm) for each operating pump for a total flow rate of { } gpm for three pumps including a { } gpm reverse flow through the idle loop resulting in a total core flow rate of { } gpm.
  - Forward flow rates were reduced in the system analysis for typical design considerations by ~ 4% design factor between nominal/best-estimate flow and thermal design flow (U.S. EPR FSAR Tier 2, Table 4.1-1).

2.5.4 Core inlet flow factors – U.S. EPR (32-9031524-000)

- Determination of the core inlet flow factors to the thermal-hydraulic sub-channel model.

  From the scaled model hydraulic testing results, the inlet flow distribution data were divided into two flow zones based on observation of the effect of the Flow Diversion Device (FDD) on the inlet flow profile.

  - Average inlet flow profile for both 1/4 and 1/8 core configurations were calculated.

- Core maximum and minimum inlet flow factors for both four pump and partial pump operation were calculated for each zone.
- Calculated the relative inlet velocity flow profile and the bounding (95/95) lower tolerance factor on the minimum flow for the limiting fuel bundle.
- Incorporated the non-uniform and partial pump operation reverse flow effects.

2.5.5 Bias Curves and DBR Algorithm Uncertainty (32-9032134-001, -003)

- Generation of the trip uncertainty terms.
  - Determined the updated bias curves used to adjust the algorithm local thermal-hydraulic parameters (channel enthalpy and inlet mass flux rate).
  - Determine a conservative statistical description of the algorithm uncertainty on Minimum Departure from Nucleate Boiling Ratio (MDNBR) and quality calculations.

2.5.6 U.S. EPR Safety Related Core I&C Functional Requirements (51-9095690-001)
U.S. EPR System Description Protection System (PS) (15-9025647-002, -003)

- Reference Flow Rate (Q_{ref}) in system description and functional requirements.

2.5.7 Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report (TR) (43-10287P-002)

2.6. TECHNICAL EVALUATION

The audit team reviewed the bases for the { } value used for three-loop flow. The calculation of this flow follows the procedure below:

- The pump homologous curves were available from the manufacturer. The reverse (leakage) flow estimated for a stopped pump is { } based on the pump curves.
- In addition to subtracting the reverse flow, the following conservative factors were added to the total core flow:
  - { } is used as a standard design factor to cover possible future changes
  - A { } is used to account for flow asymmetry in the vessel with only three out of four pumps operating

The assessment (the base uncertainties in addition to the volumetric flow rate penalty) consisted of performing detailed sub-channel DNB calculations with the LYNXT solver (BAW-10156PA, Revision 1, "LYNXT-Core Transient Thermal-Hydraulic Program") over a variety of conditions (core exit pressure, difference in DNB predictions between the detailed (LYNXT) and simplified (ANP-1087 DNB algorithm) analysis). From the results of the DNB predictions, the correction factors could be determined.

Based on the above information, the audit team reviewed the calculations to support operation with only three loops. The following actions are taken when the plant is in three loop operation:

- Power is reduced <60 percent of full power
- Hot channel flow input is { . }. This is the flow that the DNBR algorithm assumes for the DNBR calculation.
- The parameter transfer is a software switch based on measurements of pump status.

2.7. STAFF OBSERVATIONS

The audit team concludes that three-loop operation is very unlikely, or even unachievable because of the multiple trips that could be activated to shut down the reactor. Even if steady state operation is achieved with three loops, operation is only allowed for a maximum of 2 hours during which time an adjustment to the volumetric flow rate flow ({ }) is applied to the online DNBR algorithm and, along with non-uniform effects captured through the uncertainty terms, provides sufficient conservatism to the DNBR algorithm in three-loop operation. Therefore, the audit team concludes that AREVA adequately addressed the Q_{ref} and algorithm development for three RCS loop operation.

3.0 TREATMENT OF AN UNDETECTED SPND FAILURE (APPENDIX H)

3.1. INTRODUCTION

Appendix H of ANP-10287P describes the methodology to account for an undetected single failure of an SPND. The standard methodology described elsewhere in ANP-10287P accounts for SPND failures that have already been detected by either the operator manually or the protection system automatically.

An undetected SPND failure is one where the detector output is frozen in place at a "reasonable" value (e.g., not zero or max voltage) so that during normal operation, the detector appears to function; but when a transient occurs, the detector value remains frozen and is, thus unresponsive.

The treatment of an undetected single failure of an SPND is described in detail in Appendix H. The methodology is complex because of the large number of special cases that must be considered. However, underlying all this complexity is a simple approach: { }.

3.2. TECHNICAL EVALUATION

The U.S. EPR Setpoint Methodology addresses the presence of undetected SPND failures by { }. The details of implementation depend on the type of setpoint and can be quite complex, but in all cases, the implementation uses a conservative approach instead of a statistical 95/95 criterion to determine the undetected failed SPND { }.

The static setpoint methodology follows a Monte Carlo approach for detected SPND failures. A large number ({ }) of cycle specific power shapes are used to generate simulated SPND readings. Up to five SPND failures (which are assumed to have been detected by the protection system) are assumed to generate the setpoint sets. The failures are sampled randomly ({ }), with failure rules enforced to satisfy technical specifications. The setpoint sets are selected so that a significant number of the sampled cases satisfy the specific criterion (DNBR, exit quality, linear power density (LPD), or imbalance, both for trips and limiting conditions for operations (LCOs) with a 95% probability and 95% confidence. For each of these random samples, { }.

The staff noted that for symmetric cases, the protection system hardware is designed to ignore the most responsive SPND and scram on the second most responsive. This feature is intended to minimize the probability of spurious scrams. For those symmetric cases, the static setpoint methodology { }.

The transient setpoint methodology uses a simpler approach. { } { }. No distinction is made between undetected or undetected failures. In all cases, one undetected failure is always assumed, and different setpoints are generated for 0, 1, 2, 3, 4, or 5 detected failures. For example, for 3 detected failures, { }.

The NRC staff also noted that the standard transient procedure involves identifying the most sensitive SPND for each type of trip (e.g. LPD or DNBR), and the most sensitive SPNDs may be different for each trip type. This procedure allows for a more complex calculation where the same SPNDs are assumed failed for all trip types. { }. This second more complex calculation is only envisioned when the simpler setpoints are judged too limiting for reactor operation.

3.3. STAFF OBSERVATIONS

Based on the audit on the regulatory bases described in the Audit Plan, Section C, and regulatory guidance (RG) identified in the reference section of this report, the evaluation of the methodology described in the ANP-10287 Revision 2 and supporting documents provided in the electronic reading room and during the site audit, the staff observes that:

1. { } is a conservative approach to account for the presence of undetected SPND failures.
2. The proposed Monte Carlo implementation for static setpoints { } { } is conservative both for symmetric and asymmetric cases.
3. The proposed transient setpoint methodology, where the { } { } is conservative.
4. The proposed refinement to the transient methodology, where the failed SPNDs are { } is reasonable.

4.0 APPLICABILITY OF CURRENT FSAR RESULTS TO FUTURE CYCLES, INCLUDING THE EQUILIBRIUM CYCLE

4.1. TECHNICAL EVALUATION

The analysis supporting the incore setpoint methodology is performed in three steps:

1. First, the static methodology (STATICS) is applied to select the setpoint.
2. Engineering judgment is used to round up the setpoints to ensure that they will be valid for future cycles and for transient conditions.
3. The transient analysis methodology (TRICS) is used to confirm the setpoints. If TRICS requires an adjustment, the setpoints are updated.

The static and transient analyses for low DNBR and high LPD utilize two major inputs:

1. A series of core power shapes ({ }) that represent the potential core power distributions that can be reached during normal operation for the current cycle.
2. A series of system analyses that provide boundary conditions (power, flow, temperature, etc.)

Uncertainties are accounted for in the methodology, including algorithm and instrument uncertainties.

The core power shapes are generated using the methodology and ensure that they bound the maximum allowed peaking factors, including large axial power imbalance and large radial peaks that are expected to bound the maximum operational transients.

These power shapes are calculated on a cycle specific basis to confirm the setpoints for every future cycle. However, the cycle-to-cycle variations (enrichment burnable poison, 18 or 24 month cycle) have a small impact on the resulting power peaking. The largest impact on peaking factors is typically caused by the off-normal situations assumed like Xenon transients and rod bank misalignments.

Cycle-to-cycle differences would affect kinetic parameters for the TRICS; thus is not clear a priori that these analyses would be bounding for the life of the plant. To achieve the desired level of conservatism over the plant lifetime, the transient methodology documented in the FSAR Chapter 15 ensures that the system level responses are bounding by imposing conservative reactor kinetic inputs (e.g., beta-effective) and conservative performance characteristics of safety system actuation and operation. These assumptions are documented in Sections 15.0.6 and 15.0.11. The NRC staff reviewed these added conservatisms and concluded that they are likely to ensure applicability of the TRICS results for the plant lifetime.

The transient analysis uses the conservative power shape set in a manner that is mostly independent of the cycle-specific designs. The system boundary conditions, such as power, flow, and temperature, are superimposed on the conservative power shape set to generate the 3D transient power and the SPND response during the transient (including SPND failures).

After evaluation of the methodology, the NRC staff observes that the U.S. EPR setpoint methodology uses conservative power shapes for both static and dynamic thresholds, and no

single shape is more or less appropriate. The complete set of shapes is used in a statistical 95/95 treatment to obtain setpoints and their applicability is confirmed and updated based on the transient analyses result. This evaluation will be performed for each reload cycle, and the setpoints will either be confirmed or updated; but changes from cycle-to-cycle are not expected to be significant, because the main driver of the calculation is the superimposed off-normal conditions like Xenon transients or extreme rod patterns.

4.2. STAFF OBSERVATIONS

Based on this evaluation, the regulatory basis identified in Subsection 8.C of the audit plan, and the regulatory guidance described in the Standard Review Plans of Chapters 4, 7, and 15, the NRC staff observes the first cycle application of the U.S. EPR setpoint methodology described in the FSAR is representative of future cycles, including a hypothetical equilibrium cycle. However, the actual cycle-by-cycle core design is expected to result in changes in the set points.

5.0 CONDITION REPORTS AUDITED

AREVA U.S. launched the effort to respond to the NRC RAI 505, Question 07.01.03 regarding the undetected SPND single failure. In the process of performing the new analyses, AREVA U.S. identified more than 20 deficiencies of their previous analyses supporting the submittal of the original TR of ANP-10287P “In-core Set Point Methodology.” As the result, they issued 20 Condition Reports (CRs). Based on these 20 CRs, an Apparent Cause Evaluation Report was prepared to identify the large number of condition reports in a concentrated engineering design and analysis area. The NRC staff audited these 20 CRs and the apparent cause evaluation report.

These 20 CRs can be categorized into five categories.

1. STATICS coding/input errors (4 CRs)
2. TRICS coding/input errors (6 CRs)
3. TR errors (4 CRs)
4. Chapter 15 FSAR errors/issues (4 CRs)
5. Control Rod Misoperation Analysis errors (2 CRs)

The NRC staff reviewed these 20 CRs and noted that the error fixes had been incorporated into the relevant FSAR and TR sections. Based on these error fixes and the improvement of the method considering undetected single failure, AREVA updated the TR in Revision 2 and submitted the latest version to NRC in August of 2014.

6.0 ELIMINATION OF THE TESTING METHOD METHODOLOGY

Based on a common sampling methodology, the original version of ANP-10287 proposed to use two methods for establishing and evaluating the variations of in-core setpoints. Setpoints were calculated using a statistical methodology prior to each fuel cycle. The first method was the coverage method applied independently to each separate trip, limitation, and LCO function. The coverage method requires no predetermined setpoints. The second method was the iterative testing method, which required either knowledge of the in-core setpoints or an estimate of the in-core setpoints. The iterative testing method was intended to be used as a refinement to produce less restrictive setpoints, but required additional complex calculations.

During the comprehensive review of the methodology that resulted in Revision 2 of ANP-10287, the applicant found some inconsistencies in the application of the iterative testing method. Apparently, it was being applied with different assumptions in different sections in the report. Since the iterative testing method was not used to generate the setpoints or results presented in the FSAR, the applicant decided to eliminate the iterative testing method as an option in the methodology in Revision 2 of ANP-10287. Consequently, Revision 2 has deleted Section 5.5, which used to describe this method. The NRC staff notes that the removal of Section 5.5 and the iterative testing method results in more conservative setpoints.

7.0 CONCLUSIONS

The NRC staff observed that the information and calculations provided by AREVA properly supported the applicant's revision of ANP-10287P to include Appendix H that addressed RAI 505, Question 07.01-33 regarding the revised set points that compensate for an undetected single failure of a SPND detector. In addition, AREVA provided adequate information to address the staff concerns described in RAI 544, Questions 04.04-67 and 04.04-68 regarding the determination of the reference volumetric flow rate and DNBR algorithm relating to the effects of reduced coolant flow, reverse flow, and penalties associated with non-uniform flow when one of the four RCS pumps is tripped or in a reduced flow capacity during plant operations. No technical findings were identified.

APPENDIX I

LIST OF DOCUMENTS AUDITED

The following documents were audited by the NRC staff:

Group 1 Engineering Calculation Reports

1. STATICS V2.0 – STATIC SPND Setpoints Program for (32-9050676-001)
2. U.S EPR DC Single Undetected Failed SPND: Impact on US EPR Design (51-7014125-000)
3. U.S EPR. Evaluation of Incore STATIC Setpoints (32-9061134-001)
4. U.S EPR. Input Summary for Calculations Using STATICS 2.0 (51-7013091-000)
5. TRICS Theory and User's Manual (51-9059741-002)
6. Verified Computer Code TRICS (32-9107307-001)
7. U.S. EPR Input Summary for Calculations Using TRICS 2.0
8. Thermal-Hydraulic Assessment Of Request For Addition Information RAI NO 544 (32-7013658-000)
9. U.S. EPR Representative DNBR and FCM Plots for Chapter 15 FSAR RAIS (32-9086230-001)
10. U.S. EPR DNB and FCM Evaluation OPF SRP 15.1 Events
11. { } (32-9058055-002)
12. U.S. EPR DCD: TH Performance of Single Rod Drop Event (SRP 15.4.3)
13. { } (32-701514-000)
14. { } (32-9093769-001)
15. { } (32-9090643-001)
16. { } (32-7006163-001)
17. STATIC Setpoint Methodology (51-9050656-000)

18. Bias Curves and DNBR Algorithm Uncertainty (32-9032134-001)

19. EPR System Description Protection System (15-9025647-002)

20. EPR System Description Document (15-9025647-003)

21. EPR Core I&C Functional Requirements for System Design Description (SDD) Input

Group 2 Conditional Reports

1. CR 2013-1429 “{ }
2. CR 2013-82 “{ } { }
3. CR 2013-3389 “{ } { }
4. CR 2013-3305 “{ } { }
5. CR 2013-2056 “{ }
6. CR 2013-5057 “{ } { }
7. CR 2013-1913 “{ } { }
8. CR-2013-2819 “{ }
9. CR-2013-1967 “{ }
10. CR-2013-1435 “{ }
11. CR-2013-1349 “{ } { }
12. CR-2012-9698 “{ }
13. CR-2013-1432
14. CR-2013-2152 “{ }
15. CR-2013-1433 “{ }
16. CR-2014-5614 “CR Process” 10/07/2014
17. CR-2013-1431 “{ }
18. CR-2013-1430 “{ }

Group 3 Calculation Summary Sheets

1. CSS 32-9061134-001 “US EPR: Evaluation of Incore Static Setpoints”
2. CSS 32-9050676-001 “STATICS v2.0-Static SPND Setpoints Program for EPR”
3. CSS 32-9061804-002 “US EPR: DNB and FCM Evaluation of SRP 15.1 Events”
4. CSS 32-70150-000 “{ }”
5. CSS 32-9038020-004, 005 "{ }
6. 32-9058055-002 “{ }”
7. CSS 32-9090643-001 “{ }”
8. CSS 32-9093769-001 “{ } { }
9. CSS 32-7013658-000 “Thermal-Hydraulic Assessment of Request for Additional Information (RAI) No. 544”
10. CSS 32-9032134-001, -003 “Bias Curves and DNBR Algorithm Uncertainty”
11. CSS 32-9031524-000 “Core Inlet Flow Factors- U.S. EPR”
12. CSS 32-7004271-001 “{ } { }
13. CSS 32-7006163-001 “{ }”
14. CSS 32-9051893-001 “EPR DC - Large Variation SPNDRES Sensitivity Analysis”
15. CSS 32-9044150-001 “{ } { }”
16. CSS 32-9030073-003 “FORTRAN 95 Coding for the Low DNBR Channel Trip Algorithm for EPR”
17. CSS 32-9030073-001 “FORTRAN 95 Coding for the Low DNBR Channel Trip Algorithm for EPR”
18. CSS 32-9019057-004, -005 “Open Shop Code – SPNDRES”
19. CSS 32-9054828-001 “{ }”

20. CSS 32-9051922-001 “{ } { }”
21. CSS 32-9049572-001 “{ } { }”
22. CSS 32-9032026-000 “U.S. EPR DC – { }”
23. CSS 32-9051444-001 “UCBWAT Power TH Calculations for U.S. EPR DC (SRP CH. 15.4.2)”
24. CSS 32-9032134-004 “US EPR: Bias Curves and DNBR Algorithm Uncertainty”
25. CSS 32-9061804-002 “US EPR: DNB and FCM Evaluation of SRP 15.1 Events”
26. CSS 32-9048521-001 “US EPR:{ }”
27. CSS 32-9086230-001 “US EPR: Representative DNBR and FCM Plats for Chapter 15 FSAR RAIS”
28. CSS 32-9088052-000 “US EPR: Representative DNBR Plots for Chapter 15.2 FSAR RAIS”
29. CSS 32-9061773-002 “{ }”
30. CSS 32-9107307-001 “Verified Computer Code TRICS”
31. CSS 32-9011635-002 “USEPR RCS Hydraulic Analysis”

Group 4 Design Report (DR)

1. DR 38-9020405-000 (NFPSD DC 1025) "{ } { }”

Group 5 Engineering Information Record (EIR)

1. EIR 51-7013091-000 “US EPR DC Input Summary for Calculations using STATICS 2.0”
2. EIR 51-9059741-002 “TRICS Theory and User’s Manual”
3. EIR 51-7013093-000 “US EPR DC Input Summary for Calculations using TRICS 2.0”

4. EIR 51-9095690-001 “U.S. EPR Safety Related Core I&C Functional Requirements for SDD Input”
5. EIR 51-7013091-000 “US EPR DC Input Summary for Calculations using STATICS 2.0”
6. EIR 51-7013093-000 “US EPR DC Input Summary for Calculations using TRICS 2.0”
7. EIR 51-7014125-000 “US EPR DC Single Undetected Failed SPND: Impact on US EPR Design”
8. EIR 51-9050656-000 “EPR - STATIC Setpoint Methodology”
9. EIR 51-7013067-000 “US EPR RAI 505 Q07.01-33 Input Summary for Calculations Using SPNDRES 2.1”

Group 6 Miscellaneous Documents

1. 43-10287P-001 “Incore Trip Setpoint and Transient”
2. 15-9025647-002, -003, "U.S. EPR System Description Protection System (PS) (KKS:JR)”
3. ANP-10287P ” Incore Trip Setpoint and Transient Methodology for U.S. EPR”

APPENDIX II

AUDIT PLAN

Audit Plan for Review of Topical Report ANP-10287P, "Incore Trip Setpoint and Transient Methodology for U.S. Evolutionary Power Reactor," in regard to the Design Documentation and Supplemental Information.

A. Dates

This audit will be conducted in two phases and in two locations, as defined below:

Phase 1: November 3, 2014, through November 6, 2014, at AREVA Headquarters in Lynchburg, VA

Phase 2: (Optional) **Phase 2 meeting was not needed.**
[Dates TBD: Period is August 2014 thru June 2015]
AREVA Twinbrook Office in Rockville, MD

B. Background

AREVA submitted Topical Report (TR) ANP-10287-P, "Incore Trip Setpoint and Transient Methodology for U.S. EPR" for U.S. Nuclear Regulatory Commission (NRC) staff review and approval on November 27, 2007. The NRC staff performed a safety evaluation (SE) on June 16, 2009, that found the TR acceptable. However, since the safety evaluation report (SER), the TR has undergone a major revision in the modeling of the trip setpoint and transient methodology.

Therefore, the NRC staff determined it is necessary to review the documents that support the analyses and development associated with the revision of the TR and supplemental information in order to complete its review of ANP-10287.

C. Regulatory Audit Bases

General Design Criterion (GDC) 10, "Reactor Design"

GDC 11, "Reactor Inherent Protection"

GDC 13, "Instrumentation and Control" GDC 20, "Protection System Functions"

GDC 25, "Protection System Requirements for Reactivity Control Malfunctions"

GDC 29, "Protection Against Anticipated Operational Occurrences"

10 CFR Part 50, Appendix B, Criterion XI, "Test Control," and Criterion XII, "Control of Measuring and Test Equipment"

10 CFR 50.36(c)(3), "Technical Specifications"

10 CFR 50.36(d)(1)(ii)(A), "Technical Specifications"

10 CFR 50.55a (h), "Protection and Safety Systems"

D. Regulatory Audit Scope

The audit team will review documents, drawings, analysis, specifications and other relevant design information associated with the latest update of AREVA TR ANP-10287P, "Incore Trip Setpoint and Transient Methodology for U.S. EPR," and supplemental information in regard to DCD Chapters 4 and 15.

The following itinerary lists the audit topics for each portion of the audit: Phase 1, (Dates November 3, 2014, through November 7, 2014)

- Based on the previous audit preparation meeting, NRC staff will look for documentations and justifications regarding why the latest new set points are conservative to take into account the presence of the unidentifiable SPND single failure.
- On May 30, 2013, AREVA pointed out that 29 internal CRs were issued related to the set point methodology topical report and the relevant Chapter 15 anticipated operational occurrence (AOO) analyses. During the audit, the NRC staff will need to access all these condition reports (CRs) and relevant calculations. The NRC staff will review both the technical approach and the QA process related to these CRs.
- All the new Chapter 15 AOO analyses using the newly revised set point methodology. Staff needs to access the new calculations supporting the latest FSAR revisions in Chapter 15.
- Based on the response of RAI 544, Question 04.04-67, Supplement 2, staff will look for documentations and justifications regarding the development of the DNBR online algorithm related to capturing the effects of reduced coolant flow, reverse flow, and penalties associated with non-uniform flow for 3-loop RCS operation.

Phase 2, [Dates TBD: Period is August 2014 thru June 2015] (Optional)

- Follow-up items from previous audit

E. Information and Other Material Necessary for the Regulatory Audit

The NRC staff requires access to knowledgeable personnel regarding the AREVA ANP-10287-P SPNDs calibration with respect to incore trip setpoint and transient methodology. Specific areas of personnel expertise is requested in the topics listed in Section D Phase 1.

F. Team Assignments

John Budzynski, Reactor Systems Engineer, NRO (Audit Team Lead)
Jose March-Leuba, ORNL Consultant
Shanlai Lu, Senior Reactor Systems Engineer, NRO
William Gleaves, Project Manager, NRO

G. Logistics

This audit consists of multiple dates, to cover different areas of review related to initial core process. The audit will be conducted at the AREVA location identified above. The audit is scheduled to begin at 8:30 A.M. each day of the audit. Participating individuals will meet at the audit location.

H. Special Requests

Appropriate handling and protection of proprietary information shall be acknowledged and observed throughout the audit.

NRC may request an ad-hoc extension of the audit if findings during the ongoing audit reveal the need for additional time. Such an extension will be requested before the audit is adjourned by the NRC staff responsible for the audit.

I. Deliverables

The audit team will issue a regulatory audit summary within 90 days after completing the audit. The audit outcome will be used to identify any additional information to be submitted for making regulatory decisions. The audit will assist the NRC staff in the preparation and issuance of further RAIs for the licensing review of ANP-10287-P.

APPENDIX III

NRC GUIDANCE

1. RG 1.203 “Transient and Accident Analysis Methods”
2. RG 1.206 “Combined License Applications for Nuclear Power Plants (LWR Edition)”
3. SRP 4.2 “Fuel System Design”
4. SRP 4.3 “Nuclear Design”
5. SRP 4.4 “Thermal and Hydraulic Design”
6. SRP 7.2 “Reactor Trip System”
7. SRP 15.1.1-15.1.4 “Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve”
8. SRP 15.1.5 “Steam System Piping Failures Inside and Outside of Containment (PWR)”
9. SRP 15.2.1-15.2.5 “Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)”
10. SRP 15.3.1-15.3.2 “Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions”
11. SRP 15.3.3-15.3.4 “Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break”
12. SRP 15.4.1 “Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition”
13. SRP 15.4.2 “Uncontrolled Control Rod Assembly Withdrawal at Power”
14. SRP 15.4.3 “Control Rod Misoperation (System Malfunction or Operator Error)”
15. SRP 15.4.4-15.4.5 “Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate”
16. SRP 15.4.6 “Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR)”
17. SRP 15.4.7 “Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position”
18. SRP 15.4.8 “Spectrum of Rod Ejection Accidents (PWR)”
19. SRP 15.6.1 “Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve”