Attachment A

Inspection Report Number 05200020/2013-203
Second Reply to Notice of Violation

Restatement of NRC Request:

Violation No. 05200020/2013-203-01 was cited in the NOV because the NRC inspectors identified two examples where AREVA failed to adequately implement measures to assure that conditions adverse to quality are promptly identified and corrected. In the first example, AREVA failed to evaluate the extent of condition for the input errors in the RELAP5/MOD2-B&W input decks developed for a large break loss-of-coolant accident (LBLOCA) analysis. In the second example, AREVA failed to evaluate the extent of condition for the programmatic issue of open design change requests (DCRs) that were suspended when work was delayed in 2010 and not identified as needing to be completed to support restart of the Design Certification application.

AREVA's response to Violation No. 05200020/2013-203-01 did not provide adequate detail for the staff to conclude that your proposed corrective actions would completely address the identified violation. Specifically, the NRC requested in the cover letter that your response to the Notice should also address extent of condition for input errors to software performing safety-related calculations, and for open DCRs that were suspended when work was delayed in 2010 and not identified as needing to be completed to support restart of the Design Certification application. The AREVA response identified that there were additional DCRs and Condition Reports (CRs) that needed to be tracked and that there was a potential extent of condition for input errors to software performing safety-related calculations. The NRC staff needs additional information related to the extent of condition to determine the significance and the adequacy of the corrective actions that AREVA has initiated.

The NRC requests that AREVA provide a list of the DCRs and CRs needed to be tracked and the software performing safety-related calculations that have a potential extent of condition for input errors. The list should include a title, subject matter description, and AREVA reference number. The information should be descriptive enough for the NRC staff to determine the significance of the extent of condition and the adequacy of the corrective actions that AREVA has initiated.

AREVA NP Response:

Table 1 provides the list of CRs and DCRs identified in AREVA's extent of condition review as needing to be tracked by the U.S. EPR Design Certification project schedule. The table provides a title and AREVA reference number for each item, as well as a brief description of the subject matter and the current status of the CR or DCR. As stated by AREVA in Reference 2, the U.S. EPR DC project schedule was updated to incorporate and track these CRs and DCRs.

Table 2 provides a list of the software packages used to perform safety-related calculations that AREVA identified as having a potential extent of condition for user input errors, similar to the identified error using RELAP5/MOD2-B&W. The table provides the name and description of each program. As stated by AREVA in Reference 2, individual reviews of these codes are ongoing and are being performed by the appropriate AREVA personnel. An interim error notice was assigned to each code identified to alert users to potential limitations of the code while mitigating actions are being developed. Full compliance will be achieved by May 16, 2014.

Table 1 – CRs and DCRs Identified as Needing to Be Tracked on the U.S. EPR DC Project Schedule

Item	Title	Description	Status
CR 2010-8895	FSAR Chapter 15 Manual Operator Action Not Supported by Design	Operator action to trip two RCPs 30 minutes after reactor trip in support of Chapter 15 feedwater line break analyses is not documented in I&C functional requirements document.	CR resolved with no FSAR impact.
CR 2011-4236	2010-778 Software Impact Assessment identified that the FSAR is impacted by Software Error Notice 407-001.	Vendor provided software error notice that the MAAP4 code is susceptible to a truncation error. Neither the MAAP manual nor the example input deck specified that values beyond the 80th column would not be processed by MAAP's input processor.	Changes implemented in Interim Rev. 6 of FSAR
CR 2012-3877	Failure to Update EQ curves in Appendix 3 D of the FSAR.	The response to RAI 209 Question 06.02.01-14 was transmitted to the NRC in RAI 209, Supplement 1 on December 17, 2009. The response stated that U.S. EPR FSAR Tier 2 Sections 6.2.1 and Section 6.2.2 had been revised accordingly and were superseded in their entirety by the FSAR markups. However, this letter did not provide any changes to the EQ curves in Appendix 3 D of the FSAR.	Changes implemented in Rev. 5 of FSAR
CR 2013-4193	Contact radiation monitors cannot currently detect beta dose, i.e., tritium (H-3) due to the low energy.	It was not identified that monitor points R-35 to R-38 and R-66 to R-69 also need to have a grab sample and tritium (H-3) analysis capability. Tritium is a low-energy beta emitter such that volumetric measurement is not possible with currently available contact radiation monitors.	In schedule
CR 2013-4705	Quality issues with response to RAI 553, Supplement 3, Question 06.03-18, ECCS pump NPSH	Quality issues were identified concerning AREVA's response to RAI 553, Supplement 3, Question 06.03-18, and the supporting FSAR markups. Insufficient information was provided in the response in regards to the IRWST water level and available NPSH. A review of the response also identified unclear statements of LHSI / MHSI pump operation and duration over the event.	Changes implemented in Rev. 5 of FSAR
CR 2013-4788	Debris Sources from MSLBs and MFLBs Affecting ECCS Performance	For GSI-191, debris sources from MSLBs and MFLBs were not considered in the AREVA evaluation for potential breaks that could cause sump strainer debris blockage	In schedule

Item	Title	Description	Status
CR 2013-5385	Errors in U.S. EPR FSAR Tier 2 Table 6.2.4-1	The subject of this CR addresses whether or not U.S. EPR Nuclear Island Vent and Drain System valves KTD10AA008 and KTD10AA025 (Reactor Shield Building isolation valves for the Annulus Sump) should be considered as Containment Isolation Valves (CIVs). The valves currently are identified as CIVs and are listed in the U.S. EPR FSAR Tier 2 Table 6.2.4-1.	In schedule
CR 2013-5744	Volumes for tanks of the Solid Waste Processing System used in the U.S. EPR FSAR are inconsistent with volumes identified in supporting documentation	The volumes of solid waste management tanks used in the Solid Waste Management System section (U.S. EPR FSAR Tier 2 Chapter 11.4) are inconsistent with volumes identified in the Solid Waste Processing System Description. This resulted in incorrect, though conservative, volumes used in Radiation Protection calculations.	In schedule
CR 2013-5773	Technical Specification Bases were not revised to include power supplies for PSRV1 and PSRV1 as identified in CR 2010-7766	An evaluation determined that it was not required to list the power supply for the solenoid valves of PSRV1 and PSRV2 in the Technical Specifications (Tech Specs) Bases, because Tech Specs do not list all the necessary attendant items that are required for components to perform their specified safety function.	CR resolved with no FSAR impact.
CR 2013-5781	Incorrect markup in response to RAI 370, Supplement 24	AREVA's response to RAI 370, Supplement 24, incorrectly referenced two sections for new COL Item 2.5-13: Section 2.5.4.10.1 and Section 3.7.2.8. This is a Chapter 2 COL item and should therefore not make reference to Chapter 3.	Changes implemented in Interim Rev. 6 of FSAR
CR 2013-6134	U.S. EPR FSAR Chapter 15.0 - Updates necessary to reflect the recent SGTR re-analysis	Three SGTR-related FSAR updates were determined to be necessary following the recent revision of the U.S. EPR SGTR analysis. These updates are editorial in nature and are necessary to reflect the SGTR re-analysis completed in April 2013.	Changes implemented in Interim Rev. 6 of FSAR
CR 2013-6162	U.S. EPR FSAR referenced Topical/Technical Report (TR) Revision numbers are inconsistent with current revisions available.	In a recent review of TRs supporting the FSAR, two generic issues were identified: (1) TR revision numbers referenced within the FSAR are not consistent with the current revision of the report available, and (2) FSAR Table 1.6-1 is inconsistent with the reference pages found within the specific FSAR sections, as well as the in-text citations.	Changes implemented in Interim Rev. 6 of FSAR

Item	Title	Description	Status
CR 2013-6218	Containment temperature and ultimate pressure capacity limits	FSAR Table 3.8-6, "Containment Ultimate Pressure Capacity (Pu) at Accident Temperature of 310°F," provides the pressure in the Containment Building. The temperature is at 310°F, which corresponds to the saturation temperature of the design pressure of 62 psig. However, this is for DBA (Design Basis Accident). The ELAP event and Containment Building ultimate pressure capacity limits are for Beyond Design Basis Events. The saturation temperature increases as pressure increases. The ultimate pressure capacity limit may not be as high at an analyzed higher temperature.	In schedule
CR 2013-6290	U.S. EPR Technical Specification 3.7.22 is incomplete.	A review of the SGTR event analysis should be performed to determine if the opening function of the transfer valves meets the definitions in 10 CFR 50.36. If so, TS 3.7.22 and the corresponding bases should be revised as necessary.	In schedule
CR 2013-6315	U.S. EPR FSAR Tier 2 Table 3.10-1 has duplicate tag numbers on 2 different components	The tag number for the electrical room supply air prefilter was erroneously assigned as 30SAD13AT003 while marking up the revisions to FSAR Figure 9.4.9-1. However the recirculation unit moisture separator on the same figure already existed with the same tag number. The correct tag number for the electrical room supply air prefilter is 30SAD13AT004.	Changes implemented in Interim Rev. 6 of FSAR
DCR 113-7007690-000	Revised U.S. EPR LOCA and MSLB Pressure/Temperature Profiles for Structural Analyses	A revision to "Pressure/Temperature Profiles for EQ and Structural Analyses" is required to address the implementation of a multi-node GOTHIC model. The revised profiles include updated MSLB and LOCA profiles resulting from analyses that utilized a multi-node GOTHIC model of the U.S. EPR Containment and the changes to the LOCA methodology presented in AREVA NP Technical Report ANP-10299P, "Applicability of AREVA NP Containment Response Evaluation Methodology to the U.S. EPR for Large Break LOCA Analysis."	Implemented
DCR 113-7013889-000	EDG Auxiliary System Revisions in Response to CR 2013-2419	Relief valves are required on the tube side and shell side of each of the three heat exchangers/train to meet ASME Section III requirements. Tube side and shell side relief valves should be added to the EDG Jacket Water Loop Heat Exchangers, the EDG Intercooler Loop Heat Exchangers, and the EDG Lube Oil Coolers.	Implemented

Item	Title	Description	Status
DCR 113-9113736-000	Revisions to U.S. EPR Documents per WebCAP 2009-1209	An AREVA calculation was reviewed to answer U.S. EPR RAI 155 03.08.05-8 Part 9 in regards to the gusset section's (at base of RCB and RSB walls) ability to resist the lateral sliding force of the RBIS basemat. During this review, it was determined that the gusset section was not properly designed to resist radial shear, the effects of accidental torsion were not included in the design of the gusset section, and the torsional moment was not considered for the flexural design of the gusset section.	Implemented
DCR 113-9113737-000	Revisions to U.S. EPR Documents per WebCAP 2009-1322	The Polar Crane live load was incorrectly assumed to be included in the results of the Nuclear Island static model. Due to this assumption, the Polar Crane live load was not included in the calculations listed in various calculation files.	Implemented

Table 2 – AREVA Software Identified as Potentially Susceptible to Input Restrictions

Software Name	Description
aeolus	Aeolus estimates atmospheric transport and dispersion for routine releases from nuclear power facilities.
ANF-RELAP	ANF-RELAP is a derivative of RELAP5/MOD2 used for non-LOCA transient system analyses.
ANSYS	ANSYS is a general purpose Finite Element Analysis program for multiphysics.
ANYOLS	ANYOLS is a stepwise regression program. The software is used to fit data using ordinary or weighted least squares method.
CONTEMPT	CONTEMPT is used to calculate the reactor building pressure-temperature responses following an accident, as well as mass and energy releases.
DATATRAK	DATATRAK is a tool for managing and reporting component, cable, and raceway information by fire area to support 10 CFR 50 Appendix R and NFPA-805 compliance analyses.
DORT	DORT is a 2D/3D discrete ordinates neutron transport code.
ELISA2	ELISA2 is a FORTRAN-77 software package for the radiological evaluation of licensing and Severe Accidents at nuclear power plants. It is based on deterministic models for the computation of radioactivity levels, gamma spectra, and radiation exposures following routine and accidental releases of fission products, activation products, and actinides.
ETAP	ETAP is a fully-graphical electrical transient analyzer program. ETAP performs load flow, short circuit, motor starting, coordination, harmonic analysis, transient stability, battery sizing, and discharge analysis.
FIBWR2	FIBWR2 is a thermal-hydraulics code used for steady state and transient analysis of BWR cores.
FTREX	The EPRI FTREX is a software tool that generates and quantifies minimal cutsets from reliability models for probabilistic risk assessments.
galepg	Galepg calculates the release of radioactive material in gaseous effluents from PWRs.
galepl	Galepl calculates the release of radioactive material in liquid effluents from PWRs.
gaspar2	Gaspar2 evaluates the radiological impact of radioactive releases to the atmosphere from nuclear power facilities.
GENRUP	GENRUP evaluates radiological impacts of steam generator tube rupture accidents at commercial nuclear power stations. GENRUP computes TEDE doses and includes the alkalis.
GIP	GIP reads nuclide-organized cross-section libraries and prepares a group-organized library.
ladtap2	Ladtap2 performs environmental dose analyses for releases of radioactive effluents from nuclear power plants into surface waters.
LYNXT	LYNXT is a versatile thermal-hydraulics crossflow code capable of predicting flow and temperature (enthalpy) distributions in confined geometries where wall shear forces are more dominant than intra-fluid shear forces.
MAAP4	MAAP4 is a thermal-hydraulics code used to calculate severe accident phenomenon during both the in-vessel and ex-vessel phases.
MACCS2	MACCS2 is an atmospheric dispersion code that estimates the radiological and economic impacts of a release of radioactive material into the atmosphere.
METROSE	METROSE produces joint frequency distribution tables of wind speed and wind direction as a function of atmospheric stability class.
MONK	MONK is a Monte Carlo program for nuclear criticality safety and reactor physics analyses.

Software Name	Description
PAVAN	PAVAN is an NRC atmospheric dispersion code for accidental releases.
PIEOSG2	PIEOSG2 is a B&W computer code developed for the Bellefonte Project to predict IE-OTSG secondary side fluid thermal-hydraulics conditions.
precip	Precip calculates the monthly precipitation sum and the weighted monthly percent frequency of occurrence of precipitation, averaged over a user-specified timeframe.
PTPC	PTPC is used to calculate reactor vessel Pressure-Temperature operating limits.
PTSPWR2	PTSPWR2 is a non-LOCA transient analysis code.
RELAP5-3D	RELAP5-3D is a highly generic code that, in addition to calculating the behavior of a reactor coolant system during a transient, can be used for simulation of a wide variety of hydraulic and thermal transients in both nuclear and nonnuclear systems.
RELAP5MOD2	RELAP5MOD2 is a multi-purpose light water transient analysis FORTRAN code used for a variety of thermal-hydraulic analyses. It is widely used for LOCA, Non-LOCA Safety Analysis, and plant performance analyses.
RiskSpectrum PSA	RiskSpectrum PSA is an advanced fault tree and event tree software tool.
RODEX2	RODEX2 incorporates models for significant physical phenomena which contribute to defining the deformational-composition-thermal conditions within a fuel rod.
RODEX2-2A	RODEX2-2A is used for SEM/PWR-98 LBLOCA analyses.
RODEX3A	RODEX3A simulates the thermal and mechanical response of a fuel rod in a coolant channel as a function of exposure for the normal and power ramp conditions encountered in pressurized and boiling water reactors.
SCALE	SCALE is a comprehensive modeling and simulation suite for nuclear safety analysis and design.
SHAPEPWR	SHAPEPWR adjusts potential axial shapes generated from neutronics Xenon transient calculations to an axial shape peaked such that it simultaneously satisfies Fq and Fdh technical specification limits per Appendix K of 10 CFR 50 for LOCA analyses.
SKIRON2	SKIRON2 determines atmospheric dispersion factors for accidental releases.
SORREL	SORREL is a program used to generate an energy dependent neutron source for the DORT discrete ordinates computer program.
S-RELAP5	S-RELAP5 is a RELAP5-based thermal-hydraulics system code for performing realistic analyses of LBLOCA in pressurized water reactors in compliance with the revised LOCA Emergency Core Cooling System rule. It is also suitable for analyzing SBLOCA and non-LOCA transients.
SWAN	The SWAN computer code is a third-generation wave model for obtaining realistic estimates of wave parameters in coastal areas, lakes, and estuaries from given wind, bottom, and current conditions.
TCPYA	Post-Processing RETRAN BWR Hot Channel/Transient Simulation to obtain MCPR.
temprh	Temprh calculates temperature, humidty, and dewpoint averages and extremes.
TOTUNC	TOTUNC calculates uncertainty for a set of Traversing Incore Probe (TIP) traces.
VAGEN	VAGEN is a steady-state thermal-hydraulic code for one-dimensional parallel or crossflow, single pass tube, and shell heat exchangers. The program is used to calculate performance of once-through steam generators and integral economizer once-through steam generators.
XOQDOQ	XOQDOQ estimates atmospheric transport and dispersion of gaseous effluents in routine releases from nuclear power plants.