November 1, 2007

Mr. Ronnie L. Gardner
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SUBJECT: AREVA NP, INC. – REQUEST FOR ADDITIONAL INFORMATION REGARDING ANP-10284, "U.S. EPR INSTRUMENTATION AND CONTROL DIVERSITY AND DEFENSE-IN-DEPTH METHODOLOGY TOPICAL REPORT" (TAC NO. MD5884)

Dear Mr. Gardner:

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated June 20, 2007, AREVA NP, Inc. submitted for NRC staff review, ANP-10284, Revision 0, "U.S. EPR Instrumentation and Control Diversity and Defense-in-Depth Methodology Topical Report." By letter dated August 20, 2007, the NRC staff accepted this topical report for review.

The NRC staff is reviewing your submittal and has determined that additional information is required in order to complete the review. The specific information requested is addressed in the enclosure to this letter, and is substantially the same as the draft that was provided to your staff electronically on September 27, 2007 and discussed during a telephone conference on October 8, 2007. AREVA has agreed to provide a response within 90 days of receipt of this letter.

The NRC staff considers that timely responses to requests for additional information help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of staff resources. If you should have any questions, I may be reached at (301) 415-3361.

Sincerely,

/RA/

Getachew Tesfaye, Sr. Project Manager
EPR Projects Branch
Division of New Reactor Licensing
Office of New Reactors

Project No. 733

Enclosure: Request for Additional Information
cc: See next page

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REQUEST FOR ADDITIONAL INFORMATION

ANP-10284, "U.S. EPR INSTRUMENTATION AND CONTROL

DIVERSITY AND DEFENSE-IN-DEPTH METHODOLOGY

TOPICAL REPORT"

PROJECT NO. 733

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated June 20, 2007 (Agencywide Documents Access and Management System (ADAMS Accession Number ML071760187), AREVA NP, Inc. submitted for NRC staff review, ANP-10284, Revision 0, "U.S. EPR Instrumentation and Control Diversity and Defense-in-Depth Methodology Topical Report." By letter dated August 20, 2007 (ADAMS Accession Number ML072150532), the NRC staff accepted this topical report for review. The NRC staff is reviewing AREVA's submittal and has determined that the following additional information is required in order to complete the review.

RAI 1. Provide the diversity and defense-in-depth (D3) analysis that is described in the methodology topical report (TR).

ANP-10284, Section 1.1 states, "AREVA NP requests the approval of the following items in this report:..The adequacy of the proposed design features to mitigate the consequences of a postulated CCF [common cause failure] in the safety I&C [Instrumentation and Control] systems."

However, NUREG-0800, "Standard Review Plan," Branch Technical Position (BTP) 7-19, "Guidance for Evaluation of Diversity and Defense-Indepth in Digital Computer-Based Instrumentation and Controls System," states, "...the NRC has established the following …position on D3...

Point 1. The applicant/licensee should assess the D3 of the proposed I&C system to demonstrate that vulnerabilities to common-cause failures have been adequately addressed.

Point 2. In performing the assessment, the vendor or applicant/licensee should analyze each postulated common-cause failure for each event that is evaluated in the accident analysis section of the safety analysis report (SAR) using best-estimate or SAR Chapter 15 analysis methods. The vendor or applicant/licensee should demonstrate adequate diversity within the design for each of these events." ….

Therefore BTP 7-19 states that the adequacy of proposed design features to mitigate the consequences of postulated CCF should be based on an assessment that has been performed.

In order to evaluate the proposed design features, an assessment or a justification, as to how the alternative proposed (i.e., the methodology for performing the assessment) provides an acceptable method of complying with the rules or regulations, is needed.

RAI 2. Provide a description of how adequate quality is achieved for the diverse actuation system (DAS).

One of the requirements that is addressed by BTP 7-19 is 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants." Section 50.62(c)(1) states, "...This equipment must be designed to perform its function in a reliable manner..."

AREVA has stated that the DAS will be used to address ATWS requirements. (ANP-10284, Section 4.2, states, "...ATWS evaluations to determine required functionality of the DAS for ATWS mitigation.")

Provide a description of the DAS that describes how it is designed to perform its function in a reliable manner (see Generic Letter 85-06, "QA Guidance for ATWS Equipment that is not Safety Related," and NUREG-0800, Chapter 7, "Instrumentation and Controls – Overview of Review Process," Section 7.8, "Diverse Instrumentation and Control System").

RAI 3. Provide a description of how the DAS performs its function independent (from sensor output to the final actuation device) from the protection system (PS).

One of the requirements that is addressed by BTP 7-19 is 10 CFR 50.62. Section 50.62 (c)(1) states, "...This equipment must... be independent (from sensor output to the final actuation device) from the existing reactor trip system."

AREVA has stated that the DAS will be used to address ATWS requirements. (ANP-10284, Section 4.2: "... ATWS evaluations to determine required functionality of the DAS for ATWS mitigation.")

AREVA has not provided a description of the DAS that describes how it performs its function independent (from sensor output to the final actuation device) from the PS.

ANP-10284, Section 2, states, "The I&C architecture for the U.S. EPR is depicted in Figure 2-1. The I&C architecture is arranged into three levels—Level 2 (Supervisory Control), Level 1 (System Level Automation), and Level 0 (Process Interface). In general, functions (both automatic and manual) are allocated to the various Level 1 systems depending on the safety classification of the function, and what the function is designed for (e.g., rod control, initiation of safety injection). Interfaces are provided within the Level 2 I&C systems for manual functions."

ANP-10284, Section 2.5, states, "In general, the lines of defense apply to the architecture level 1 automation systems."

The assessment methodology proposed by AREVA does not seem address the regulatory basis, since Level 0 (Process Interface) is excluded from the analysis.

RAI 4. Please demonstrate that the DAS will be sufficiently reliable to perform its diverse backup function for the Protection System (PS).

The DAS subsystem serves as a diverse backup to the PS. The description of the DAS however does not allow for an evaluation of DAS reliability. In view of the fact that simple systems are generally more reliable than complex systems (i.e. complex systems have a greater likelihood of failures and software errors), simple digital systems have been identified to be acceptable. The proposed DAS will be expected to perform this function reliably.

It is understood that the D3 report is not intended to include a description of the DAS, but rather to identify those D3 related design constraints on the DAS that are required to support the D3 analysis.

RAI 5. Describe how the proposed design features or assessment methodology addresses sensor independence between the echelons of defense.

BTP 7-19 identifies NUREG/CR-6303, "Method for Performing Defense-in-Depth and Diversity Analyses of the Reactor Protection System" as relevant regulatory guidance, and acceptance criteria, for addressing the regulatory requirements addressed by BTP 7-19. NUREG/CR-6303, Section 2.2, states, "All four echelons depend upon sensors to determine when to perform their functions, and a serious safety concern is to ensure that no more than one echelon is disabled by a common sensor failure or its direct consequences" (see also quotation from NUREG-0493, "A Defense-in-Depth and Diversity Assessment of the RESAR-414 Integrated Protection System," in RAI No. 8).

ANP-10284, Section 3.2.1.4, states, "The PAS provides diverse processing of sensor information because the PAS obtains sensor information independently of the PS and SAS [Safety Actuation System] software." This implies that the same sensors are used by both safety and non-safety systems (see also Figure 3-5).

AREVA has proposed to define different echelons of defense than those defined in NUREG/CR-6303. In addition, the analysis methodology proposed does not seem to address sensor independence (see also quotations from Section 2 and 2.5 in RAI No. 3 above). Therefore, the D3 TR does not appear to address the concern of sensor independence.

Note: NUREG-0800, Chapter 7, Section 7.5, "Information Systems Important to Safety," contains guidance for meeting regulatory requirements. The acceptance criteria contained in NUREG-0800, Section 7.5 include Regulatory Guide 1.97, Revision 4, which endorses the Institute of Electrical and Electronics Engineers (IEEE) STD 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations –Description." IEEE 497-

2002, Clause 6.2, “Common Cause Failure,” contains additional guidance for “instrumentation using microprocessor based sensors, data acquisition, or display equipment.” For example:

- “Common cause failures for the instrumentation channels shall be addressed at the variable level.”
- “System interaction between accident monitoring microprocessor-based instrumentation systems and other systems that may be served by the data acquisition and display system shall be considered as part of the common cause failure evaluation.”

RAI 6. Describe how the proposed design features or assessment methodology addresses display independence.

BTP 7-19 states, “This BTP has the objective of confirming that vulnerabilities to common-cause failures have been addressed in accordance with the guidance of the SRM on SECY-93-087, specifically: ... Verify that the displays and manual controls for critical safety functions initiated by operator action are diverse from computer systems used in the automatic portion of the protection systems.”

AREVA has proposed to define different echelons of defense than those defined in BTP 7-19. In addition, the analysis methodology proposed does not seem to address display independence (see quotations from Section 2 and 2.5 in RAI No. 3 above).

RAI 7. Describe how the proposed design features or assessment methodology addresses IEEE STD 603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations –Description,” manual control independence requirements.

IEEE 603, Section 6.2.1 states, “Means shall be provided in the control room to implement manual initiation at the division level of the automatically initiated protective actions. The means provided shall minimize the number of discrete operator manipulations and shall depend on the operation of a minimum of equipment consistent with the constraints of 5.6.1.”

ANP-10284, Section 2.1, states, “For the initiation of critical safety functions at the system level (e.g., reactor trip, safety injection), conventional means (i.e., buttons, switches) are provided on the SICS [Safety Instrumentation and Control System]. These signals bypass the TXS computers and are hardwired directly to actuation devices (e.g., reactor trip devices or priority actuation and control (PAC) modules).”

ANP-10284, Section 4.3.1, states, “The inventory of hardwired controls on SICS is developed using the following requirements:

- System-level manual actuation for critical safety functions, which include: reactor shutdown, core inventory control, decay heat removal, containment isolation and containment integrity.

- System level manual actuation of those safety functions that were credited for manual operator action in Step 2."

It is not clear that "the automatically initiated protective actions" of IEEE 603 are a subset of "critical safety functions" as described in the D3 TR. If these are in fact two names for the same set of controls, then why use two different names?

AREVA uses different terms than those used in IEEE 603. In addition, the analysis methodology proposed by AREVA, does not seem to address assessment of IEEE 603 manual control independence (see quotations from Section 2 and 2.5 in RAI No. 3 above).

RAI 8. Describe how the proposed methodology addresses the assessment of the separation of protection and control systems.

One of the requirements that is addressed by BTP 7-19 is 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," which states, "Criterion 24--Separation of protection and control systems. ... Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired."

NUREG-0493 (first complete paragraph at the top of page 4-12) documents that a plant, has in the past, been accepted to "... be control from the same measurements with which it is protected ... " based on a " ...demonstration of insignificant impairment to safety."

ANP-10284, Section 4.3.2, states, "Safety-related plant equipment will have the capability of being controlled manually at the component level from the PICS via the PAS and PACS. This will fulfill the requirement of performing manual functions that don't require system level manual actuation."

The description of the proposed design features does not address the regulatory requirement for separation of protection and control systems. In addition, the analysis methodology does not seem to address the demonstration of insignificant impairment to safety (nor assessment of the separation of protection and control systems) (see also RAI No. 5 above.).

RAI 9. Provide additional documentation to support the assumption that the AV42 is not susceptible to CCF.

NUREG-0800, Chapter 7, Table 7.1 identifies "SRM to SECY-93-087 II.Q" as acceptance criteria for D3 assessments.

SECY-93-087, Enclosure 1, Page 54 states, "... digital I&C systems share more data transmission functions and shares more process equipment than their analog counter parts. Redundant trains of digital I&C systems may share databases (software) and process equipment (hardware). Therefore, a hardware design error, software design error, or software programming error may result in a common-mode or common-cause failure of redundant equipment."

It is now possible to design hardware by using software design methodologies (i.e., programmed in C, C++, or VHDL). Therefore, hardware that is designed by using software methodologies could be considered susceptible to CCF. The AV42 uses a programmable logic device (PLD) and, therefore, could be considered susceptible to common mode failures.

ANP-10284, Section 3.1.1.2, states, "Based on the design features and testing described above the AV42 is not susceptible to a CCF." However insufficient information has been provided to allow the NRC to reach this conclusion independently. The proposed methodology does not include any validation of this assumption.

A description of the PACS has not been submitted, but it is understood that the PACS will consist primarily of AV42 modules. Note: ANP-10284, Section 3.0 states, "... the main line of defense consists of the automatic safety functions performed by ... PACS, and therefore these are the systems of interest when considering CCFs."

RAI 10. Are all of the engineered safety feature (ESF) functions as described in IEEE STD 603-1991, implemented in four systems: 1) PS, 2) SAS, 3) SICS, and 4) PACS? Are there any other safety systems that implement ESF functions?

ANP-10284, Section 3.2.1.2, states, "The PS is the primary means of performing ESF actuations." Does this PS function fulfill the requirements of IEEE 603, Section 6.1?

ANP-10284, Section 3.2.1.4, states, "The SAS is the primary means of performing ESF control functions." Does the SAS address IEEE 603, Section 6.2.2?

ANP-10284, Section 3.2.1.2, states, "... manual means of actuating an ESF system is provided on the SICS in the MCR." Does this SICS function fulfill the requirements of IEEE 603, Section 6.2.1?

RAI 11. Provide an explanation of how the D3 TR lines of defense are comparable to those of NUREG/CR-6303, since the lines have different scopes.

The scope of the AREVA lines of defense do not seem to include the sensors, displays or controls associated with the I&C systems (see quotes from Section 2 and 2.4. in RAI No. 3). In addition, Section 3.0 states, "... the main line of defense consists of the automatic safety functions performed by the PS, SAS, and PACS, and therefore these are the systems of interest when considering CCFs." The displays and controls are in the SICS or PICS and are therefore, not in the scope to the D3 TR.

ANP-10284, Section 2.5: "The U.S. EPR lines of defense are compared to these four echelons of defense discussed in NUREG/CR-6303 in Table 2-2."

The scope of the lines of defense in NUREG/CR-6303 is clarified by the definitions for those systems in NUREG-0493 and IEEE STD 603-1991.

- NUREG-0493, Section 1.2.2.1: "The scram system consists of sensors, signal processors, logic, and actuation initiation devices necessary to affect the reactor trip or scram, including essential auxiliary systems. This echelon of defense performs a safety function. The scram system is also known as the reactor trip system."
- NUREG-0493, Section 1.2.2.2: "The ESF system consists of sensors, signal processors, logic, and actuation initiation devices necessary to affect engineered safety features (for example, auxiliary feedwater, containment isolation, emergency core cooling, emergency power), including essential auxiliary systems. This echelon of defense performs a safety function."
- NUREG-0493, Section 1.2.2.3: "The control system consists of all instrumentation and control equipment not included in the scram or ESF actuation systems, including automatic and manual process controls, presentation of information to the operator (plant monitoring system), and plant computer(s) that are not part of the scram or ESF actuation systems. This echelon of defense does not perform a safety function, but is nevertheless important to the defense-in-depth principle."
- IEEE 603: See Figure 1 & 3

These lines of defense explicitly include sensors, displays and controls.

RAI 12. Figure 2-2 does not include the PACS as part of the main line of defense. However, ANP-10284, Section 3.0, states, "... the main line of defense consists of the automatic safety functions performed by the PS, SAS, and PACS..." Please clarify (see also related RAI No. 15).

RAI 13. Provide a description of why the PAS should be considered to be simple digital equipment.

BTP 7-19 identifies the SRM on SECY-93-087 as relevant regulatory guidance for addressing the regulatory requirements addressed by BTP 7-19. The SRM on SECY-93-087 approved the staff position in SECY-93-087 with minor changes.

SECY-93-087 states (Enclosure 1, bottom of page 56): "The staff has concluded that analyses that demonstrate adequate rather than equivalent, defense against the postulated common-mode failures would be allowed in the diversity assessment required of the applicant. ... The staff will not require only analog equipment and will consider allowing simple digital equipment."

AREVA has stated that the PAS will be used to address CCF. (ANP-10284, Section 3.2.1.3 states, "The SAS is the primary means of performing ESF control functions. Assuming a postulated CCF renders the SAS inoperable, the PAS is available as a diverse means of executing ESF control functions." ANP-10284, Section 3.2.1.4, states, "The PAS provides diverse processing of sensor information because the PAS obtains sensor information independently of the PS and SAS software.")

AREVA has not provided a description of the PAS that describes why it should be considered to be simple digital equipment. The methodology described does not include an assessment of the simplicity of the PAS (see also related RAI No. 4).

RAI 14. Do SICS displays address the requirements of IEEE STD 603-1991, Sections 5.8.1 and 5.8.4?

ANP-10284, Section 3.2.1.4, states, "... including type A, B and C post accident monitoring variables as defined in Regulatory Guide 1.97 ... The PS and SAS are the credited means of processing these variables, and the SICS is the credited means for display."

RAI 15. List all safety systems that will be analyzed.

ANP-10284, Section 2.4, states, "In general, the lines of defense apply to the architecture level 1 automation systems." However, ANP-10284, Section 4.1, states, "An analysis of the safety I&C systems will be performed to determine their susceptibility to a CCF." Therefore, without an explicit list of the systems that will be analyzed in ANP-10284, Section 4.1, it is not clear if the PACS and SICS will be included in the D3 analysis (see also related RAI No. 12).

RAI 16. How will the analysis methodology described in the D3 TR differ from the methodology in NUREG/CR-6303.

ANP-10284, Section 4.1, states, "This analysis addresses Point 1 of NUREG-0800, BTP 7-19, and will be performed using NUREG/CR-6303 as a model." How is "using NUREG/CR-6303 as a model" different from following the methodology described in NUREG/CR-6303?

RAI 17. Describe why a postulated failure of the TXS platform is more conservative than outputs assumed to fail in a manner that is credible but that produces the most detrimental consequences.

NUREG/CR-6303, Section 3.5, "Guideline 5 - Method of Evaluation" states, "Block output signals must be assumed to fail in a manner that is credible but that produces the most detrimental consequences when analyses in accordance with Guideline 9."

ANP-10284, Section 4.1, states, "The following assumptions are to be used when performing this analysis: ... A CCF of the TXS platform is postulated (conservative assumption). This postulated CCF is such that the TXS based I&C systems do not perform their functions when required."

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