AREVA

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Washington, D.C. 20555-0001

Response to a Request for Additional Information Regarding ANP-10287P, "Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report" (TAC No. Q00013)

Ref. 1: Letter, Ronnie L. Gardner (AREVA NP Inc.) to Document Control Desk (NRC), "Request for Review and Approval of ANP-10287P, 'Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report'," NRC:07:068, November 27, 2007.

Ref. 2: Letter, Getachew Tesfaye (NRC) to Ronnie L. Gardner (AREVA NP Inc.), "Second Request for Additional Information Regarding ANP-10287P, 'Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report (TAC No. Q00013)'," January 15, 2009.

AREVA NP Inc. (AREVA NP) requested the NRC's review and approval of topical report ANP-10287(P), "Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report" in Reference 1. The NRC provided a Request for Additional Information (RAI) regarding this topical report in Reference 2. The response to this RAI is enclosed with this letter.

AREVA NP considers some of the material contained in the enclosure to this letter to be proprietary. As required by 10 CFR 2.390(b), an affidavit is provided to support the withholding of the information from public disclosure. Proprietary and non-proprietary versions of the enclosure to this letter are provided on the enclosed CDs.

If you have any questions related to this submittal, please contact Ms. Sandra M. Sloan, Regulatory Affairs Manager for New Plants Deployment. She may be reached by telephone at 434-832-2369 or by e-mail at sandra.sloan@areva.com.

Sincerely,

Mark Burzynski for

Ronnie L. Gardner, Manager
Corporate Regulatory Affairs
AREVA NP Inc.

Enclosure

cc: G. Tesfaye
Docket 52-020

D077
NRO

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. (AREVA NP) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in report ANP-10287Q2P, "Second Round Request for Additional Information (RAI), ANP-10287P, 'Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report," and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 26th

day of February 2009.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/10
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
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My Commission Expires Oct 31, 2010

Response to a Request for Additional Information – ANP-10287P "Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report" (TAC No. Q00013)

RAI-15. *Describe the impact on power distribution uncertainty of the AMS measurement. Provide an uncertainty analysis that considers the effectiveness of the AMS to assess margin to relevant operating and safety limits based on inputs to various computer models and core instrument gains. The analysis should address the influence on the AMS measurement of the following:*

a. Signal to noise,
b. Activated aeroball impurities,
c. Contaminated or activated aeroball tubing,
d. Aeroball transit time,
e. Sensitivity to calibration techniques and intervals,
f. Equipment environmental conditions,
g. Neutron/gamma transport models or correlations, and
h. Sensitivity changes that are duty related.

Response to RAI 15:

The Aeroball Measurement System (AMS) is an electromechanical, computer-controlled, fully automated, online flux mapping measurement system. It uses movable activation probes contained in lances that extend downward into the core, an activation probe transport system located outside of the reactor vessel, and activation probe measuring equipment located in a remote area. The movable activation probes, or aeroballs, are steel balls composed of carbon, chromium, iron and vanadium. The vanadium isotope V^{51} undergoes the following nuclear reaction:

$$V^{51} + n_0^1 \rightarrow V^{52} + \beta + \gamma \rightarrow Cr^{52}$$

When exposed to a neutron flux, the V^{51} in the aeroballs absorbs a neutron and reaches a higher energy state to become the isotope V^{52}. With a half life of 3.75 minutes, V^{52} undergoes β- decay to Cr^{52} and emits a gamma ray. The gamma radiation (approximately 1.43 MeV) emitted during this decay is a distinctive signature that is measured by the AMS. The activity distribution along the aeroball stacks is proportional to the neutron flux density, and thus to the power density. If measurements are performed immediately following one another, the AMS software factors residual decay energy from previous measurements into the calculations.

The aeroballs are used for the life of the plant, since there is negligible depletion of V^{51} during each measurement. However, the aeroballs can be replaced if transport problems occur.

The primary function of the self-powered neutron detectors, (SPND) is to accurately and continuously assess the 3D power density distribution and peak power density in the U.S. EPR. The SPND signals are used for core surveillance as well as for core protection. The SPNDs are (n, β) detectors with cobalt emitters that do not require a polarization voltage power supply during operation. The cobalt isotope Co^{59} is used as emitter material because of its ability to promptly generate signals that follow the change of neutron flux, while having low gamma ray sensitivity. The emitter conductor that transmits the detector signal is subject to the same gamma radiation as the emitter material, and produces a current over its entire length. A compensating conductor running alongside the emitter conductor compensates for the affect of this induced current on the measurement signal. The emitter conductor and the compensating conductor both use mineral insulated, metal-sheathed cable.

The main interactions that form electrons and contribute to the measured signal are:

- Radiative capture (n,γ) followed by a delayed β^- (i.e., (n, β^-) interaction).
- Radiative capture *(n,γ)* followed by prompt secondary electron production by Compton, photoelectric, and pair production (i.e., *(n,γ) (γ,e)* interactions).
- Absorption of external gamma and secondary electron production by Compton, photoelectric, and pair production (i.e., *(γ,e)* interaction).

The production of the Compton electrons from the Co^{59} (n,γ) Co^{60} interaction allows the SPNDs to be self-powered.

The SPNDs must be calibrated against the AMS at least once within a maximum effective full power days (EFPD) window (typically around 15 EFPD). However, SPND calibrations may be conducted at any time and may even be conducted multiple times within this window. The SPND calibration process is used to periodically determine the power density profile on the departure from nucleate boiling ratio (DNBR)-limiting fuel rod in the core, as well as the peak planar linear power density (LPD). This calibration process uses the AMS for the discrete flux measurements in the core, followed by a reconstruction algorithm that produces a fine resolution map of the core. The principles of SPND calibration are provided in ANP-10287P, "Incore Trip Setpoint and Transient Methodology for U.S. EPR", Appendix B.

As a consequence of using the AMS and the reconstruction algorithm, the reconstructed power distribution is subject to uncertainties which represent the ability of the AMS and incore monitoring software to resolve the DNBR-limiting rod and the axial offset. A methodology for assessing measured power distribution uncertainties is provided in ANP-10287P, Appendix E.

a. Signal to noise ratio

Computational corrections eliminate the residual activities of V^{52} and Mn^{56}, and the background activity of Cr is suppressed by pulse height discrimination. Thus, the noise signal does not affect the AMS results. The computational correction for V^{52} and Mn^{56} has an impact only if aeroball measurements are performed in rapid sequence, since the V^{52} activity decays after 45 minutes and Mn^{56} after 24 hours. After a rapid sequence of aeroball measurements, the validity of the corrections can be experimentally checked by performing a so called "fictitious" aeroball measurement, where the ball stacks are not activated again but only measured in the measuring table. When the results of this zero activation measurement are included in the computation correction, the corrected activation values should be near zero.

b. Activated aeroball impurities

After fabrication, the specified alloy element contents of the aeroballs are verified by chemical analysis to be within their tolerances. During commissioning, two ball stacks will be activated until they reach saturation, their activity will be measured, and the decay constants of the material components determined.

When the aeroballs are irradiated, interference background activities build up which cannot be eliminated and cannot be distinguished electronically from the V^{52} activity. These stem from the Mn^{56} activity with a half-life of 154 minutes; γ radiation emitted by In^{116} with a half-life of 54 minutes also acts as a form of background interference. Both these interference nuclides are taken into account in the computational correction of the measured count rates. The interference activities that build up over an irradiation time of three minutes results in count rates amounting to 0.8% – 1.5% of the total V^{52} count rate, depending on the time elapsed since activation.

For the aeroballs manufactured for operating plants, only V^{52} and Mn^{56} can be detected. Other impurities with small decay constants (e.g., In^{116}) that could influence the aeroball measurement results are not detectable.

c. Contaminated or activated aeroball tubing

The indicator nuclide is V^{52}, the involved reactions are neutron capture reactions, and the ball stacks are measured outside the core; therefore the activation of the aeroball tubing is not relevant. To facilitate aeroball transport, the ball stacks are lubricated with molybdenumdisulfide, which is also not detectable during the decay measurements. Finally, there is no need to correct for contamination of the tubing with primary coolant or by activated tubing material. The tubes in the detector room have no contact with primary coolant and are not exposed to a neutron flux, so they are not activated.

d. Aeroball transit time

The dwell time of aeroballs in the active core depends on the required activation time and on their position in the core. All aeroballs, with the exception of those in the uppermost cell of the core, pass through a part of the core during transport to their active zone where they receive small additional activation. This additional activation is greatest for the aeroballs in the lowermost part of the core and is approximately 0.8% – 1% of the activity measured during an activation time of three minutes. Computational corrections account for this additional activation, so variations in the aeroball dwell times in the core are a second-order error.

e. Sensitivity to calibration techniques and intervals

In addition to the interference activities that build up during activation, some of the V^{52}, Mn^{56}, and In^{116} remain from previous activations, and these are accounted for by computational corrections using the "residual activity files." They calculate the residual activities at the time of measurement by accounting for the elapsed time between measurements. The results are considered, along with interference activities built up during the current measurement, by determining an "overall background activity." For the AMS, the residual activity can be measured at any time from the control panel by using the "residual activity measurement" program in the control computer. The residual activity file can also be updated if necessary. The accuracy of the corrections can be checked using the "residual count rate measurement" program and other special measures, such as starting an aeroball measurement with disabled solenoid stops, so that the aeroballs are not activated.

Calibration of the measuring table's silicon surface detectors is performed using a Co^{60} source. This is typically done once per cycle, usually at the end of cycle before plant shutdown. This permits the impact of the calibration on activation values to be checked while the plant is operating at power, and allows a comparison of these activation values with the values from a previous measurement. A single detector calibration is performed only if an individual detector drifts or if a failed detector has been replaced.

f. Equipment environmental conditions

Reactor Cavity

- Solenoid aeroball stops
- Aeroball transport tubes

Minimum Temperature	Maximum Temperature	Humidity	Dose Rate
59°F	(normal) 122°F (abnormal) 131°F	Non-Condensing	<500 rad/hr

AMS Instrumentation Room

- Load cabinet (power supply cabinet)
- Control cabinet for ball transport system
- AMS nitrogen valve rack

Minimum Temperature	Maximum Temperature	Humidity	Dose Rate
59°F	86°F	30 – 70%	<25 mrem/hr

AMS Control Equipment Room

- AMS measuring table
- Measuring equipment instrumentation cabinet

Minimum Temperature	Maximum Temperature	Humidity	Dose Rate
59°F	86°F	30 – 70%	<25 mrem/hr

I&C Service Center / Computer Room

- Aeroball system computer
- POWERTRAX computer
- Operator station

Minimum Temperature	Maximum Temperature	Humidity	Dose Rate
68°F	79°F	30 – 60%	<0.05 mrem/hr

g. Neutron/gamma transport models or correlations

Decrease in V^{52} activity between end of activation and start of activity measurement

The time correction factors e λV52ΔT (λv52 = decay constant of V52, ΔT = time interval between end of activation and start of measurement) for determining activation values from the measured count rates are 1.1 – 1.8, depending on the measurement time. The specified decay constant λv52 (3.06x10-3s-1) was verified experimentally.

Interference of γ quanta from the adjacent detector mounting beam

Despite shielding of the detector mounting beams, γ quanta from the aeroball stacks lying under adjacent beams interfere with the semiconductor detectors or radiation extraction ports of the detector mounting beams. The interference is about 1% of the count rate measured in the adjacent beams. The only residual errors are inaccuracies in the calculation of interference factors. Their effect on power density values is reduced even further during normalization of the power density distribution.

Dead time corrections of measured count rates

The dead time losses occurring during the measurement of detector pulses must be eliminated in the event of high count rates due to the finite time resolution of the instrumentation electronics. With an activation time of three minutes at rated power, the correction for the highest count rates is about 1% of the measured count rates. The only residual error is the difference in the dead time for the various instrumentation channels, which is a second-order error.

h. Sensitivity changes that are duty related

There are no duty-related sensitivity changes in the Aeroball Measurement System. The depletion of the detector nuclide V^{52} is negligible because of the short irradiation time. Experience has shown that after 700 measurements with the same aeroball stacks, the results are indistinguishable from the results obtained with new aeroball stacks. This can be verified following an aeroball stack exchange.

RAI-16. *Section 3.2.1 of ANP-10287P indicates that RCCA position, SPND imbalance, and 3-loop operation signals are input to provide protective capability against asymmetric events. The RCCA inputs and SPND imbalance signals are discussed in Section 3.2.4 and shown in Figure 3-2. The function of the 3-loop operation signal is not discussed in detail in ANP-10287P, nor is the signal identified in any topical report figures. Provide a detailed discussion of the 3-loop operation signal and identify how this signal interacts with the RCCA and SPND imbalance signals to influence the DNBRIMB/RD trip.*

Response to RAI 16:

As described in U.S EPR FSAR Tier 2, Section 7.2.1.2.5, if a low flow condition is present in any one RCS loop, a three loop operation signal is generated. This signal is used to modify other PS functions which assume a nominal flow rate through the core.

The Low DNBR Channel uses an indicated flow rate based upon the indicated reactor coolant pump (RCP) speed (Ω), a reference RCP speed (Ωref) and volumetric flow rate (Qref), per equation C-3 of ANP-10287P. Upon receipt of a global three loop operation signal (indicating only three pumps operating), the reference volumetric flow rate (Qref) for the flow calculation is replaced with a constant, lower volumetric flow rate consistent with three pump operation. This three loop flow rate is obtained from design calculations, and stored in the plant computer as a constant. Each division will have DNBR and exit quality calculated using this adjusted reference volumetric flow rate and pump speed.

The three loop operation signal affects only the calculation of the mass flow rate and the resultant DNBR and exit quality. As shown in the U.S. EPR FSAR, Tier 2, Figure 7.2-6, the rest of the protection system logic for determining the RT thresholds is unaffected. The RCCA rod drop signals and SPND imbalance signal are used with comparisons to the first minimum DNBR signal and first maximum quality signal relative to their respective RT setpoints to determine whether the RT signal is actuated from the low DNBR Channel.

RAI-17. *Section 2.0 of ANP-10287P indicates that Section 4.2.II.2(e) of Reference 2 [NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," Nuclear Regulatory Commission, March, 2007 revision] stipulates that fuel centerline melting is not permitted for AOOs in order to preclude fuel failure. It appears that the actual citation from Reference 2 should be Section 4.2.II, SRP Acceptance Criteria, 1.B.IV. Correct or address.*

Response to RAI 17:

The citation on page 2-1 will be changed to read "Section 4.2.II, SRP Acceptance Criteria, 1.B.IV," as shown on the enclosed markup.

RAI-18. *Page A-5 of Appendix A to ANP-10287P references equations A-7 and A-8 at the bottom of the page; "The distinctions between Equation (A-7) and (A-8)..." These equations do not exist. Correct or address.*

Response to RAI 18:

The sentence that refers to equations A-7 and A-8 on page A-5 of Appendix A should refer to equations A-5 and A-6 respectively. This will be changed as shown on the enclosed markup.

RAI-19. *Page E-4 of Appendix E to ANP-10287P contains a note to "see Section 4.2 of Ref. 5" It appears this should refer to Ref 6. Correct or address.*

Response to RAI 19:

The note will be updated to state: "see Section 4.2 of Ref. 18." Reference 18 contains the derivation of the equations in Appendix E, therefore it is more applicable than Reference 6. Section 4.2 of Reference 6 is a cross reference to Section 4.2 of Reference 18.

RAI-20. *In response to RAI #3, the SPND decalibration between AMS measurement periods of 15 days has been identified for DNBR and LPD. Is this source of uncertainty included in the peak power uncertainty?*

Response to RAI 20:

Yes, the uncertainty due to the decalibration between the self-powered neutron detectors and the Aeroball Measurement System (AMS) is included in the peak power uncertainty. The mean and standard deviation of the burnup decalibration are used to apply a statistical adjustment to the hot-rod power. The way the hot-rod power is adjusted to account for this uncertainty is described in ANP-10287P, Section 5.3.11, pages 5-17 and 5-18.

RAI-21. *Provide a summary of representative technical specifications on the AMS system. Specifically, address what actions are taken if the AMS system becomes unavailable for longer than 15 days after the last calibration measurement.*

Response to RAI 21:

The scope of the U.S. EPR Technical Specifications satisfies the four criteria of 10 CFR 50.36 "Technical specifications". The Protection System Technical Specifications satisfy Criterion 3:

> "A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier."

The Self Powered Neutron Detectors (SPNDs) are explicitly included in the Technical Specifications for the Protection System (LCO 3.3.1). The SPNDs are required to be calibrated every 15 days (Surveillance Requirement (SR) 3.3.1.2). As discussed in the Technical Specification Bases for SR 3.3.1.2:

> "CALIBRATION of SPND instrumentation is performed to compensate for a decrease in SPND sensitivity during the fuel cycle and to account for peak power density factor change over the fuel cycle. The Aeroball Measurement System (AMS) assists in generating the measured relative neutron flux density in the core, which is used in conjunction with the predicted power distribution based on actual core operation to calibrate the incore SPND instrumentation."

As required by SR 3.0.1:

> "SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3."

If the SPNDs cannot be calibrated in accordance with the SR, they will be declared inoperable and the appropriate Required Action taken.

The AMS does not satisfy any of the four criteria in 10 CFR 50.36. The AMS does not:

1. Detect a significant abnormal degradation of the reactor coolant pressure boundary;

2. Represent an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

3. Form part of the primary success path which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; or

4. Has been shown by operating experience or probabilistic risk assessment to be significant to public health and safety.

Therefore, the AMS system is not explicitly included in the Technical Specifications.

RAI-22. *The topical report specifies that the on-line DNBR algorithm produces a "fault" if the enthalpy iterations do not converge. How is the fault treated in the reactor protection system?*

Response to RAI 22:

The convergence of the enthalpy rise iterations is evaluated as part of the selection of optimized bias curves. This limits the potential for software-related faults due to the online departure from nucleate boiling ratio algorithm enthalpy iteration routine. As described in ANP-10281P, "U.S. EPR Digital Protection System Topical Report," Section 7.3, faulty input signals automatically modify the reactor trip voting logic in the reactor protection system.

RAI-23. *The article noted in ANP 10287P, Reference 14 is listed as being extracted from Reliability Engineering and System Safety Journal, Volume 91. The actual collection should be Reliability Engineering and System Safety Journal, Volume 87. Correct or address.*

Response to RAI 23:

The citation on page 11-2 of ANP-10287P should read [ ] This will be updated as shown on the enclosed markup.

RAI-24. *The Introduction to ANP 10287P notes that setpoints are established for multiple functions, including the low DNBR limitation function. The low DNBR limitation function appears in Table 3-2, but it is not described in Chapter 3, nor is it described elsewhere in the Topical Report. Discuss the DNBR limitation function purpose and the establishment of the DNBR limitation setpoints.*

Response to RAI 24:

The limitation functions are plant functions designed to intercede and produce a partial reactor trip and turbine runback whenever the plant nears the protection system (PS) trip setpoints. This brings the plant into a controlled mid-power state (~50% of rated thermal power), and permits the plant operator to make an assessment of the root cause. The limitation functions increase plant availability because they help avoid full reactor trips in situations where corrective actions can be performed quickly.

The limitation functions are discussed in Section 2.1.2 of ANP-10287P, and the synthesis of the function is shown in Figure 6-1 and the revised Figure 3-1. Section 9.6 of the topical report discusses the methods used to establish [ ]

The limitation function setpoints are based on operational constraints [ ] As such, additional margin between the limitation and protection setpoints may be implemented.

With regard to the scope of the ANP-10287P methodology topical, there are specific limitation functions dedicated to the Low DNBR and High LPD Channels. The calculated departure from nucleate boiling ratio (DNBR) and linear power density (LPD) inputs to the DNBR and LPD limitation functions are not independently synthesized, but are exactly the same inputs used in the PS functions. The online calculated DNBR and LPD values are independently assessed against both the limitation and trip setpoints. [ ] If the limitation function setpoint is reached, there are no restrictions placed on the reactor trip. Therefore, if the calculated online DNBR or LPD continues to degrade and reaches the PS setpoint (after reaching the limitation function setpoint), a full reactor trip will occur.

Although the limitation function signals for DNBR and LPD are completely formulated in the safety grade protection system, the partial trip actuation signals are sent to the control grade reactor control, surveillance, and limitation system. For these reasons, they are considered a control grade system.

[

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Figure 3-1 was revised to show the implementation of the DNBR limitation functional logic (consistent with Figure 6-1, which already shows the high linear power density (HLPD) limitation function logic). Additionally, Figure 6-1 was revised to fix font issues and [

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RAI-25. *The P2 interlock is not shown in Figures 3-1, 3-2, and 3-3, though it is discussed in Section 3.2.7. It is not clear from the text that the P2 interlock applies to all cases of generating a low DNBR trip signal. As appropriate, the P2 interlock should be included in Figures 3-1, 3-2, and 3-3 for clarity. Figure 6-1 illustrates a similar link to the P2 interlock.*

Response to RAI 25:

[

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RAI-26. *Figure 6-1 and 7-1 show 72 SPND inputs into each protection division. The text in Section 6.2.3 indicates that the SPNDs are apportioned among the four divisions, which would seem to indicate only 18 SPND outputs are processed by each division. Clarify in the text and the figures the actual SPND inputs to each division. If the output of all 72 SPNDs is input into each protection division, discuss the independence between divisions with respect to the high LPD LSSS trip and LCO actions.*

Response to RAI 26:

ANP-10281P, "U.S. EPR Digital Protection System Topical Report," Section 7.2 describes the concept of acquisition and distribution of self-powered neutron detector (SPND) measurements between the protection system (PS) divisions. An illustration of this concept is provided in Figure 7-2 of the protection system (PS) topical report.

Eighteen SPND measurements (three strings of six detectors) are initially acquired by the remote acquisition units (RAU) in each division of the PS. The RAUs in each division distribute their 18 measurements to acquisition and processing units (APU) in all four PS divisions. Hence, the APU in each division receive all 72 SPND measurements for processing. The high linear power density (HLPD) and low departure from nucleate boiling ratio (LDNBR) reactor trip calculations are performed in the APUs in each division. The calculation in each division is redundant to the same calculation being performed in each of the other three divisions.

The protection system functionality and architecture satisfy the IEEE Std. 603-1998, Clause 5.6.1 requirement for independence between redundant portions of a safety system. To achieve independence between PS divisions, the following three measures are implemented:

- Physical separation
- Electrical isolation
- Communications independence

Each PS division is physically separated from the other divisions by its location in a separate Safeguards Building. In this way, internal hazards such as fire and flooding, can only affect one division.

As described above, the SPND measurements are distributed from the RAUs to the APUs in other PS divisions. Electrical isolation is provided on these connections between PS divisions by using fiber optic cabling.

Data communication is used to transfer information between the PS divisions. Communication isolation is required so that a failure in one division cannot prevent correct operation of another division through inter-divisional communication channels. Section 12 of ANP-10281P describes the measures taken to establish communications independence between the PS divisions.

The fact that the same 72 SPNDs are used for processing within each of the four divisions is not related to the concept of independence. No specific group or collection of SPNDs is redundant to another specific group of SPND; therefore, independence requirements do not apply to the SPNDs or their measurements. Instead, the adequacy of the U.S. EPR SPND acquisition and distribution concept is demonstrated by compliance to the single failure criterion.

No single SPND failure or single PS equipment failure results in the inability to perform the HLPD or LDNBR reactor trip functions. A series of graduated reactor trip setpoints accommodate up to five failed SPNDs for the HLPD function, and any number of failed SPNDs up to five strings for the LDNBR function. U.S. EPR FSAR, Tier 2, Section 7.2 contains a failure modes and effects analysis for the reactor trip functions of the PS, and concludes that the functions are performed for any credible single failure. Additionally, an automatic reactor trip function is implemented in the case of seven or more invalid SPND measurements. This trip provides protection in case RAU1 is taken out of service for testing and a single failure occurs in RAU2 within the same division, resulting in 18 invalid SPND measurements. Refer to the response to RAI 27 for further details regarding this additional functionality.

The HLPD limiting conditions for operations actions are non-safety-related functions and do not require demonstration of independence.

RAI-27. *In a separate staff RAI, a possible mechanism is proposed to fail 25% of the core SPNDs (18) if RAU1 is taken out of service for testing and a single failure occurs in RAU2 within the same division. The failed-SPND logic for the in-core setpoint methodology (ANP-10287P) assumes that failures of 5 or more SPNDs are covered by the same 5-failed-SPND setpoint. It is not clear from the document that this methodology would cover such a large number of failures (18). The staff notes that, since the 18 failures are deterministic (a complete division) and their location is known and homogeneously distributed around the core, the 5+ failed-SPND setpoint could indeed be conservative. Demonstrate by analysis of one particular example that the 5+ failed-SPND setpoint is conservative if all 18 SPNDs in a division fail.*

Response to RAI 27:

The large number of self-powered neutron detector (SPND) failures that the remote acquisition units (RAU) failure scenario presents was not included in the scope of the original analysis conducted to establish the incore trip setpoints. Accordingly, an analysis of a particular example has not been analyzed.

Protection for the RAU failure scenario, and other scenarios that lead to numbers of SPND failures outside of the scope of the current safety analysis, is provided by an automatic reactor trip function that has been recently added to the system. This function initiates a reactor trip when a specific number SPND signals become unavailable. The reactor trip is set to occur when seven or more invalid SPND measurements are reported. This setpoint is based on assumptions in the safety analyses for certain transients that rely on the incore trip functions, and on the extent to which SPND failures have been considered in the setpoint determination analysis.

RAI-28. *In response to RAI #1, AREVA states that "Mixed core methodology is not being proposed for the U.S. EPR; therefore, AREVA NP is not requesting NRC review and approval of this methodology for use for mixed cores." Please confirm the intention of AREVA to design EPR allowing only one type of fuel to be used in the core.*

Response to RAI 28

The methodology documented in ANP-10287P does not currently consider combinations of fuel assemblies with dissimilar hydraulic characteristics. The following proposed wording, for use as a restriction in the safety evaluation report, would identify the limitation of the method while allowing for fuel assembly modifications that do not impact the present thermal margin and the core flow distribution among fuel assemblies:

"The setpoint methodology documented in ANP-10287P, Revision 0 (Reference 1) is only applicable to cores that consist entirely of hydraulically equivalent fuel assemblies (i.e., assemblies that have the same pressure drop characteristics and flow geometries)."

RAI-29. *The response to RAI #14 does not provide any experimental data for pin powers that is sufficient to justify the use of the quoted pin power uncertainty for exposed fuel. The only data provided thus far consists of predictions of measurements of fresh fuel which is not applicable to exposed fuel. Therefore, please supplement your existing database by providing fuel pin gamma scans (or equivalent) to justify the application of the pin power uncertainty to both fresh and exposed fuel.*

Response to RAI 29:

The local peaking factor uncertainty for pressurized water reactors (PWR) has historically been defined based on critical experiments with fresh fuel. Fuel vendors both in the U.S. and Europe have used this approach. The information is also applicable to exposed fuel, in that it reflects the impact of guide tubes (or instrument tubes) and the influence of flux gradients across the fuel assembly. The information does not reflect any uncertainties in the exposure accumulation on a rod-by-rod basis; this has historically been treated as negligible. This approach has been previously approved by the U.S. NRC for the methodologies of fuel vendors and for all currently operating PWRs in the U.S.

AREVA NP is not aware of experiments where PWR exposed fuel has been gamma scanned to define uncertainties in the local peaking factor. This is because the use of critical experiments with fresh fuel is adequate to define the local peaking factor uncertainty for both fresh and exposed fuel. Gamma scans have been done on exposed fuel for boiling water reactors (BWR) in order to capture the effects of voids and the use of inserted control blades at full power during the cycle depletion. The gamma scans of BWR fuel do not cover a full range of burnups, but are done at end-of-cycle (EOC) conditions (again only for one burnup state). BWR gamma scans only identify the average rod peaking distribution that exists at the EOC for the selected assembly.

Therefore, in compliance with established practices, the response to RAI #14 (ANP-10287Q1P) is based on critical experiments involving fresh fuel and exposed fuel measured reaction rate data taken at power from many plants, cycles, and at numerous times during the cycle. Unlike BWR cores where the incore detectors are located between assemblies, incore detectors in PWR cores are located internal to the assembly and see the localized peaking response from surrounding rods. This point was documented in the response to NRC question #14. Consequently, the ability of PWR incore detectors to see localized peaking for a significantly large number of locations in the core (axially and radially), and for significantly large surveillance intervals, provides a measurement database that will identify the potential existence of abnormal local peaking. Additionally, the uncertainty analysis is developed for the particular code and cross-section library being used compared to this measured data, which is also part of the established practice for this evaluation. Any uncertainty associated with calculated peaking is summarily reflected in the derived uncertainty based on a large quantity of measured data.

ANP-10287 Markups

Markup pages:
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2.0 REGULATORY REQUIREMENTS

The LSSSs and LCOs that are the subject of this report are designed for safe operation in accordance with the specifications outlined in the 10 CFR 50, Appendix A (Reference 1). In particular, General Design Criteria (GDC) II:10 and II:13 (multiple fission barrier protection), and Criteria III:20, III:25, and III:29 (protection and reactivity control systems) stipulate that the SAFDLs should be protected by these functions during normal operation and anticipated operational occurrences (AOO). The SAFDLs are experimental or analytical limits on the fuel and cladding which preclude fuel damage, set conservatively with respect to the safety limits. The SAFDLs germane to the incore based trips for the U.S. EPR are the following:

1. Departure from nucleate boiling
2. Fuel centerline melt

The setpoint analysis must demonstrate that, with 95 percent confidence, the probability of violating either of the SAFDLs is less than or equal to 5 percent (i.e., satisfying the 95/95 criterion).

The DNB acceptance criterion is given on page 4.4-5 of Section 4.4 of Reference 2 and is as follows:

> *Standard Review Plan (SRP) Section 4.2 specifies the acceptance criteria for the evaluation of fuel design limits. One criterion provides assurance that there is at least a 95-percent probability at the 95-percent confidence level that the hot fuel rod in the core does not experience a DNB or transition condition during normal operation or AOOs.*

Section 4.2.II, SRP Acceptance Criteria, 1.B.IV of Reference 2 also stipulates that fuel centerline melting is not permitted for AOOs in order to preclude fuel failure.

Figure 3-1 Low DNBR Channel Symmetric Trip Scenario

Figure 6-1 High LPD Channel LSSS Functional Diagram

11. [ ]

12. Regulatory Guide 1.126, Revision 1, "An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification," U.S. Nuclear Regulatory Commission, March 1978.

13. [ ]

14. [ ]

15. [ ]

16. [ ]

17. ANP-10263PA, *Codes and Methods Applicability Report for U.S. EPR*, AREVA NP, Inc., August 2007.

18. EMF-96-029(P)(A), Volumes 1-2 and Attachment, *Reactor Analysis System for PWRs*, Siemens Power Corporation, January 1997.

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A.1.3 Testing Approach for Multiple Setpoints

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E.4 Extrapolation of the Node Fluxes

After the optimal fluxes have been calculated in the instrumented nodes, the fluxes in the remaining non-instrumented nodes are determined. The flux extrapolation is calculated by the physical models involved in the NEM method and not by a purely mathematical variance minimization approach. Therefore fuel assembly (FA) quantities like cross sections, flux and burnup gradients remain unchanged. By means of the optimal neutron fluxes in the instrumented FAs the nodal balance equation is modified and the flux distribution for the non-instrumented nodes n is determined by solving the following equation (see Section 4.2 of Ref. 18):