ArevaEPRDCPEm Resource

From: Pederson Ronda M (AREVA NP INC) [Ronda.Pederson@areva.com]
Sent: Thursday, March 19, 2009 12:07 PM
To: Getachew Tesfaye
Cc: BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); PORTER Thomas (EXT)
Subject: Response to U.S. EPR Design Certification Application RAI No. 103, Supplement 1
Attachments: RAI 103 Supplement 1 Response US EPR DC.pdf

Getachew,

AREVA NP Inc. (AREVA NP) provided responses to 34 of the 82 questions of RAI No. 103 on November 26, 2008. The attached file, “RAI 103 Supplement 1 Response US EPR DC.pdf” provides technically correct and complete responses to 11 of the remaining 48 questions, as committed.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the response to RAI 103 Questions 16-193, 16-194, and 16-195.

The following table indicates the respective pages in the response document, “RAI 103 Supplement 1 Response US EPR DC.pdf,” that contain AREVA NP’s response to the subject questions.

Question #	Start Page	End Page
RAI 103 — 16-137	2	3
RAI 103 — 16-185	4	4
RAI 103 — 16-193	5	8
RAI 103 — 16-194	9	10
RAI 103 — 16-195	11	11
RAI 103 — 16-196	12	13
RAI 103 — 16-198	14	15
RAI 103 — 16-203	16	17
RAI 103 — 16-204	18	18
RAI 103 — 16-205	19	19
RAI 103 — 16-206	20	21

The schedule for technically correct and complete responses to the remaining 37 questions is unchanged and provided below:

Question #	Response Date
RAI 103 — 16-138	June 30, 2009
RAI 103 — 16-144	June 30, 2009
RAI 103 — 16-145	June 30, 2009
RAI 103 — 16-146	June 30, 2009
RAI 103 — 16-147	June 30, 2009
RAI 103 — 16-148	June 30, 2009
RAI 103 — 16-149	June 30, 2009
RAI 103 — 16-150	June 30, 2009
RAI 103 — 16-151	June 30, 2009
RAI 103 — 16-153	June 30, 2009
RAI 103 — 16-154	June 30, 2009
RAI 103 — 16-155	June 30, 2009
RAI 103 — 16-156	June 30, 2009
RAI 103 — 16-158	June 30, 2009

RAI 103 — 16-159	June 30, 2009
RAI 103 — 16-160	June 30, 2009
RAI 103 — 16-162	June 30, 2009
RAI 103 — 16-163	June 30, 2009
RAI 103 — 16-164	June 30, 2009
RAI 103 — 16-165	June 30, 2009
RAI 103 — 16-168	June 30, 2009
RAI 103 — 16-169	June 30, 2009
RAI 103 — 16-172	June 30, 2009
RAI 103 — 16-173	June 30, 2009
RAI 103 — 16-175	June 30, 2009
RAI 103 — 16-176	June 30, 2009
RAI 103 — 16-177	June 30, 2009
RAI 103 — 16-178	June 30, 2009
RAI 103 — 16-179	June 30, 2009
RAI 103 — 16-180	June 30, 2009
RAI 103 — 16-181	June 30, 2009
RAI 103 — 16-182	June 30, 2009
RAI 103 — 16-183	June 30, 2009
RAI 103 — 16-186	June 30, 2009
RAI 103 — 16-189	June 30, 2009
RAI 103 — 16-190	June 30, 2009
RAI 103 — 16-191	June 30, 2009

Sincerely,

Ronda Pederson
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From: Pederson Ronda M (AREVA NP INC)
Sent: Wednesday, November 26, 2008 1:40 PM
To: 'Getachew Tesfaye'
Cc: PORTER Thomas (EXT); BENNETT Kathy A (OFR) (AREVA NP INC); DUNCAN Leslie E (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 103 (1270), FSAR Ch. 16

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 103 Response US EPR DC.pdf" provides technically correct and complete responses to 34 of the 82 questions.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the response to RAI 103 Questions16-129, 16-130,16-131,16-132,16-133,16-134, 16-135, 16-136, 16-139, 16-140, 16-141, 16-142, 16-143, 16-152, 16-157, 16-161, 16-166, 16-167, 16-170, 16-171, 16-174, 16-187, 16-192, 16-200, 16-202, and 16-207.

The following table indicates the respective page(s) in the response document, “RAI 103 Response US EPR DC.pdf,” that contain AREVA NP’s response to the subject questions.

Question #	Start Page	End Page
RAI 103 — 16-128	2	2
RAI 103 — 16-129	3	3
RAI 103 — 16-130	4	4
RAI 103 — 16-131	5	5
RAI 103 — 16-132	6	6
RAI 103 — 16-133	7	7
RAI 103 — 16-134	8	8
RAI 103 — 16-135	9	9
RAI 103 — 16-136	10	10
RAI 103 — 16-137	11	11
RAI 103 — 16-138	12	12
RAI 103 — 16-139	13	13
RAI 103 — 16-140	14	14
RAI 103 — 16-141	15	15
RAI 103 — 16-142	16	16
RAI 103 — 16-143	17	17
RAI 103 — 16-144	18	18
RAI 103 — 16-145	19	19
RAI 103 — 16-146	20	20
RAI 103 — 16-147	21	21
RAI 103 — 16-148	22	22
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RAI 103 — 16-164	41	41
RAI 103 — 16-165	42	42
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RAI 103 — 16-209	93	93

A complete answer is not provided for 48 of the 82 questions. The schedule for a technically correct and complete response to this question is provided below.

Question #	Response Date
RAI 103 — 16-137	June 30, 2009
RAI 103 — 16-138	June 30, 2009
RAI 103 — 16-144	June 30, 2009
RAI 103 — 16-145	June 30, 2009
RAI 103 — 16-146	June 30, 2009
RAI 103 — 16-147	June 30, 2009
RAI 103 — 16-148	June 30, 2009
RAI 103 — 16-149	June 30, 2009
RAI 103 — 16-150	June 30, 2009
RAI 103 — 16-151	June 30, 2009
RAI 103 — 16-153	June 30, 2009
RAI 103 — 16-154	June 30, 2009

RAI 103 — 16-155	June 30, 2009
RAI 103 — 16-156	June 30, 2009
RAI 103 — 16-158	June 30, 2009
RAI 103 — 16-159	June 30, 2009
RAI 103 — 16-160	June 30, 2009
RAI 103 — 16-162	June 30, 2009
RAI 103 — 16-163	June 30, 2009
RAI 103 — 16-164	June 30, 2009
RAI 103 — 16-165	June 30, 2009
RAI 103 — 16-168	June 30, 2009
RAI 103 — 16-169	June 30, 2009
RAI 103 — 16-172	June 30, 2009
RAI 103 — 16-173	June 30, 2009
RAI 103 — 16-175	June 30, 2009
RAI 103 — 16-176	June 30, 2009
RAI 103 — 16-177	June 30, 2009
RAI 103 — 16-178	June 30, 2009
RAI 103 — 16-179	June 30, 2009
RAI 103 — 16-180	June 30, 2009
RAI 103 — 16-181	June 30, 2009
RAI 103 — 16-182	June 30, 2009
RAI 103 — 16-183	June 30, 2009
RAI 103 — 16-185	March 19, 2009
RAI 103 — 16-186	June 30, 2009
RAI 103 — 16-189	June 30, 2009
RAI 103 — 16-190	June 30, 2009
RAI 103 — 16-191	June 30, 2009
RAI 103 — 16-193	March 19, 2009
RAI 103 — 16-194	March 19, 2009
RAI 103 — 16-195	March 19, 2009
RAI 103 — 16-196	March 31, 2009
RAI 103 — 16-198	June 30, 2009
RAI 103 — 16-203	March 31, 2009
RAI 103 — 16-204	March 31, 2009
RAI 103 — 16-205	March 31, 2009
RAI 103 — 16-206	March 31, 2009

Sincerely,

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From: Getachew Tesfaye [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Tuesday, October 28, 2008 2:00 PM
To: ZZ-DL-A-USEPR-DL
Cc: Joseph DeMarshall; Michael Marshall; Peter Hearn; Joseph Colaccino; John Rycyna
Subject: U.S. EPR Design Certification Application RAI No. 103 (1270), FSARCh. 16

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on October 20, 2008, and on October 28, 2008, you informed us that the RAI is clear and no further clarification is needed. As a result, no change is made to the draft RAI. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period so that the staff can assess how this information will impact the published schedule.

Thanks,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP
(301) 415-3361

Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 316

Mail Envelope Properties (5CEC4184E98FFE49A383961FAD402D31C2E735)

Subject: Response to U.S. EPR Design Certification Application RAI No. 103, Supplement 1
Sent Date: 3/19/2009 12:07:03 PM
Received Date: 3/19/2009 12:07:07 PM
From: Pederson Ronda M (AREVA NP INC)

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Tracking Status: None

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MESSAGE	10121	3/19/2009 12:07:07 PM
RAI 103 Supplement 1 Response US EPR DC.pdf		320047

Options
Priority: Standard
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Response to

Request for Additional Information No. 103, Supplement 1

10/28/2008

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 16 - Technical Specifications
Application Section: 16

QUESTIONS for Technical Specification Branch (CTSB)

Question 16-137:

LCO 3.3.1, Protection System

Provide a technical justification for the omission of safety-related Reactor Trip initiation signals in Table 3.3.1-2, Section A (Reactor Trip).

Table 3.3.1-2, Section A, does not identify the following safety-related Reactor Trip initiation signals:

- “Safety Injection System (SIS) Actuation”
- “Emergency Feedwater System (EFWS) Actuation”
- “Manual RT signals from SICS”

Each of these actuation signals are indentified in the FSAR, Section 7.2.1.2 (Pg 7.2-3) as safety-related Reactor Trip Initiation Signals. Provide a technical justification for not including these signals in Table 3.3.1-2, or revise Table 3.3.1-2, accordingly.

This technical justification is needed to ensure the accuracy and completeness of the Reactor Trip Initiation Signals referenced in the EPR GTS, Table 3.3.1-2.

Response to Question 16-137:

The required content of the Technical Specifications is specified in 10 CFR 50.36. The U.S. EPR Protection System and its reactor trip isolation signals satisfy Criterion 3 of 10 CFR 50.36:

> "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."

As discussed in the Final Policy Statement on Technical Specifications Improvements for Nuclear Power Plants (FR Doc. 93-17344):

> "Discussion of Criterion 3: A third concept in the adequate protection of the public health and safety is that in the event that a postulated Design Basis Accident or Transient should occur, structures, systems, and components are available to function or to actuate in order to mitigate the consequence of the Design Basis Accident or Transient. Safety sequence analyses or their equivalent have been performed in recent years and provide a method of presenting the plant response to an accident. These can be used to define the primary success paths.
>
> A safety sequence analysis is a systematic examination of the actions required to mitigate the consequences of events considered in the plant's Design Basis Accident and Transient analyses, as presented in Chapters 6 and 15 of the plant's FSAR (or equivalent chapters). Such a safety sequence analysis considers all applicable events, whether explicitly or implicitly presented. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criteria), so that the plant response

to Design Basis Accidents and Transients limits the consequences of these events to within the appropriate acceptance criteria.

It is the intent of this criterion to capture into Technical Specifications only those structures, systems, and components that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path for a particular mode of operation does not include backup and diverse equipment (e.g., rod withdrawal block which is a backup to the average power range monitor high flux trip in the startup mode, safety valves which are backup to low temperature overpressure relief valves during cold shutdown)."

U.S. EPR FSAR Tier 2, Table 15.0-7, "Reactor Trip Setpoints and Delays Used in the Accident Analysis" and Table 15.0-8, "Engineered Safety Features Functions Used in the Accident Analysis," list the safety-classified protection and safety systems credited in the accident analyses. As implied by their absence from these tables, the reactor trips on safety injection system (SIS) actuation, emergency feedwater system (EFWS) actuation, or the manual reactor trip are not credited in the safety analysis. Therefore, they are not included in the Technical Specifications.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-185:

LCO 3.3.1, Protection System

Provide additional information and any corrections necessary to clarify the EPR GTS, Table 3.3.1-2, LTSP values for ESFAS Functions B.11.b and B.11.c.

The EPR GTS, Table 3.3.1-2, ESFAS Function B.11.b (CVCS Charging Line Isolation on ADM at Shutdown Condition, RCP not operating), specifies a LTSP value of 927 ppm. The EPR GTS, Table 3.3.1-2, ESFAS Function B.11.c (CVCS Charging Line Isolation on ADM at Standard Shutdown Conditions), specifies a LTSP value referenced in Footnote (d), "[a]s specified in the COLR." Provide an explanation as to why the LTSP values for ESFAS Functions B.11.b and B.11.c are not both specified as either specific valves or Footnotes referencing the COLR.

This additional information is needed to ensure the accuracy, and completeness of the EPR GTS.

Response to Question 16-185:

The chemical and volume control system (CVCS) charging line isolation on anti-dilution mitigation (ADM) at shutdown condition—reactor coolant pump (RCP) not operating—limiting trip setpoint (LTSP) is a fixed value. The setpoint is low enough to provide an operating envelope that prevents unnecessary isolations but high enough to mitigate a dilution event in the shutdown condition where the RCPs are not in operation. The LTSP of 927 ppm will not change from cycle to cycle.

The CVCS charging line isolation on ADM at standard shutdown conditions LTSP is a function of core reactivity conditions which can vary between cycles and, as such, will be provided in the core operating limits report (COLR).

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-193:

LCO 3.3.1, Protection System

Provide additional information in the EPR Bases to describe the overall approach to Surveillance Requirement Testing with respect to Protection System Instrumentation. Include an explanation as to how that approach ensures that all Reactor Trip and ESFAS functions in the EPR GTS, Table 3.3.1-2, are adequately tested. Provide a figure which depicts a "Summary of Protection System Testing."

The Protection System (PS) is an integrated digital Reactor Protection System and ESF Actuation System. The PS detects plant conditions that indicate the occurrence of AOOs and postulated events, and actuates safety-related process systems required to mitigate an event. The PS maximizes use of the TELEPERM XS (TXS) digital I&C platform design features, including continuous on-line self-testing and diagnostics that allow early detection of failure.

Additional information is needed that describes the approach to PS Surveillance Requirement Testing with respect to the following specifics:

a. The EPR GTS, Table 3.3.1-1, use of common components/sensors by the PS to support all Reactor Trip functions, ESFAS functions and Permissives. Surveillance Requirements are specified in the EPR GTS, Table 3.3.1-1, for components/sensors only. Surveillance Requirements are not specified in EPR GTS, Table 3.3.1-2, for any of the individual Reactor Trip or ESFAS functions. This is a deviation from NUREG-1431 (WOG STS). It is unclear how Surveillance Requirement Testing specified at the component/sensor level ensures that each of the Reactor Trip Functions, ESFAS Functions, and associated permissives are properly surveilled.
b. The absence of Reactor Trip and ESFAS System "Response Time Testing." Response Time Testing ensures actuation response times are less than or equal to the maximum values assumed in the accident analysis.
c. The absence of Channel Checks. Channel Check Surveillance Requirements associated with Reactor Trip and ESFAS Instrumentation have not been included. This is a deviation from NUREG-1431.
d. The transition from a "Channelized" concept to a "Divisional" concept as it relates to surveillance testing requirements applicable to the TXS digital I&C platform design, specifically:
    1. Performance of a "CALIBRATION" vice "CHANNEL CALIBRATION"
    2. Performance of a "DIVISION OPERATIONAL TEST" vice "CHANNEL OPERATIONAL TEST"
    3. Performance of an “ACTUATING DEVICE OPERATIONAL TEST” vice “TRIP ACTUATING DEVICE OPERATIONAL TEST”
    4. Addition of a "SENSOR OPERATIONAL TEST"
e. The absence of the "DIVISIONAL OPERATIONAL TEST" from the Surveillance Requirements of LCO 3.3.1. The "DIVISIONAL OPERATIONAL TEST" is specified in the EPR GTS, "Definitions" Section of 1.0, USE AND APPLICATIONS (pg 1.1-2).

This additional information is needed to ensure that all Reactor Trip and ESFAS functions are being properly surveilled under the TELEPERM XS digital I&C platform, especially considering the fact that these functions (including permissives) share common components/sensors.

Response to Question 16-193:

By letter NRC:99:056, dated December 28, 1999, Siemens submitted report EMF-2341 (P), "Generic Strategy for Periodic Surveillance Testing of TELEPERM XS Systems in U.S. Nuclear Generating Stations," for staff review. By letter NRC:00:017 dated March 3, 2000, Siemens provided additional clarification on recommended periodic surveillance test requirements for TXS applications. These are the references cited in Section 4.2, "Surveillance Testing of the TXS System", in the May 5, 2000 Safety Evaluation, Acceptance for Referencing of Licensing Topical Report EMF-2110(NP), Revision 1, "TELEPERM XS: A Digital Reactor Protection System" (TAC No. MA1983).

Figure 16-193-1 summarizes the testing of the U.S. EPR Protection System, which provides a correlation between specific sections of Siemens Topical Report EMF-2341(P), "Generic Strategy for Periodic Surveillance Testing of TELEPERM XS Systems in U.S. Nuclear Generating Stations," and the surveillance testing specified in U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.3.1 "Protection System (PS)". Also refer to the response to RAI 122, Question 16-243, for a discussion of the differences between definitions used in the Standard Technical Specifications for Westinghouse Plants (NUREG-1431) and the U.S. EPR Generic Technical Specifications.

With regards to the specific additional information requested in the Question 16-193:

a. There is no fundamental difference between a function-based surveillance testing approach and a component-based approach. Since a function is performed by components, a component-based approach only specifies an additional level of detail by defining which specific surveillances apply to which specific components. The U.S. EPR FSAR Tier 2, Technical Specifications Section 3.3.1 are grouped into a set of four functional components (See Technical Specifications Bases Page B 3.3.1-4):

   - Sensors (which include the associated signal conditioning),
   - Manual actuation switches,
   - Signal processors, and
   - Actuation devices.

   Performance of the surveillance testing specified for each of these components provides assurance that the functions supported by these components are operable.

   In RAI 103 Question 16-138, NRC requested AREVA provide a technical justification for the omission of permissive signals from LCO 3.3.1. Required Actions to address the impact of permissives will be addressed in response to that question.

b. Response time testing will be added to U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 1.1 "Definitions," Section 3.3.1 "Protection System (PS)," and Technical Specification Bases Section B 3.3.1 "Protection System (PS)".

c. Refer to the response to RAI 103, Question 16-197 for a discussion of Channel Checks.

d. Refer to the response to RAI 122, Question 16-243, for a discussion of a "Channelized" concept versus a "Divisional" concept and a discussion of the differences between definitions used in NUREG-1431 and the U.S. EPR Generic Technical Specifications.

e. Refer to the response to RAI 122, Question 16-243, for a discussion of a DIVISIONAL OPERATIONAL TEST and a discussion of the differences between definitions used in NUREG-1431 and the U.S. EPR Generic Technical Specifications.

The Surveillance Requirements Section for U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section B 3.3.1 "Protection System (PS)" contains the same level of detail provided in the Surveillance Requirements Sections in the Standard Technical Specifications for Westinghouse Plants (NUREG-1431), Bases 3.3.1 "Reactor Trip System (RTS) Information" and 3.3.2 "Engineered Safety Feature Actuation System (ESFAS) Instrumentation".

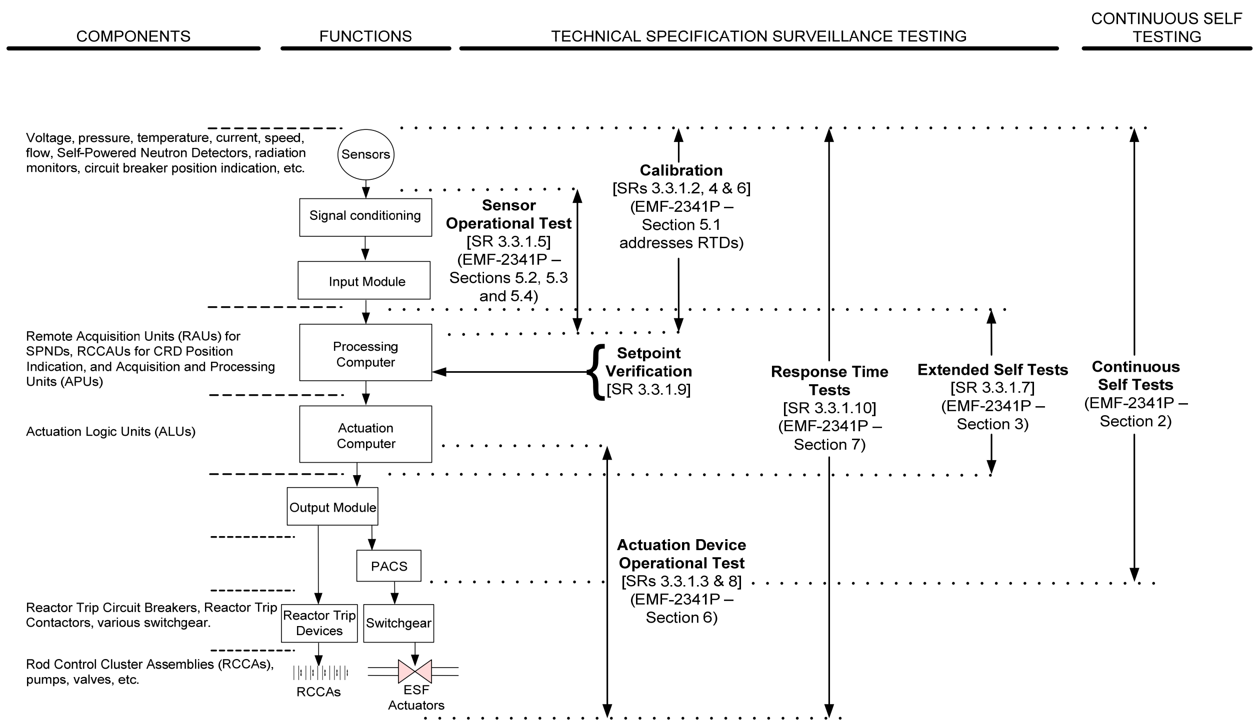
FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 1.1, "Definitions," Section 3.3.1 "Protection System (PS)" and U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section B 3.3.1 "Protection System (PS)" will be revised as described in the response and indicated on the enclosed markup.

Figure 16-193-1

SUMMARY OF PROTECTION SYSTEM TESTING

(Comparison with EMF-2341P, Generic Strategy for Periodic Surveillance Testing of TELEPERM XS Systems)



Question 16-194:

LCO 3.3.1, Protection System

Provide additional information and any corrections necessary to better describe the Limiting Trip Setpoint (LTSP) values and references to “SENSOR OPERATIONAL TEST” and “channel” in the EPR Bases.

The EPR Bases, Background Section (pg B 3.3.1-3, first paragraph), states that the “LTSP is a predetermined setting for a protective device chosen to ensure automatic actuation prior to the process variable reaching the Analytical Limit and thus ensuring that the SL would not be exceeded.” The last paragraph on page B 3.3.1-3 states that “however, there is also some point beyond which the device would have not been able to perform its function due, for example, to greater than expected drift. This LTSP specified in Table 3.3.1-2 is the least conservative value of the as-found setpoint that a channel can have during testing such that a channel is OPERABLE if the trip setpoint is found conservative with respect to the Allowable Value during a SENSOR OPERATIONAL TEST (SOT). As such, the Allowable Value differs from the LTSP by an amount greater than or equal to the expected instrument channel uncertainties, such as drift, during the surveillance interval.” The phrase “this LTSP specified in Table 3.3.1-2 is the least conservative value” implies that the least conservative values of the LTSPs have been specified in the table. Would this statement be more accurate if it stated “this as-found value of the LTSP specified in Table 3.3.1-2 is the least conservative value.”

The reference to SENSOR OPERATIONAL TEST (SOT) should be replaced by DIVISION OPERATIONAL TEST (DOT) based on the definitions in the EPR GTS, Section 1.1, “1.0, USE AND APPLICATIONS.” The DOT definition states that the test “shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for division OPERABILITY such that the setpoints are within the necessary range and accuracy.” The SOT definition makes no reference to setpoints.

In the EPR Bases, Background Section (pg B 3.3.1-3 last paragraph), the two references to "channel" in the Bases statement should be replaced by the word "division." The TELEPERM XS digital I&C platform utilizes a "divisional approach" as opposed to "channelized approach.”

This additional information is needed to ensure the accuracy, completeness and consistency of the EPR Bases.

Response to Question 16-194:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section B 3.3.1 "Protection System (PS)" will be revised to more accurately reflect setpoint relationships and the suggested wording from Technical Specification Task Force (TSTF) Traveler TSTF-493, Revision 2.

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section B 3.3.1 "Protection System (PS)" will be revised to refer to "CALIBRATION" instead of "SENSOR OPERATIONAL TEST (SOT)". Refer to the response to RAI 122, Question 16-243, for a discussion of the differences between definitions used in NUREG-1431 and the U.S. EPR Technical Specifications. Refer to the response to Question 16-193 for a summary of Protection System testing.

The cited references to "channel" will be revised to refer to "division" in U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section B 3.3.1 "Protection System (PS)".

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section B 3.3.1 "Protection System (PS)" will be revised as described in the response and indicated on the enclosed markup.

Question 16-195:

LCO 3.3.1, Protection System

Provide additional information and any corrections necessary to justify the Self-Powered Neutron Detector (SPND) value specified in the EPR GTS, Table 3.3.1-1, for “Minimum Required for Functional Capability.”

The EPR GTS, Table 3.3.1-1, specifies 51 SPNDs as the “Minimum Required for Functional Capability.” In accordance with the EPR FSAR, Section 7.1.1.5.2, there are 72 SPNDs that continuously measure the neutron flux at given positions in the core to provide a three-dimensional flux distribution. Remote Acquisition Units (RAUs) in each division acquire one-fourth (18) of the total SPND measurements and distribute those measurements to APUs in all four divisions allowing for an accurate calculation over the whole core in each division. Provide additional information or corrections as necessary to address the minimum number of SPNDs required.

This additional information is needed to ensure accuracy, completeness and consistency within the EPR GTS and FSAR.

Response to Question 16-195:

The U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications and Technical Specifications Bases will be revised to specify 67 SPNDs as the “Minimum Required for Functional Capability.” Topical report ANP-10287P, Revision 0, “Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report,” AREVA NP Inc., November 2007 in section 5.4.6 shows how trip setpoints will be automatically changed to reflect up to five SPND failures.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Section 3.3.1 "Protection System (PS)" and U.S. EPR FSAR Tier 2, Chapter 16, Technical Specifications Bases Section B 3.3.1 "Protection System (PS)" will be revised as described in the response and indicated on the enclosed markup.

Question 16-196:

LCO 3.3.1, Protection System

Provide additional information and any corrections necessary to justify the Limiting Trip Setpoint (LTSP) time range specified in the EPR GTS, Table 3.3.1-2, for ESFAS Function B.10.a.

The EPR GTS, Table 3.3.1-2, ESFAS Function B.10.a (EDG Start on Degraded Grid Voltage), specifies a LTSP time range of ≥ 270 seconds and ≤ 300 seconds without a Safety Injection System actuation signal. On the basis of engineering judgment, this time frame appears to be excessive considering the consequences associated with an extended degraded voltage condition. Provide additional information or corrections as necessary to address the length of time specified.

This additional information is needed to ensure the accuracy of the EPR GTS.

Response to Question 16-196:

The limiting trip setpoint (LTSP) degraded grid voltage (DGV) w/o safety injection system (SIS) time range was selected utilizing the philosophy outlined in paragraph A.4, “Degraded Voltage Relay Time Delay Settings”, of IEEE Std 741-1997 (R2002), IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Stations.[1]

As observed in NRC IN 89-83, “Sustained Degraded Voltage on the Offsite Electrical Grid and Loss of Other Generating Stations as a Result of a Plant Trip”: “The offsite power system is the preferred and the most reliable source of power for nuclear plant safety systems.” [2] Regulation 10 CFR 50 Appendix A, General Design Criterion 17 states “Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear unit…” [3] However, the degraded voltage scheme itself actually increases the probability of a Loss of Offsite Power.[4] In the absence of a SIS actuation signal, the emergency power supply system (EPSS) buses are lightly loaded, voltage drop to the loads is reduced, and since few loads are connected to the EPSS buses, few loads are directly affected by any reduction in voltage.

For existing plants, the most limiting equipment for degraded voltage is frequently the magnetic contactor.[5] The contactor circuits are usually powered from the incoming bus feeder. In the

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1. IEEE Std 741-1997 (R2002), IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Stations.
2. NRC Information Notice 89-83, Sustained Degraded Voltage on the Offsite Electrical Grid and Loss of Other Generating Stations as a Result of a Plant Trip.
3. 10 CFR 50 Appendix A, General Design Criterion for Nuclear Power Plants, Criterion 17, Electric Power Systems.
4. Kueck, Attarian, Leake and Sims, “Risk Factors Regarding the Application of Degraded Voltage Relaying at Nuclear Generating Stations, IEEE Transactions on Energy Conversion, Vol 16, No. 4, December 2001.
5. Kueck, et al., “A Discussion of Degraded Relaying for Nuclear Generating Stations, IEEE, May 1998.

U.S. EPR, this is avoided by using DC-controlled contactors. Therefore, the remaining question is whether the few EPSS loads that may be energized during a non-SIS degraded voltage event would be damaged during the brief time permitted by the non-SIS degraded voltage time delay. Although numerous degraded voltage events have occurred in the industry, little or no operating experience is available to indicate measurable motor damage will occur during a brief (five minute) degraded voltage event. It also does not appear likely or even possible to maintain stable degraded grid conditions such that sustained voltages can be maintained just above the loss of voltage setpoint. In addition, most transmission operators have established contingency plans with base line voltage limits in the 5–10 percent range. Transmission operators also have special notification and mitigation protocols for nuclear plants. As an example, PJM transmission operators require notification within 15 minutes for contingency violations and immediately for actual voltage violations.[6] Thus, while it is possible for a voltage violation to last for several minutes, the combined actions of the on load tap changer and the transmission system operator preclude stable system voltages just above the loss of voltage setpoint.

Based on the design inputs and competing considerations outlined above, the non-SIS emergency diesel generator (EDG) voltage setpoint is a matter of engineering judgment rather than the result of exact calculation. AREVA NP has sampled some of the plants with unique non-SIS settings, and has found four plants (Catawba 1 & 2[7], McGuire 1 & 2[8]) with nominal setpoints of approximately ten minutes, Dresden Unit 2 with a setpoint of ≥ 279 seconds and ≤ 321 seconds[11] is approximately the same setting as proposed by the U. S. EPR, San Onofre Unit 2 in the 78-128 second range[9], and North Anna Unit 1 at 56 ±7 seconds.[10] A five minute setting does not appear unusual or excessive in light of the existing settings at operating plants. Furthermore, U.S. EPR FSAR Tier 2, Section 8.3.1.1.3 states that “the second time delay is sufficient to allow bus voltage to be restored by the EAT on-load tap changer” and “If the degraded condition exists at the end of the first time delay, an alarm will alert the operator to the condition so that corrective action can be taken.” One possible corrective action is to take manual control of the on-load tap changer. Any reduction in the time delay for the non-SIS DGV relay reduces the likelihood that this corrective action would be possible and effective.

Therefore, AREVA NP recommends no change to the proposed second level (non-SIS) EDG voltage time delay.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

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6. PJM Manual 03, Transmission Operations, Revision 32, October 3, 2008.
7. Catawba Docket Numbers 50-413/Operating License NPF-35 and 50-414/Operating License NPF52, ADAMS Accession Numbers ML052990150 and ML052990194.
8. McGuire Docket Numbers 50-369/Operating License Number NPF-9 and 50-370/Operating License Number NPF-17, ADAMS Accession Numbers ML052870418 and ML052870419.
9. San Onofre, Unit 2 Docket Number 50-361/Operating License NPF-10, ADAMS Accession Number ML053130316.
10. North Anna Unit 1, Docket Number 50-338/Operating License NPF-4, ADAMS Accession Number ML052990145.
11. Dresden Unit 2 Docket Numbers 50-237/Operating License Number DRP-19, ADAMS Accession Number ML052990131.

Question 16-198:

LCO 3.3.3, Remote Shutdown System (RSS)

Provide a technical justification for omission of the Channel Check Surveillance Requirement from LCO 3.3.3.

Performance of a CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison between the parameter indicated on one channel to a similar parameter on other channels. This surveillance is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviation between the two instrument channels could be an indication of excessive instrument drift in one of the channels or an indication of something more serious. CHANNEL CHECKS will detect gross channel failure and are therefore important in verifying that instrumentation continues to operate properly between CALIBRATIONS. This is a deviation from NUREG-1431, “Standard Technical Specifications Westinghouse Plants.”

This additional information is needed to ensure that Remote Shutdown System instrumentation is being properly surveilled and that deviations from STS are adequately justified.

Response to Question 16-198:

A CHANNEL CHECK for the U.S. EPR remote shutdown station (RSS) is not required for the following reasons:

- As discussed in the response to RAI 110 Question 16-197, extensive continuous self-testing is performed by the TELEPERM XS Digital Reactor Protection System that eliminates the need for a manual CHANNEL CHECK. This position has been reviewed and approved by NRC for the Protection System. This includes information regarding the primary RSS functions (i.e., core reactivity control, reactor coolant system (RCS) pressure, decay heat removal, and RCS inventory control).

- As discussed in the response to Question 16-215, the displays and controls at the RSS are functionally the same as the displays and controls normally used by the operator to achieve and maintain Mode 3 from the main control room. These displays and controls are continuously provided by the process information and control system (PICS) in the main control room. There is no separate and unique analog instrumentation located in the RSS which requires a surveillance.

- There is no basis for requiring a surveillance on instrumentation in the RSS when there is no surveillance for the same instrumentation in the main control room.

- The design basis of the RSS includes the required instrumentation and functions to achieve safe shutdown during a postulated fire or achieve safe shutdown during an event that could cause the control room to become uninhabitable coupled with a single failure. The degradation (which may be detected by a CHANNEL CHECK) or complete loss of an instrument (which may not be detected by a CHANNEL CHECK) as a result of the fire or single failure would not preclude the operator from placing and maintaining the unit in Mode 3. While the CHANNEL CHECK may be a recommended operational practice, it does not in itself satisfy Criterion 4 of 10 CFR 50.36(c)(2)(ii); nor is it required to demonstrate OPERABILITY of the RSS.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-203:

LCO 3.3.1, Protection System

Provide a technical justification for the omission of the "Source Range Neutron Flux" Reactor Trip function from the EPR GTS, LCO 3.3.1.

In accordance with NUREG-1431, "Standard Technical Specifications Westinghouse Plants," the Source Range Neutron Flux Reactor Trip function ensures protection against an uncontrolled RCCA bank withdrawal accident from a subcritical condition during startup. The applicable modes are Modes 2(d), 3(a), 4(a), and 5(a).

The EPR GTS, LCO 3.3.1, Protection System Reactor Trip functions "High Neutron Flux (Intermediate Range)" and "Low Doubling Time (Intermediate Range)," both protect against excessive reactivity additions during reactor startup from a subcritical or low power startup condition. Applicable modes associated with both of these functions are Modes 1(g), 2, 3(e). Although the Reactor Trip Functions are similar between NUREG-1431 and the EPR GTS, the mode applicability is different. The EPR Protection System does not provide protection against excessive reactivity additions from a subcritical condition during reactor startup in Modes 4 and 5. Provide a technical justification for this deviation from the STS and include any corrections necessary to resolve this apparent discrepancy.

This additional information is needed to ensure the accuracy and completeness of the EPR GTS.

Response to Question 16-203:

The U.S. EPR design includes protection system reactor trip functions "High Neutron Flux (Intermediate Range)", "Low Doubling Time (Intermediate Range)", and "High Neutron Flux Rate of Change (Power Range)," which protect against excessive reactivity additions during reactor startup from a subcritical or low power startup condition. The High Neutron Flux (Intermediate Range) and Low Doubling Time (Intermediate Range) reactor trips are required to be operable in Mode 1 below 10 percent Rated Thermal Power, Mode 2, and in Mode 3 with the reactor control, surveillance and limitation (RCSL) system capable of withdrawing a rod cluster control assembly (RCCA) or one or more RCCAs not fully inserted. The High Neutron Flux Rate of Change (Power Range) reactor trip is required to be operable in Modes 1 and 2 and in Mode 3 with the RCSL system capable of withdrawing a RCCA or one or more RCCAs not fully inserted. Excess reactivity additions can be postulated to occur by either RCCA withdrawal, boron dilution, or overcooling events.

Rod Cluster Control Assembly (RCCA) Withdrawal Events

RCCA withdrawal events are only possible if the reactor trip breakers are closed and the RCCAs are capable of being withdrawn. In Mode 3, the three reactor trips described previously are required whenever the RCSL is capable of withdrawing a RCCA or one or more RCCAs are not fully inserted. Above 568°F (Minimum temperature for criticality - Mode 3), with RCCAs capable of being withdrawn, the "High Neutron Flux (Intermediate Range)", "Low Doubling Time (Intermediate Range)", and "High Neutron Flux Rate of Change (Power Range)" are required to be operable. Below 568°F, the RCCA trip breakers will be open, and the RCCAs cannot be withdrawn. Therefore, RCCA withdrawal events are protected in Mode 3.

If the trip breakers are open, the RCCAs are not capable of being withdrawn, even considering an electrical fault or operator error. Thus, in Modes 4 and 5, RCCA withdrawal events are controlled by the following:

1. In Modes 4, and 5, plant procedures will restrict RCCAs from being capable of being withdrawn below the minimum temperature for criticality (568°F).

2. If the RCCAs are required to be exercised below 568°F in Modes 4 and 5, the RCS will be borated, by plant procedure, to a value greater than the ARO critical boron concentration. Alternatively, since it is not possible for the control banks and shutdown banks to be withdrawn simultaneously, it is sufficient to borate to a value greater than critical concentration with all shutdown banks out or all control banks out.

Items 1 and 2 above preclude the possibility of an inadvertent criticality by RCCA withdrawal in Modes 4 and 5.

Boron Dilution Events

Boron dilution events are protected by the chemical and volume control system (CVCS) charging line isolation function of the anti-dilution mitigation (ADM) system. ADM is required to operable in Modes 1, 2, 3, 4, and 5, and in Mode 6 with no reactor coolant pumps (RCP) in operation.

Overcooling Events

Overcooling events, with a negative moderator temperature coefficient, can also result in an excess reactivity addition. If sufficient positive reactivity is added, the reactor could go critical. An inadvertent criticality could result in challenging fuel design limits. The limiting overcooling events from shutdown are steam system piping failures and excess steam demands. The limiting steam system piping failures were analyzed from Modes 1 and 2, and in Mode 3 from the minimum temperature for criticality, and from the thermal hydraulic conditions associated with the P12 permissive. For the spectrum of steam system piping failures in Modes 1 and 2, the limiting case is from Mode 2 and predicts a return to power. In Mode 3, the results (from minimum temperature for criticality and the P12 Permissive) show that the reactor returns to a lower power level than the limiting case from Mode 2. Thus, from the standpoint of return to power and potential fuel failure, the steam system piping failure from Mode 3 is bounded by the analysis results for Modes 1 and 2. For Modes 4 and 5, sufficient shutdown margin exists such that the most limiting overcooling event does not result in a return to power.

The U. S. EPR protection system provides protection for excess reactivity addition events in all operating modes.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-204:

LCO 3.3.1, Protection System

Provide a technical justification for the omission of the "Overtemperature Delta T" Reactor Trip function from the EPR GTS, LCO 3.3.1.

In accordance with NUREG-1431, "Standard Technical Specifications Westinghouse Plants," an "Overtemperature Delta T" Reactor Trip function ensures that the design limit DNBR is met. The applicable modes are Modes 1 and 2.

The EPR GTS, LCO 3.3.1, Protection System Reactor Trip on "Low Departure from Nucleate Boiling Ratio," protects the fuel against the risk of departure from nucleate boiling during events that lead to a decrease in the DNBR value. The EPR GTS, Table 3.3.1-2 mode applicability is Mode 1 ($\geq$ 10 percent) for Functions A.1.a through A.1.e. Although the Reactor Trip functions are similar between NUREG-1431 and the EPR GTS, the mode applicability is different. The EPR Protection System does not provide protection against Low DNBR in either Mode 1 (< 10 percent power) or Mode 2. Provide a technical justification for this deviation from the STS and include any corrections necessary to resolve this apparent discrepancy.

This additional information is needed to ensure the accuracy and completeness of the EPR GTS.

Response to Question 16-204:

At low power levels, below the P2 permissive (≤ 10 percent Rated Thermal Power (RTP)), departure from nucleate boiling ratio (DNBR) is no longer the limiting thermal parameter. Before DNB is challenged at these low power levels, hot leg saturation occurs. Protection from hot leg saturation is provided by the low saturation margin reactor trip. This trip is active at power levels above the P5 permissive (10^{-5} percent RTP). Therefore, at or below 10 percent of RTP, thermal protection is afforded by the low saturation margin reactor trip, while above 10 percent, protection is afforded by the low DNBR reactor trip. Reactivity insertion events have been analyzed from sub-critical conditions and do not pose a credible threat to DNBR limits.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-205:

LCO 3.3.1, Protection System

Provide a technical justification for the omission of the "Overpower Delta T" Reactor Trip function from the EPR GTS, LCO 3.3.1.

In accordance with NUREG-1431, "Standard Technical Specifications Westinghouse Plants," an "Overpower Delta T" Reactor Trip function ensures the integrity of the fuel (i.e., no fuel pellet melting and less than 1 percent cladding strain) under all possible overpower conditions. The applicable modes are Modes 1 and 2.

The EPR GTS, LCO 3.3.1, Protection System Reactor Trip on "High Linear Power Density," protects the fuel against melting at the center of the fuel pellet during events which lead to an increase in the linear power density within the core. The EPR GTS, Table 3.3.1-2 mode applicability is Mode 1 (≥ 10 percent) for Function A.2. Although the Reactor Trip functions are similar between NUREG-1431 and the EPR GTS, the mode applicability is different. The EPR Protection System does not provide protection against fuel pellet melting in either Modes 1 (< 10 percent power) or Mode 2. Provide a technical justification for this deviation from the STS and include any corrections necessary to resolve this apparent discrepancy.

This additional information is needed to ensure the accuracy and completeness of the EPR GTS.

Response to Question 16-205:

At low power levels, below the P2 permissive (≤ 10 percent rated thermal power (RTP)), fuel centerline melt (FCM) is not a concern because there is no combination of neutronic peaking and core average power that can lead to a peak fuel centerline temperature greater than that experienced at full power conditions. Furthermore, there is an intermediate range (high neutron flux) reactor trip that will prevent any events from increasing the power level above 25 percent of rated thermal power to protect against reactivity insertion events such as an uncontrolled control bank withdrawal. Reactivity insertion events have been analyzed from sub-critical conditions and do not pose a credible threat to FCM limits.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 16-206:

LCO 3.3.1, Protection System (4507)

Provide the additional information and any changes needed regarding the EPR GTS, Table 3.3.1-2, and associated Bases, use of the term “Limiting Trip Setpoint” (LTSP) and the LTSP values specified.

The EPR GTS, Table 3.3.1-2, and Bases BACKGROUND Section (pg B 3.3.1-3, first paragraph), states that “the LTSP is a predetermined setting for a protective device chosen to ensure automatic actuation prior to the process variable reaching the Analytical Limit and thus ensuring that the SL would not be exceeded.”

The EPR Bases, Background Section (pg B 3.3.1-2, REVIEWER’S NOTE), states that the term “limiting Trip Setpoint (LTSP)” is generic terminology for the setpoint value calculated by means of the plant-specific setpoint methodology documented in a document controlled under 10 CFR 50.59. The term LTSP indicates that no additional margin has been added between the Analytical Limit and the calculated trip setting. Where margin is added between the Analytical Limit and trip setpoint, the term Nominal Trip Setpoint is preferred.” The EPR Bases, Background Section (pg B 3.3.1-3, last paragraph), states that "[t]his LTSP specified in Table 3.3.1-2 is the least conservative value of the as-found setpoint that a channel can have during testing such that the channel is OPERABLE if the trip setpoint is found conservative with respect to the Allowable Value during a SENSOR OPERATIONAL TEST (SOT).”

Although an approved EPR instrument setpoint methodology exits, it is unclear how a calculated LTSP value can be specified based on instrumentation uncertainties when those uncertainties would not ordinarily be determined until after completion of the detailed design. Provide a technical justification for use of the term LTSP and validate the LTSP values provided in the EPR GTS, Table 3.3.1-2. Include any discussions necessary to ensure a clear understanding of how the LTSP values specified ensure that an Allowable Value will not be exceeded and that an Analytical Limit will not be challenged.

This information is needed to ensure the completeness and accuracy of the EPR GTS, Table 3.3.1-2, and associated Bases.

Response to Question 16-206:

The limiting trip setpoint (LTSP) values provided in U.S. EPR FSAR Tier 2, Technical Specifications Table 3.3.1-2 are based on the analytical limits used in the U.S. EPR safety analysis shown in U.S. EPR FSAR Tier 2, Tables 15.0-7 and 15.0-8. The final calculated values will be determined later in the design process.

The trips/actuation functions listed in U.S. EPR FSAR Tier 2, Technical Specifications Table 3.3.1-2, "Acquisition and Processing Unit Requirements referenced from Table 3.3.1-1" are also listed in U.S. EPR FSAR Tier 1, Table 2.4.1-3, "Protection System Automatic Reactor Trips" and U.S. EPR FSAR Tier 1, Table 2.4.1-4, "Protection System Automatically Actuated Engineered Safety Features". U.S. EPR FSAR Tier 1, Table 2.4.1-9, "Protection System ITAAC" Item 4.6 states an analysis will be performed to verify that the protection system setpoints are determined using the documented methodology.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

U.S. EPR Final Safety Analysis Report Markups

1.1 Definitions

PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in FSAR Chapter 14, "Verification Programs"; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates and the low temperature overpressure protection setpoints, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.4, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)."
PROTECTION SYSTEM (PS) RESPONSE TIME	The PS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its PS actuation setpoint at the division sensor until the PS equipment is capable of performing its safety function (i.e., loss of stationary gripper coil voltage, the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 4590 MWt.
SENSOR OPERATIONAL TEST (SOT)	A SOT shall be the injection of a simulated or actual signal into the division as close to the sensor as practicable to verify OPERABILITY of all devices in the ~~input circuit~~division required for sensor OPERABILITY. The SOT shall include the verification of the accuracy and time constants of the analog input modules. The SOT may be performed by

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.7	Perform EXTENDED SELF TESTS.	24 months
SR 3.3.1.8	Perform ACTUATION DEVICE OPERATIONAL TEST.	24 months
SR 3.3.1.9	Verify setpoints properly loaded in APUs.	24 months
SR 3.3.1.10	-------------------------------NOTE-------------------------------- Neutron detectors are excluded from response time testing. --- Verify PS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

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Table 3.3.1-1 (page 1 of 4)
Protection System Sensors, Manual Actuation Switches,
Signal Processors, and Actuation Devices

COMPONENT	REQUIRED NUMBER OF SENSORS, SWITCHES, SIGNAL PROCESSORS, OR ACTUATION DEVICES	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	~~MINIMUMUM~~ MINIMUM REQUIRED FOR FUNCTIONAL CAPABILITY	CONDITION	SURVEILLANCE REQUIREMENTS
A. Sensors					
1. 6.9 kV Bus Voltage	3 per EDG	1,2,3,4,(a)	2 per EDG 16-193	O	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
2. Boron Concentration - Chemical and Volume Control System (CVCS) Charging Line	4	3(b),4(b),5,6	2	P	SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.10
3. Boron Temperature - CVCS Charging Line	4	3(b),4(b),5,6	2	P	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
4. CVCS Charging Line Flow	4	3(b),4(b),5(b)	2	P	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
5. Cold Leg Temperature (Narrow Range)	4	≥ 10% RTP	3	H	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
6. Cold Leg Temperature (Wide Range)	4	1,2(c)	3	J	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
	4	3,4,5,6(b)	2	P	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
7. Containment Pressure	4 per area	1,2,3	3 per area	M	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
8. Hot Leg Pressure (Wide Range)	4	1,2,3	3	M	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
	4	(d)	2	Q	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10

(a) When associated EDG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown."

(b) With three or more reactor coolant pumps (RCPs) in operation.

(c) ≥ 10^{-5}% power on the intermediate range detectors.

(d) When Pressurizer Safety Relief Valves (PSRVs) are required to be OPERABLE per LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)."

Table 3.3.1-1 (page 2 of 4)
Protection System Sensors, Manual Actuation Switches,
Signal Processors, and Actuation Devices

COMPONENT	REQUIRED NUMBER OF SENSORS, SWITCHES, SIGNAL PROCESSORS, OR ACTUATION DEVICES	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM~~UM~~ REQUIRED FOR FUNCTIONAL CAPABILITY	CONDITION	SURVEILLANCE REQUIREMENTS
9. Hot Leg Temperature (Narrow Range)	4 per division, 4 divisions	1,2[(c)]	3 per division, 3 divisions	J	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
10. Hot Leg Temperature (Wide Range)	4	3[(e)]	3	M	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
11. Intermediate Range	4	1[(f)],2,3[(g)]	3	K	SR 3.3.1.5 SR 3.3.1.6
12. Power Range	2 per division, 4 divisions	1,2,3[(g)]	2 per division, 3 divisions	K	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6
13. Pressurizer Level (Narrow Range)	4	1,2,3	3	M	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
14. Pressurizer Pressure (Narrow Range)	4	1,2,3[(h)]	3	L	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
15. Radiation Monitor - Containment High Range	4	1,2,3,4	3	~~K~~N	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
16. Radiation Monitor - Control Room HVAC Intake Activity	4	1,2,3,4	3	N	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
	4	5,6,(i)	3	R	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
17. RCP Current	3 per RCP	1,2,3	2 per RCP	M	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10

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(c) $\geq 10^{-5}$ % power on the intermediate range detectors.

(e) When Table 3.3.1-2, Trip/Actuation Function B.3.a is disabled.

(f) ≤ 10% RTP.

(g) With the Reactor Control, Surveillance and Limitation (RCSL) System capable of withdrawing a Rod Cluster Control Assembly (RCCA) or one or more RCCAs not fully inserted.

(h) With pressurizer pressure ≥ 2005 psia.

(i) During movement of irradiated fuel assemblies.

Table 3.3.1-1 (page 3 of 4)
Protection System Sensors, Manual Actuation Switches,
Signal Processors, and Actuation Devices

COMPONENT	REQUIRED NUMBER OF SENSORS, SWITCHES, SIGNAL PROCESSORS, OR ACTUATION DEVICES	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM~~UM~~ REQUIRED FOR FUNCTIONAL CAPABILITY	CONDITION	SURVEILLANCE REQUIREMENTS
18. RCP Delta P Sensors	2 per RCP	1,2,3	1 per RCP	M	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
19. RCP Speed	4	≥ 10% RTP	3	H	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
20. Reactor Coolant System (RCS) Loop Flow	4 per loop	1,2[c]	3 per loop	J	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
21. Reactor Trip Circuit Breaker Position Indication	4	1,2[g],3[g]	3	M	SR 3.3.1.5 SR 3.3.1.8 SR 3.3.1.10
22. Self-Powered Neutron Detectors	72	≥ 10% RTP	~~51~~67	H	SR 3.3.1.2 SR 3.3.1.5
23. Steam Generator (SG) Level (Narrow Range)	4 per SG	1,2[j],3[j]	3 per SG	M	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
24. SG Level (Wide Range)	4 per SG	1,2,3	3 per SG	M	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
25. SG Pressure	4 per SG	1,2,3	3 per SG	M	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.10
B. Manual Actuation Switches					
1. Reactor Trip	4	1,2,3[g]	3	K	SR 3.3.1.8
	4	4[g],5[g]	3	S	SR 3.3.1.8
2. Safety Injection System (SIS) Actuation	4	1,2,3,4	3	N	SR 3.3.1.8
3. SG Isolation	4 per SG	1,2,3	3 per SG	M	SR 3.3.1.8

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(c) $\geq 10^{-5}$ % power on the intermediate range detectors.

(g) With the RCSL capable of withdrawing a RCCA or one or more RCCAs not fully inserted.

(i) During movement of irradiated fuel assemblies.

(j) Except when all main feedwater (MFW) isolation valves are closed.

Table 3.3.1-1 (page 4 of 4)
Protection System Sensors, Manual Actuation Switches,
Signal Processors, and Actuation Devices

COMPONENT	REQUIRED NUMBER OF SENSORS, SWITCHES, SIGNAL PROCESSORS, OR ACTUATION DEVICES	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	MINIMUM~~UM~~ REQUIRED FOR FUNCTIONAL CAPABILITY	CONDITION	SURVEILLANCE REQUIREMENTS
C. Signal Processors					
1. Remote Acquisition Units (RAUs)	2 per division, 4 divisions	≥ 10% RTP	~~1~~2 per division, 4 divisions	H	SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.10
2. Acquisition and Processing Units (APUs)	5 per division, 4 divisions	Refer to Table 3.3.1-2	Refer to Table 3.3.1-2 16-193	Refer to Table 3.3.1-2	SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.10
3. Actuation Logic Units (ALUs)	4 per division, 4 divisions	1,2,3,4	3 per division, 4 divisions	N,O,P,Q,R,T	SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.10
	4 per division, 4 divisions	5,6,(i)	3 per division, 4 divisions	O,P,Q,R,T	SR 3.3.1.5 SR 3.3.1.7 SR 3.3.1.10
D. Actuation Devices					
1. Reactor Coolant Pump Bus and Trip Breakers	2 per pump	1,2,3,4	1 per pump	N	SR 3.3.1.8 SR 3.3.1.10
2. Reactor Trip Circuit Breakers	4	1,2,3[g]	3	K	SR 3.3.1.3 SR 3.3.1.10
3. Reactor Trip Contactors	4 per set, 23 sets	1,2,3[g]	3 per set, 23 sets	K	SR 3.3.1.3 SR 3.3.1.10

(g) With the RCSL capable of withdrawing a RCCA or one or more RCCAs not fully inserted.

(i) During movement of irradiated fuel assemblies.

(j) Except when all main feedwater (MFW) isolation valves are closed.

BASES

BACKGROUND (continued)

Licensees are to insert the name of the document(s) controlled under 10 CFR 50.59 that contain the LTSP and the methodology for calculating the as-left and as-found tolerances, for the phrase "a document controlled under 10 CFR 50.59" in the specifications.

The LTSP is a predetermined setting for a protective device chosen to ensure automatic actuation prior to the process variable reaching the Analytical Limit and thus ensuring that the SL would not be exceeded. As such, the LTSP accounts for uncertainties in setting the device (e.g., CALIBRATION), uncertainties in how the device might actually perform (e.g., repeatability), changes in the point of action of the device over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments (Ref. 6)). In this manner, the LTSP ensures that SLs are not exceeded. As such, the LTSP meets the definition of a SL-LSSS (Ref. 1). The LTSPs are determined as part of the safety analysis (Ref. 5).

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in Technical Specifications as "...being capable of performing its safety function(s)." However, use of the LTSP to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it were applied as an OPERABILITY limit for the "as-found" value of a protective device setting during a Surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protective device with a setting that has been found to be different from the LTSP due to some drift of the setting may still be OPERABLE since drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the LTSP and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as-found" setting of the protective device. Therefore, the device would still be OPERABLE since it would have performed its safety function and the only corrective action required would be to reset the device to the trip setpoint to account for further drift during the next surveillance interval.

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However, there is also some point beyond which the device would have not been able to perform its function due, for example, to greater than expected drift. The Allowable Value~~This LTSP specified in Table 3.3.1-2~~ is the least conservative value of the as-found setpoint that a division~~channel~~ can have when~~during~~ tested~~ing~~ such that a division~~channel~~ is OPERABLE if the as-found~~trip~~ setpoint is ~~found~~

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BACKGROUND (continued)

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conservative with respect to the Allowable Value during a ~~SENSOR~~ CALIBRATION~~OPERATIONAL TEST (SOT)~~. As such, the Allowable Value differs from the ~~LTSP~~ Nominal Trip Setpoint by an amount greater than or equal to the expected instrument channel uncertainties, such as drift, during the surveillance interval. In this manner, the actual setting of the device will ensure that an SL is not exceeded at any given point of time as long as the device has not drifted beyond that expected during the surveillance interval. Note that, although the channel is OPERABLE under these circumstances, the setpoint must be left adjusted to a value within the as-left tolerance, and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found). If the actual setting of the device is found to be non-conservative with respect to the Allowable Value, the device would be considered inoperable from a Technical Specification perspective. This requires corrective action including those actions required by 10 CFR 50.36 when automatic protective devices do not function as required.

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The departure from nucleate boiling ratio (DNBR) shall be maintained above the SL value to prevent departure from nucleate boiling (DNB),
- Fuel centerline melting shall not occur; and
- The RCS pressure SL of 2803 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 100 (Ref. 2) criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fractiojn of 10 CFR 100 limits. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event. However, these values and their associated LTSPs are not considered to be LSSS as defined in 10 CFR 50.36.

The PS is segmented into four interconnected modules and associated LCOs for the reactor trips and ESF functions. These modules are:

- Sensors, which include the associated ~~instrumentation~~signal conditioning;
- Manual actuation switches;

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SURVEILLANCE REQUIREMENTS (continued)

needed. A SOT shall be the injection of a simulated or actual signal into the division as close to the sensor as practicable to verify OPERABILITY of all devices in the division required for division OPERABILITY. The SOT shall include the verification of the accuracy and time constants of the analog input modules.

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~~The maximum permissible response time for analog input modules is prescribed by the process engineering of the specific application. Thus for each applicable PS function, the limiting response times will be shown to be consistent with the safety requirements.~~

~~The response time testing is performed in overlapping steps:~~

- ~~Verification of time constants of the input divisions during input module tests, and~~

- ~~Verification of the signal propagation time within the digital system.~~

~~The response time of the analog input divisions are tested periodically by injection of test signals in the input circuits. For this purpose, an external test computer is temporarily connected to the I&C system via permanently installed test plugs. While the input from the process is deactivated (by switching off the associated division(s) power supply), a binary input is provided to the data acquisition computers. The signal distribution to other computers is designed in the application software in the same way as for the normal measuring signals. Separate outputs are provided in the voting computers for each path. During the response time tests, the test machine connected to the I&C system generates a start signal and measures the reaction time of each signal path separately to verify that it does not exceed the worst case conditions specified for the specific system configuration. The measurements are performed a number of times to determine the statistical characteristics of each signal path.~~

The SOT may be performed by means of any series of sequential, overlapping, or total steps.

SR 3.3.1.6

A CALIBRATION of each PS sensor (except neutron detectors) every 24 months ensures that each instrument division is reading accurately and within tolerance. A CALIBRATION shall be the adjustment, as necessary, of the sensor output such that it responds within the necessary range and accuracy to known values of the parameter that the sensor monitors. The CALIBRATION shall encompass all devices in the division required for sensor OPERABILITY. CALIBRATION of instrument divisions with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place qualitative assessment of sensor behavior and

BASES

SURVEILLANCE REQUIREMENTS (continued)

normal CALIBRATION of the remaining adjustable devices in the division. The CALIBRATION may be performed by means of any series of sequential, overlapping, or total steps.

SR 3.3.1.7

The features of continuous self-monitoring of the PS system are described in Reference 8. Additional tests, which require the processor to be inoperable are not normally performed during operation. These EXTENDED SELF TESTS are performed at start-up of a computer each cycle. The startup sequence is as follows:

- Hardware basic test using the internal diagnosis monitor,
- Start-up self test of the operating system, and
- Switch over to normal operation after approximately two minutes.

Additional information is provided in Section 3 of Reference 8.

SR 3.3.1.8

SR 3.3.1.8 is the performance of an ADOT every ~~31 days~~24 months. ~~This test shall verify OPERABILITY by actuation of the RCP Bus and Trip Breakers.~~ The ADOT may be performed by means of any series of sequential, overlapping, or total steps.

SR 3.3.1.9

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SR 3.3.1.9 verifies that the Limiting Trip Setpoint and Permissive values have been properly loaded into the applicable APU.

SR 3.3.1.10

SR 3.3.1.10 verifies that the individual division actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in a document controlled under 10 CFR 50.59. Individual component response times are not modeled in the analyses.

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SURVEILLANCE REQUIREMENTS (continued)

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The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment reaches the required functional state (i.e., control and shutdown rods fully inserted in the reactor core, pumps at rated discharge pressure, or valves in full open or closed position). For divisions that include dynamic transfer Functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer Function set to one, with the resulting measured response time compared to the appropriate FSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value, provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total division measurements, or by the summation of allocated sensor, signal processing and actuation logic response times with actual response time tests on the remainder of the division. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) in place, onsite, or offsite (e.g., vendor) test measurements, or (3) utilizing vendor engineering specifications.

The allocations for sensor, signal conditioning, and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where response time could be affected is replacing the sensing assembly of a transmitter.

As appropriate, each division's response must be verified every 24 months on a STAGGERED TEST BASIS. Testing of the final actuation devices is included in the testing. Response times cannot be determined during unit operation because equipment operation is required to measure response times. Experience has shown that these components usually pass this surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.3.1.10 is modified by a Note stating that neutron detectors are excluded from PS RESPONSE TIME testing. The Note is necessary because of the difficulty in generating an appropriate detector input

BASES

SURVEILLANCE REQUIREMENTS (continued)

signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

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REFERENCES

1. ANP-10275P-A, ~~Revision 0,~~ "U.S. EPR Instrument Setpoint Methodology Topical Report," ~~March 2007~~February 2008.

2. 10 CFR 100.

3. 10 CFR 50, Appendix A, GDC 21.

4. ANP-10287P, "Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report," November 2007.

5. FSAR Chapter 15.

6. 10 CFR 50.49.

7. ANP-~~10271P~~10282P, ~~Revision 0, US EPR Nuclear Incore Instrumentation Systems Report, December 2006.~~ "POWERTRAX/E Online Core Monitoring Software for the U.S. EPR Technical Report," November 2007.

8. EMF-2341(P), Revision 1, "Generic Strategy for Periodic Surveillance Testing of TELEPERM XS Systems in U.S. Nuclear Generating Stations," March 2000.

Table B 3.3.1-1 (page 1 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
A. REACTOR TRIP~~S~~						
1. Low Departure from Nucleate Boiling Ratio (DNBR) a. Low DNBR d. High Quality	≥ 10% RTP 16-195	3	A total of ~~51~~ 67 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / ~~One~~ Two Remote Acquisition Units per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of ~~51~~ 67 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / ~~One~~Two Remote Acquisition Units per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of ~~51~~ 67 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / ~~One~~Two Remote Acquisition Units per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of ~~51~~ 67 Self-Powered Neutron Detectors (SPND) in any of the four divisions Pressurizer Pressure (NR) Cold Leg Temperature (NR) Reactor Coolant Pump Speed (1 of 2) Reactor Coolant System Loop Flow (3 of 4) / ~~One~~Two Remote Acquisition Units per division with a required OPERABLE SPND Acquisition and Processing Unit

Table B 3.3.1-1 (page 2 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
1. Low Departure from Nucleate Boiling Ratio (DNBR) b. Low DNBR and ~~(~~Imbalance or Rod Drop~~)~~ c. Variable Low DNBR and Rod Drop e. High Quality and Imbalance or Rod Drop	≥ 10% RTP	3	A total of 65 RCCA Position Indicators in any of the four divisions	A total of 65 RCCA Position Indicators in any of the four divisions	A total of 65 RCCA Position Indicators in any of the four divisions	A total of 65 RCCA Position Indicators in any of the four divisions
			A total of ~~51~~ 67 Self-Powered Neutron Detectors (SPND) in any of the four divisions	A total of ~~51~~ 67 Self-Powered Neutron Detectors (SPND) in any of the four divisions	A total of ~~51~~ 67 Self-Powered Neutron Detectors (SPND) in any of the four divisions	A total of ~~51~~ 67 Self-Powered Neutron Detectors (SPND) in any of the four divisions
			Pressurizer Pressure (NR)	Pressurizer Pressure (NR)	Pressurizer Pressure (NR)	Pressurizer Pressure (NR)
			Cold Leg Temperature (NR)	Cold Leg Temperature (NR)	Cold Leg Temperature (NR)	Cold Leg Temperature (NR)
			Reactor Coolant Pump Speed (1 of 2)	Reactor Coolant Pump Speed (1 of 2)	Reactor Coolant Pump Speed (1 of 2)	Reactor Coolant Pump Speed (1 of 2)
			Reactor Coolant System Loop Flow (3 of 4)	Reactor Coolant System Loop Flow (3 of 4)	Reactor Coolant System Loop Flow (3 of 4)	Reactor Coolant System Loop Flow (3 of 4)
			/	/	/	/
			One RCCA Unit per division with a required OPERABLE RCCA position indicator	One RCCA Unit per division with a required OPERABLE RCCA position indicator	One RCCA Unit per division with a required OPERABLE RCCA position indicator	One RCCA Unit per division with a required OPERABLE RCCA position indicator
			~~One~~ Two Remote Acquisition Units per division with a required OPERABLE SPND	~~One~~ Two Remote Acquisition Units per division with a required OPERABLE SPND	~~One~~ Two Remote Acquisition Units per division with a required OPERABLE SPND	~~One~~ Two Remote Acquisition Units per division with a required OPERABLE SPND
			Acquisition and Processing Unit	Acquisition and Processing Unit	Acquisition and Processing Unit	Acquisition and Processing Unit

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Table B 3.3.1-1 (page 3 of 7)
Protection System (PS) Functional Dependencies

TRIP/ACTUATION FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	COMPLETE DIVISIONS FOR FUNCTIONAL CAPABILITY SENSORS / PROCESSORS	DIVISION 1	DIVISION 2	DIVISION 3	DIVISION 4
2. High Linear Power Density	≥ 10% RTP 16-195	3	A total of ~~51~~ 67 Self-Powered Neutron Detectors (SPND) in any of the four divisions / ~~One~~ Two Remote Acquisition Units per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of ~~51~~ 67 Self-Powered Neutron Detectors (SPND) in any of the four divisions / ~~One~~ Two Remote Acquisition Units per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of ~~51~~ 67 Self-Powered Neutron Detectors (SPND) in any of the four divisions / ~~One~~ Two Remote Acquisition Units per division with a required OPERABLE SPND Acquisition and Processing Unit	A total of ~~51~~ 67 Self-Powered Neutron Detectors (SPND) in any of the four divisions / ~~One~~ Two Remote Acquisition Units per division with a required OPERABLE SPND Acquisition and Processing Unit
4. High Core Power Level	1,2[(a)]	3	Cold Leg Temperature (WR) Hot Leg Temperature (NR) (3 of 4) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) (3 of 4) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) (3 of 4) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit	Cold Leg Temperature (WR) Hot Leg Temperature (NR) (3 of 4) Hot Leg Pressure (WR) Reactor Coolant System Loop Flow (3 of 4) / Acquisition and Processing Unit

(a) $\geq 10^{-5}$ % power on the intermediate range detectors.