REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**301-8280**
SRP Section:	**07.01 – Instrumentation and Controls – Introduction**
Application Section:	**7.1**
Date of RAI Issued:	**11/10/2015**

Question No. 07.01-52

The staff reviewed the response to RAI 34-7870, Question 7.1-9 and found that additional information was needed as described below.

10 CFR 50.36(c)(1)(ii)(A) states, in part, "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded." Technical Report (TeR) APR1400-Z-J-NR-14005, Rev.0, "Setpoint Methodology for Plant Protection System," describes the setpoint methodology applied to the Plant Protection System (PPS) and Diverse Protection System (DPS) for the APR1400 and states conformance to BTP 7-12, Regulatory Guide 1.105 - Rev.3, and Regulatory Issue Summary (RIS) 2006-17.

For Question 7.1-9, staff requested a description on when reset setpoints would be used for reactor trip functions and the basis for manually changing the setpoint value. Also, describe how the new "fixed value" setpoint is determined and how this new setpoint is consistent with the more restrictive setpoint. The applicant responded by stating that "the low pressurizer pressure trip is provided to trip the reactor when the measured pressurizer pressure falls to a low preset value. At pressures below the normal operating range, this setpoint can be manually decreased to a fixed increment below the existing pressurizer pressure down to a minimum value. The incremental and minimum values are given in Table 7.2-4. This provides the capability to trip the reactor when required during plant cooldown." The section also states, "The low SG pressure trip is provided to trip the reactor when the measured SG pressure falls below a preset value. At SG pressure below normal, the setpoint can be manually decreased to a fixed increment below the existing system pressure. This is used during plant cooldown. The fixed increment is provided in Table 7.2-4." Regarding the "fixed value" APR1400 FSAR Tier 2, Chapter 7, Table 7.2-4 (2 of 2), Note (4), states, "Setpoint can be manually decreased to a fixed increment below existing pressure as pressure is reduced during controlled plant cooldown and is automatically increased as pressure is increased maintaining a fixed increment. This fixed increment is 28 kg/cm^2 (400 psi) for pressurizer pressure and 14 kg/cm^2 (200 psi) for steam generator pressure." The staff finds the response acceptable since applicant plans to use manual reduction of the setpoints for low pressurizer pressure and low SG pressure trips to shut down the plant without any unnecessary protective actions for plant cooldown. However, it is not clear to the staff if one Setpoint Reset switch on the Safety Console applies to both low pressurizer pressure and low SG pressure trips or if there are two switches, one for each manual reduction. Clarify the capability of the Setpoint Reset switch.

Response – (Rev.2)

There are two Setpoint Reset switches on the Safety Console. One switch applies to the low pressurizer pressure (LPP) and the other switch applies to the low steam generator pressure (LSGP) trip. As shown in Attachment 1, the two Setpoint Reset switches have been incorporated into Sections 4.7.1.3 and 4.7.3 of the Safety I&C System Technical Report (APR1400-Z-J-NR-14001-P, Rev.1) issued in May 2017.

The manual reduction of the LPP and LSGP setpoints is used to shut down the plant without an unnecessary protective action for plant cooldown. The manual reset setpoint is determined by subtracting the offset from the measured process value. The offsets that are defined as a predetermined amount between the measured process value and the newly determined setpoint for the LPP and LSGP trips are 28.1 kg/cm^2 (400 psi) and 14.1 kg/cm^2 (200 psi), respectively. Since the manual reset setpoints are restrictively determined not to exceed the corresponding offset, they are multiple setpoints that meet the requirement of IEEE 603, clause 6.8.2.

The variable overpower (VO) and low reactor coolant flow (LRCF) trip setpoints are also multiple setponts because they are restrictively determined by the constraints including ceiling, floor, rate, and step. However, the low steam generator level (LSGL) trip parameter does not have multiple setpoints because the LSGL trip setpoint is not a variable setpoint but a fixed one, as shown in DCD Table 7.2-4.

Therefore, multiple setpoints listed in DCD Table 7.2-4 are the LPP, LSGP, VO, and LRCF trip setpoints. Sections 7.2.3.4 and 7.4.3.3.5 of DCD Chapter 7 will be revised to incorporate multiple setpoints, as shown in Attachment 2. In addition, Section A.6.8 of the Safety I&C System Technical Report (APR1400-Z-J-NR-14001-P, Rev.1) will be maintained with the exception of incorporating the changed title of the setpoint methodology technical report, as shown in Attachment 3.

Supplemental Response to NRC comments on BTP 7-21

NRC Comment:

APR1400-Z-J-NR-14013-P, Rev.0, "Response Time Analysis of Safety I&C System," Section A.1.2, "Conformance," first sentence, makes the following statement.

"The analytical limit of the Setpoint Methodology for Plant Protection System (Reference 3) and the response times in Tables 7.2-5 and 7.3-7 of DCD Sections 7.2 and 7.3 are credited in the safety analysis of DCD Chapter 15."

There needs to be a change of the referenced “Setpoint Methodology for Plant Protection System” to “Setpoint Methodology for Safety-Related Instrumentation” to reflect the latest draft (RAI 07.01-41). Also, revise [reference 3] in Section 8 as well and reflect the latest revision. This same change also needs to be made in DCD Chapter 15 (Reference 77). Please verify any other parts of the DCA where this change is needed.

KHNP Response:

Based on the final response to RAI 301-8280, Question 07.01-41, APR1400-Z-J-NR-14005-P, “Setpoint Methodology for Safety-Related Instrumentation,” Rev. 2 will be issued. APR1400-Z-J-NR-14013-P, Rev.1 “Response Time Analysis of Safety I&C System” will be revised to incorporate the modified title of APR1400-Z-J-NR-14005-P, Rev. 2 into Item [3] of Section 8, as shown in Attachment 4.

Refer to the response to Question 2 below regarding Reference 77 in DCD Chapter 15.

References 72 and 74 in DCD Chapter 7 will be revised as shown in Attachment 5.

References 8 and 13 in APR1400-Z-J-NR-14001-P, Rev.1 will also be revised as shown in Attachment 6. In addition, the issue date will be reflected after the revision of each technical report is completed.

NRC Comment:

Another item is the safety analysis establishes ALs and response times needed to ensure safety and they are reflected in Chapter 15. Chapter 7 evaluates that the setpoint methodology will create setpoints starting at the ALs from Chapter 15. Chapter 7 also verifies that the response time for the I&C portion for any specific safety function falls within the response times assumed and established in Chapter 15 (typically milliseconds versus seconds).Therefore, any tables in Chapter 7 would reflect I&C time response, with possible overall statement to confirm that the time responses from Chapter 15 would be met. This point needs to be clear in the Response Time Analysis TeR. In addition, discussion provided in DCD Section 15.0.0.9, “Methodology for Determining Uncertainties” (portions quoted below) is misleading and incorrectly implies that the ANSI/ISA-67.04-1994 setpoint methodology is used is establishing the analytical limits used in Chapter 15 safety analysis. The AL values established in Chapter 15 safety analysis are used as inputs to the setpoint calculations performed in accordance the ISA methodology. The instrument setpoint calculations do not establish the AL.KHNP to confirm this conflict and correct the statements in Chapter 15 accordingly.

15.0.0.9 Methodology for Determining Uncertainties

Existing uncertainties in an instrument signal are classified as random or bias errors. Random errors are basic measurement uncertainties or variations that exist in any repeated measurement. These errors are usually caused by the combination of numerous effects that exist in any measurement. An exact value of a random error cannot be predicted for a specific measurement. To account for the random errors, the unsystematic errors are enveloped by upper and lower limits, around the measured value, that bound the most probable value for the instrumentation output at any instance.

Bias errors do not exhibit random normal distribution characteristics; rather, they exhibit a correlated, predictable, fixed, or systematic behavior. A bias exists where there is a known offset of measurement from the ideal value. Both random and bias error effects of an instrument measurement loop are evaluated. Uncertainties inherent in the signal communication process are accommodated by the method of setpoint calculation recommended by ANSI/ISA-67.04-1994, “Setpoints for Nuclear Safety-Related Instrumentation.” To establish the total uncertainty in an instrument or measurement, the various random and bias error effects are combined. The errors that are considered random are combined using statistical formulae such as the square-root-sum-of-the-squares. Bias errors are algebraically combined. Finally, the resultant random and bias errors are algebraically combined to yield a total uncertainty. Some events analyzed in the safety analysis result in a more severe environment for protection system equipment than others. As a result, the expected total equipment uncertainties can be event-specific, and a trip parameter can have an accident setpoint for each design basis event. The setpoints presented in Table 15.0-2 are determined based on the methodology presented above. The main methodology for determining uncertainties and the detailed uncertainty values are provided in Reference 51, which is based on NRC RG 1.105, Rev. 3, “Setpoints for Safety- Related Instrumentation.” The setpoint methodology for plant protection system safety- Related Instrumentation is provided in Reference 77.

KHNP Response:

Since analytical limits assumed in performing safety analysis do not consider the safety I&C system channel’s uncertainties, the trip setpoints described in DCD Chapter 7 are determined considering all related uncertainties in accordance with ANSI/ISA-67.04-1994 as endorsed by NRC RG 1.105, Rev. 3. DCD Section 15.0.0.9, however, was prepared to provide the concept on methodology for determining uncertainty. Therefore, Section 15.0.0.9 will be deleted as shown in Attachment 7 because the analytical limits in Table 15.0-2 of DCD Chapter 15 are not relevant to the setpoint determination methodology, but they are determined only by safety analysis.

Impact on DCD

Sections 7.1.5, 7.2.3.4 and 7.4.3.3.5 of DCD Chapter 7 will be revised as indicated in the attachments 2 and 5.

Sections 15.0.0.9 and 15.0.5 of DCD Chapter 15 will be revised as indicated in the attachment 7.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

Sections 4.7.1.3 and 4.7.4 of the TeR APR1400-Z-J-NR-14001-P/NP, Rev.1 has been issued as indicated in the attachment 1. Sections A.6.8 and 9 of the TeR APR1400-Z-J-NR-14001-P/NP, Rev.1 will be revised as indicated in the attachments 3 and 6 respectively.

Section 8 of the TeR APR1400-Z-J-NR-14013-P/NP, Rev.1 will be revised as indicated in the attachment 4.

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4.7.5 Control Panel Multiplexers

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Automatic protective action is initiated by the RPS to provide reasonable assurance of adequate protection of the fuel, fuel cladding, and RCS boundary during specified AOOs.

Automatic protective action is initiated by the RPS to aid the ESF systems in limiting the consequences of the accidents.

The Chapter 15 safety analysis addresses PAs and AOOs including single CEA ejection, load rejection, and turbine trip. Control functions to mitigate the consequences of a plant load rejection and turbine trip are addressed in Subsection 7.7.1.1. The RPS has no reliance on plant instrument air or cooling water to vital equipment.

7.2.3.3 Test and Inspection

The RPS complies with the test requirements of IEEE Std. 338. Test intervals and their bases are included in the Technical Specifications (Chapter 16).

Periodic testing complies with NRC RG 1.22 and NRC RG 1.118.

Multiple Setpoints

7.2.3.4 ~~Restrictive Setpoints~~ ~~Not Used~~

~~Restrictive setpoints are not used for the RPS.~~

Multiple setpoints comply with the restrictive setpoint requirement of IEEE std. 603, as described in Reference 26.

7.2.3.5 Conformance to General Design Criteria

Conformance with the applicable GDC is described in Reference 26, and cross references to relevant information are provided in Table 7.1-1.

7.2.3.6 Conformance with IEEE Std. 603

Conformance with IEEE Std. 603 is described in Reference 26.

7.2.3.7 Conformance with IEEE Std. 7-4.3.2

Conformance with IEEE Std. 7-4.3.2 is described in Reference 26.

7.2.4 Combined License Information

COL 7.2(1) The COL applicant is to provide site-specific CPCS startup test requirement.

7.4.3.3.2 Loss of Cooling Water to Vital Equipment

Loss of cooling water to vital equipment does not affect the safe shutdown function because the safety-related component cooling water system has two separate divisions of cooling water systems. Therefore, the loss of a single division does not hinder the safe shutdown function.

7.4.3.3.3 Loss of safety-related HVAC system

The safety-related HVAC systems connected to I&C equipment rooms or remote multiplexer rooms of each division maintain the mild (non-harsh) environments to meet the cabinet environmental design requirements.

A long-term loss of safety-related HVAC system may result in a loss of safety-related I&C equipment. However, divisional redundancy ensures that if there is a loss of safety-related I&C equipment that takes one safety division out of service, the second safety division will remain in service to perform the required safety function.

7.4.3.3.4 Plant Load Rejection, Turbine Trip, and Loss of Offsite Power

In the event of a LOOP associated with plant load rejection or turbine trip, the power for safe shutdown is provided by the EDGs. The EDGs provide power for operation of pumps and valves; the batteries or EDGs via the battery chargers provide power for operation of instrumentation and control systems required to actuate and control essential components.

7.4.3.3.5 ~~Restrictive Setpoints~~ ~~Not Used~~ Multiple Setpoints

~~There are no restrictive setpoints for the APR1400.~~

Multiple setpoints comply with the restrictive setpoint requirement of IEEE std. 603, as described in Reference 4.

7.4.4 Combined License Information

No combined license (COL) information is required with regard to Section 7.4.

7.4.5 References

1. Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants," Rev. 2, U.S. Nuclear Regulatory Commission, April 2009.

2. IEEE Std. 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, 1991.

Analysis:

The bypasses are always set manually as there are no automatic bypass provisions for maintenance bypasses for the PPS.

The bypass of the PPS parameters results in 2-out-of-3 coincidence logic. The bypass of the BOP ESFAS changes 1-out-of-2 logic to 1-out-of-1 logic.

The BISI in the MCR are designed to meet the RG 1.47. The PPS channel can be placed in Manual Bypass mode to facilitate maintenance activities. Indication is provided in the main control room whenever a PPS channel has been administratively bypassed for maintenance or taken out of service.

The PPS is designed to permit an inoperable channel to be placed in a bypass condition for the purpose of troubleshooting or periodic test of a redundant channel. If the PPS channel has been bypassed for any purpose, a signal is provided to allow this condition to be continuously indicated in the MCR. During such operation, the PPS continues to satisfy the SFC.

The FMEA for the PPS assumes that one of the initial conditions is a PPS channel is placed in the Bypass Mode. This initial condition imposed on the FMEA determines the overall effect of an evaluated failure on the safety system's capability to perform the required safety functions in this non-conservative mode.

The PPS supports maintenance activities, such as periodic maintenance, instrument loop testing, troubleshooting, etc. Access to features beyond displaying data such as the maintenance bypass would be under strict administrative and physical controls. These activities would be performed in accordance with site-specific administrative (procedural) and physical-access controls to set and/or change addressable constants, setpoints, and testing while the channel is in bypass mode. Such procedures would require manipulation of the FE keyswitch.

RPS and ESFAS parameters can be bypassed for maintenance. When one channel is in bypass, the coincidence logic in the LCL reverts to 2-out-of-3. The administrative procedure prohibits more than one channel from being placed in bypass.

The protection functions of the RPS and ESFAS are maintained while the system is bypassed.

A.6.8 Setpoints

Clause 6.8 ~~6.8.1~~ "6.8.1

"~~6.8.1~~ The allowance for uncertainties between the process analytical limit documented in Clause 4, item d) and the device setpoint shall be determined using a documented methodology. Refer to ANSI/ISA S67.04-1994.

~~6.8.2 Where it is necessary to provide multiple setpoints for adequate protection for a particular mode of operation or set of operating conditions, the design shall provide positive means of ensuring that the more restrictive setpoint is used when required. The devices used to prevent improper use of less restrictive setpoints shall be part of the sense and command features."~~

6.8.2 Where it is necessary to provide multiple setpoints for adequate protection for a particular mode of operation or set of operating conditions, the design shall provide positive means of ensuring that the more restrictive setpoint is used when required. The devices used to prevent improper use of less restrictive setpoints shall be part of the sense and command features."

Analysis:

RAI 301-8280, 07.01-52_Rev.2

The environment considered when determining errors is the most detrimental realistic environment calculated or postulated to exist until the worst case time of the required RPS or ESFAS. This environment may be different for different events analyzed. For the setpoint calculation, the accident environment error calculation for process equipment uses the environmental conditions up to the longest required time of trip or actuation that results in the largest errors, thus providing additional conservatism to the resulting setpoints.

The reference leg heating component uncertainties for SG level also take into account pressure and temperature variation within the SG.

For all temperature and pressure setpoints, the trip will be initiated at a point that is not at saturation for the equipment. For level setpoints, no analysis setpoint is within 5% of the ends of the level span.

When the pressure (or temperature) reaches to saturation condition, the pressure (or temperature) is not increased by temperature (or pressure). The temperature (or pressure) could not be used for trip parameter in this range. Therefore, the trip shall be initiated before saturation condition.

Analysis setpoint for high level is determined to be less than 95% and analysis setpoint for low level to be more than 5%. This is to protect the equipment considering delay time.

Manual reduction of the setpoints for low pressurizer pressure and low SG pressure trips is used for the controlled reduction of pressurizer pressure and SG pressure. The setpoint reductions are initiated by the MCR SC two pushbutton switches for each division, one pushbutton switch for the pressurizer pressure and one pushbutton switch for both SG pressures within the division. This method of setpoint reduction provides positive assurance that the setpoint is never decreased below the existing pressure by more than a predetermined amount.

The variable overpower trip setpoint tracks the actual reactor power from a minimum value to a high value or vice versa, if the power changes slowly enough. The variable overpower trip setpoint is designed with a maximum rate of decrease or increase. If the actual power increases at too rapid rate, it will catch up with the more slowly increasing setpoint and cause a trip.

The low reactor coolant flow trip setpoint automatically tracks below the input variables by a fixed margin for all decreasing inputs with a rate less than the rate limit. The setpoint decreases at a fixed rate for all decreasing input variable changes greater than the rate limit. If the input variable decreases at too rapid rate, it will catch up with the more slowly decreasing setpoint and cause a trip. The setpoint automatically increases as the input variable increases.

Refer to the Setpoint Methodology for ~~Plant Protection System~~ Safety-Related Instrumentation TeR for detailed setpoint methodology.

~~6.8.2 Where it is necessary to provide multiple setpoints for adequate protection for a particular mode of operation or set of operating conditions, the design shall provide positive means of ensuring that the more restrictive setpoint is used when required. The devices used to prevent improper use of less restrictive setpoints shall be part of the sense and command features.~~

A.7 Execute Features - Functional and Design Requirements

A.7.1 Automatic Control

Clause 7.1:

"Capability shall be incorporated in the execute features to receive and act upon automatic control signals from the sense and command features consistent with 4.4 of the design basis."

Analysis:

The RTSG is op

The ESF compo

8 REFERENCES

[1] DCD for the APR1400.

[2] APR1400-Z-J-NR-14001-P, "Safety I&C System", Rev. 0, November 2014.

[3] APR1400-Z-J-NR-14005-P, "~~Setpoint Methodology for Plant Protection System~~ Setpoint Methodology for Safety-Related Instrumentation", ~~Rev. 0, November 2014~~ Rev. 2, Later.

[4] WCAP-16097-P-A, "Common Qualified Platform Topical Report", Rev. 3, February 2013.

[5] APR1400-A-J-NR-14004-P(WCAP-17922-P), "Common Q Platform Supplemental Information in Support of the APR1400 Design Certification", Rev.0, November 2014.

[6] BTP 7-21, "Guidance on Digital Computer Real-Time Performance", Rev. 5, March 2007.

[7] BTP 7-14, "Guidance on Software Reviews for Digital Computer-based Instrumentation and Control Systems", Rev. 5, March 2007.

72. APR1400-Z-J-NR-14005-P, "Setpoint Methodology for ~~Plant Protection System~~ Safety-Related Instrumentation," Rev ~~1~~ 2, KHNP, ~~February 2017~~ Later.

73. DI&C-ISG-04, Rev. 1, "Highly Integrated Control Rooms – Communications Issues (HICRc)," U.S. Nuclear Regulatory Commission, 2009.

74. APR1400-Z-J-NR-14013-P, "Response Time Analysis of Safety I&C System," Rev ~~1~~ 2, KHNP, ~~February 2017~~ Later.

75. NUREG-0737, Supplement No. 1, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," U.S. Nuclear Regulatory Commission, 1983.

76. IEEE Std. 1050-1996, "IEEE Guide for Instrumentation Control Equipment Grounding in Generating Stations," Institute of Electrical and Electronic Engineers, 1996.

77. WCAP-16097-P-A, "Common Qualified Platform Topical Report," Rev. 3, February 2013.

78. APR1400-A-J-NR-14003-P (WCAP-17926-P), "APR1400 Disposition of Common Q Topical Report NRC Generic Open Items and Plant Specific Action Items," Rev. 0, October 2014.

79. APR1400-A-J-NR-14004-P (WCAP-17922-P), "Common Q Platform Supplemental Information in Support of the APR1400 Design Certification," Rev. 0, August 2014.

80. IEEE Std. 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, 2002.

81. EPRI TR-102323, "Guidelines for Electromagnetic Interference Testing in Nuclear Power Plants," Electric Power Research Institute, 1997.

9 REFERENCES

"Setpoint Methodology for Safety-Related Instrumetation," Later.

1. APR1400-Z-J-NR-14002-P, “Diversity and Defense-in-Depth,” February 2017

2. APR1400-Z-A-NR-14019-P, “CCF Coping Analysis”, February 2017

3. APR1400-E-J-NR-14001-P, “Component Interface Module,” February 2017

4. DI&C-ISG-04,Rev.1, “Highly Integrated Control Rooms – Communications Issues,” 2009

5. APR1400-K-Q-TR-11005-N, “KHNP Quality Assurance Program Description (QAPD) for the APR1400 Design Certification”

6. APR1400 DC Quality Assurance Manual (QAM)

7. APR1400-Z-J-NR-14004-P, “ Uncertainty Methodology and Application for Instrumentation,” February 2017

8. APR1400-Z-J-NR-14005-P, ~~“Setpoint Methodology for Plant Protection System,” February 2017~~

9. APR1400-F-C-NR-14001-P, “CPC Setpoint Analysis Methodology for APR1400,” September 2014.

10. APR1400-Z-J-NR-14003-P, Rev. 0, “Software Program Manual”, February 2017

11. Design Control Document for the APR1400

12. WCAP-16097-P-A, “Common Qualified Platform Topical Report”, Rev. 3, February 2013

13. APR1400-Z-J-NR-14013-P, “Response Time Analysis of Safety I&C System,” ~~February 2017~~ Later

14. APR1400-Z-J-NR-14012-P, “Control System CCF Analysis,” February 2017

15. APR1400-F-C-NR-14003-P, “Functional Design Requirements for a Core Protection Calculator System for APR1400,” February 2017

16. APR1400-E-I-NR-14012-P, “Style Guide,” February 2017

17. APR1400-A-J-NR-14004-P (WCAP-17922-P, “Common Q Platform Supplemental Information in Support of the APR1400 Design Certification, “ Rev. 0, August 2014

18. APR1400-A-J-NR-14003-P (WCAP-17926-P), “APR1400 Disposition of Common Q Topical Report NRC Generic Open Item and Plant Specific Action Items,” Rev.0, October 2014

19. APR1400-E-I-NR-14011-P, “Basic Human-System Interface,” February 2017

20. APR1400-E-I-NR-14007-P, “Human-System Interface Design Implementation Plan,” February 2017

21. APR1400-E-I-NR-14004-P, “Task Analysis Implementation Plan,” February 2017

In addition, operator errors are considered in developing event initiators and in limiting single failures (see Subsection 15.0.0.4.3 for a more detailed description).

Operator actions required to mitigate accidents are described in the event evaluation subsections.

15.0.0.7 Loss of Offsite Alternating Current (AC) Power

All event analyses resulting in a turbine generator trip consider the loss of offsite power (LOOP) while applying the same acceptance criteria for the event with and without LOOP. In the analyses for which the LOOP is assumed to result from a turbine trip, the time delay between the turbine trip and LOOP is assumed to be zero. However, a 3-second time delay can be assumed between reactor trip breakers opening and the turbine trip because of the turbine trip delay circuits. This time delay is assumed in the CEA misoperation and CEA ejection events while the other events do not use the time delay conservatively.

15.0.0.8 Long-Term Cooling

15.0.0.9 Intentionally Blank

The operator can initiate a controlled system cooldown by using the auxiliary feedwater (AFW) system in conjunction with the atmospheric dump valves (ADVs). In the absence of a forced reactor coolant flow, RCS heat is removed by natural circulation along with the steam generators (SGs). After the reactor coolant temperature and pressure have been reduced to approximately 176.7 °C (350 °F) and 31.6 kg/cm^{2}A (450 psia), respectively, the shutdown cooling system (SCS) is put into operation to reduce the RCS temperature to the cold shutdown condition. Any event-specific assumptions for the transition to shutdown conditions using the SCS are described in the relevant event-specific safety analysis section.

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~~15.0.0.9 Methodology for Determining Uncertainties~~

~~Existing uncertainties in an instrument signal are classified as random or bias errors. Random errors are basic measurement uncertainties or variations that exist in any repeated measurement. These errors are usually caused by the combination of numerous effects that exist in any measurement. An exact value of a random error cannot be predicted for a specific measurement. To account for the random errors, the unsystematic errors are enveloped by upper and lower limits, around the measured value, that bound the most probable value for the instrumentation output at any instance.~~

~~Bias errors do not exhibit random normal distribution characteristics; rather, they exhibit a correlated, predictable, fixed, or systematic behavior. A bias exists where there is a known offset of measurement from the ideal value. Both random and bias error effects of an instrument measurement loop are evaluated. Uncertainties inherent in the signal communication process are accommodated by the method of setpoint calculation recommended by ANSI/ISA-67.04-1994, "Setpoints for Nuclear Safety-Related Instrumentation."~~

~~To establish the total uncertainty in an instrument or measurement, the various random and bias error effects are combined. The errors that are considered random are combined using statistical formulae such as the square-root-sum-of-the-squares. Bias errors are algebraically combined. Finally, the resultant random and bias errors are algebraically combined to yield a total uncertainty.~~

~~Some events analyzed in the safety analysis result in a more severe environment for protection system equipment than others. As a result, the expected total equipment uncertainties can be event-specific, and a trip parameter can have an accident setpoint for each design basis event.~~

~~The setpoints presented in Table 15.0-2 are determined based on the methodology presented above. The main methodology for determining uncertainties and the detailed uncertainty values are provided in Reference 51, which is based on NRC RG 1.105, Rev. 3, "Setpoints for Safety-Related Instrumentation." The setpoint methodology for plant protection system is provided in Reference 77.~~

15.0.0.10 Thermal Conductivity Degradation

The effects of thermal conductivity degradation (TCD) on non-LOCA and LOCA evaluations, except for a CEA ejection accident and LBLOCA, are negligible. The effects are provided in Reference 78.

The results of the evaluation of a CEA ejection accident and LBLOCA are provided in Subsections 15.4.8.3 and 15.6.5.3, respectively.

15.0.1 Radiological Consequence Analysis Using Alternative Source Terms

This subsection is not applicable to the APR1400 because it is prepared to review the application for the initial implementation of an alternative source terms (AST)

42. NUREG-0800, "Nuclear Regulatory Commission Standard Review Plan, Section 11.3, Branch Technical Position 11-5," Rev.3, U.S. Nuclear Regulatory Commission, March 2007.

43. NUREG-0800, "Nuclear Regulatory Commission Standard Review Plan," Section 9.1.5, "Overhead Heavy Load Handling Systems," Rev. 1, U.S. Nuclear Regulatory Commission, March 2007.

44. ANSI-N14.6, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials," American National Standards Institute, 1993.

45. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plant," U.S. Nuclear Regulatory Commission, July 1980.

46. ANSI/ANS-57.2, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants," American Nuclear Society, 1983.

47. ANSI/ANS-57.1, "Design Requirements for Light Water Reactor Fuel Handling Systems," American Nuclear Society, 1992.

48. NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, May 1979.

49. Advanced Light Water Reactor Utility Requirements Document, "Fueling and Refueling Systems," Rev. 09, Electric Power Research Institute, February 2008.

50. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, March 1972.

51. (Deleted)

~~51. APR1400-Z-J-NR-14004-P, "Uncertainty Methodology and Application for Instrumentation," Rev. 1, KHNP, February 2017.~~

52. Oak Ridge National Laboratory C00545/MNYCD00, SCALE 4.4 "Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstation and Personal Computer," 1998.

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