# REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**APR1400 Design Certification**

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

**Docket No. 52-046**

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

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## Question No. 07.01-45

Discuss how using one-sided tolerance limit factor is consistent with Regulatory Guide 1.105, "Setpoints for Safety-Related Instrumentation," Revision 3.

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 APR1400-F-C-NR-14001, Rev. 0, "CPC Setpoint Analysis Methodology for APR1400," describes "Core Protection Calculator (CPC) setpoint analysis methodology for APR1400. The methodology is applied by combining uncertainties involved in the determination of the Local Power Density (LPD) and Departure from the Nucleate Boiling Ratio (DNBR) Limiting Safety System Settings (LSSS). The overall uncertainty factors assigned to LPD and DNBR establish that the adjusted LPD and DNBR are conservative at a 95/95 (probability/confidence) level throughout the core cycle, with respect to actual core conditions."

Section 2.1.3, "LPD LSSS statistical methods," of the CPC Setpoint Analysis Methodology Technical Report, states for Equations 2.3 and 2.5 state a normal distribution confidence coefficient of 1.645 for 95% confidence. It is not clear to the staff why these coefficients are consistent with the 95/95 tolerance limit discussion in Regulatory Guide 1.105, Rev. 3, which the staff interprets the limit to correspond to an error distribution approximately equal to two sigma value, 1.96, and not 1.645 as stated in the Technical Report APR1400-F-C-NR-14001. Provide the basis for using this factor with respect to Regulatory Guide 1.105, Revision 3.

## Response - (Rev.1)

The overall uncertainty factor analyses for the core protection calculator (CPC) local power density (LPD) are performed by comparing the 3-D peaking factor (Fq) that is calculated by the reactor core simulator and the peaking factor that is calculated by the off-line CPC, as described in technical report (TeR) APR1400-F-C-NR-14001, Rev. 1, "CPC Setpoint Analysis Methodology for APR1400"

The Fq modeling error ($X_F^i$) between the CPC synthesized Fq and the actual Fq is defined as follows:

$$X_F^i = \frac{(\text{"SYN" Fq})^i}{(\text{"ACTUAL" Fq})^i} - 1$$

Where ("SYN" Fq)$^i$ and ("ACTUAL" Fq)$^i$ are the CPC Fq and the reactor core simulator Fq for the i-th case. The Fq error is analyzed for each case at each time-in-life. Approximately 1200 cases are analyzed at each time-in-life (BOC, MOC, and EOC).

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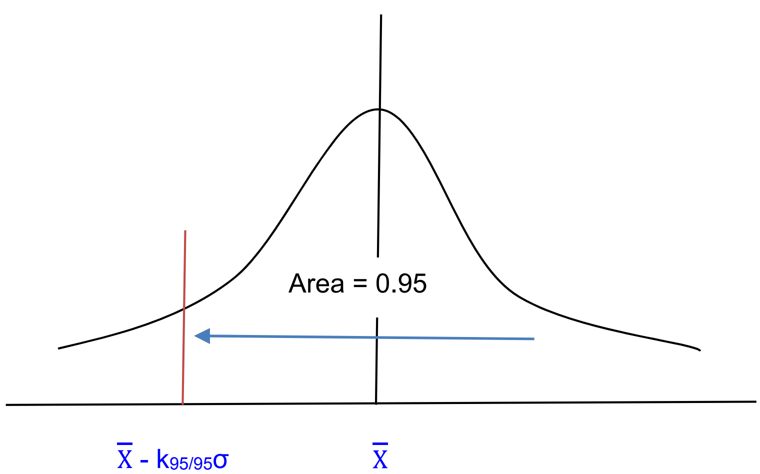
Figure 1. Probability Density Function (PDF) for Fq Modeling Errors




The value of 1.645

The CPC setpoint analysis methodology described in APR1400-F-C-NR-14001, Rev. 1, "CPC Setpoint Analysis Methodology for APR1400," is identical to that of WEC in Reference 1.

The 95/95 tolerance limit used in the CPC uncertainty factors has the form of “one-sided (or single-sided)” tolerance limit, which is described on Page 104 of Reference 2 and is repeated below:

*A one-sided tolerance limit U can be constructed so that with confidence coefficient (1-α) at least 100 p % of the normal population will be less than U*. We have

$$U = X + ks$$

where k (“the one-sided tolerance limit factor”) is defined as

$$k = \frac{k_{1-p} + (k_{1-p}^2 - ab)^{1/2}}{a}$$

$$a = 1 - \frac{k_\alpha^2}{2(N-1)}$$

$$b = k_{1-p}^2 - \frac{k_\alpha^2}{N}$$

$k_{1-p}$ = percentiles of a normal distribution for the probability p (1.645 for 95% probability)

$k_\alpha$ = percentiles of a normal distribution for the confidence coefficient (1-α) (1.645 for 95% confidence)

N = sample size.

If we use 97.5% confidence coefficient, then “1.96 (= $k_{0.025}$)” will be used instead of 1.645 (= $k_{0.05}$). This means that if we use $k_\alpha$ = 1.96, the confidence coefficient (1-α) becomes 97.5%, thus it is too conservative.

The value of 1.645 in the note of equation 2.5 is the 95/95 “one-sided tolerance limit factor” for infinite number of data points. In reality, the value of k is not 1.645 but rather depends on the sample size (N). In the description of the notes below equation 2.5, there is an editorial error. The description “(k= 1.645 for a 95/95 probability/confidence level and infinite” should be corrected as “… (k = 1.645 for a 95/95 probability/confidence level and infinite number of data points)”

Safety limit, Analytical limit and Trip setpoint for CPC

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The APR1400-F-C-NR-14001, Rev. 1, "CPC Setpoint Analysis Methodology for APR1400," will be revised, as indicated in the attachment associated with this response.

KHNP has followed the methodology for determining setpoints for safety-related instrumentation as described in RG 1.105 with the exception of the 95/95 tolerance limit. The DCD will be revised to address the exemption as shown in the attachment.

Reference

1. CEN-283(S)-P, "Statistical Combination of Uncertainties Part II," Combustion Engineering, Inc., October 1984.
2. E. L. Crow, et al, "Statistical Manual", Dover publication, Inc., New York, 1978.

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**Impact on DCD**

The Table 1.9-1, page 1.9-5, section 7.1.2.43, page 7.1-35, Table 7.1-1, section 7.2.2.7, section 15.0.0.9 and page 15.0-36 in the DCD rev.01 will be revised as shown in the attachment.

**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 Report**

Technical Report (APR1400-F-C-NR-14001) will be revised as shown in the attachment.

where

$$a = 1 - \frac{k_\alpha^{\ 2}}{2(N-1)} \tag{2-3b}$$

$$b = k_{1-p}^{\ 2} - \frac{k_\alpha^{\ 2}}{N} \tag{2-3c}$$

p

$k_{1-p}$ = percentiles of a normal distribution for the probability (1.645 for 95% probability)

$k_\alpha$ = percentiles of a normal distribution for the confidence coefficient

(1.645 for 95% confidence)

(1-α)

N = sample size.

If the error distribution is normal, the upper and lower one-sided 95/95 tolerance limits are calculated using the following equations:

$$\text{Lower 95/95 tolerance limit} = \overline{X} - k_{95/95}\sigma \tag{2-4a}$$

$$\text{Upper 95/95 tolerance limit} = \overline{X} + k_{95/95}\sigma \tag{2-4b}$$

where $\overline{X}$, σ, and $k_{95/95}\sigma$ are the sample mean, standard deviation, and one-sided tolerance limit factor, respectively.

If the error is not normally distributed, one-sided 95/95 tolerance limits are calculated using non-parametric techniques based on order statistics and the binomial probability distribution. First, the error distribution is placed in order from the smallest to the largest value. The binomial distribution is used to calculate a locator, L, from the ordered error distribution which estimates the one-sided tolerance limit at a 95/95 probability/confidence level. The locator L is calculated using the following equation. **TS**

The one-sided (upper or lower) 95/95 tolerance limit is obtained by selecting the error value (from the ordered error distribution) corresponding to the locator L. A non-parametric "kσ" is calculated from equation (2-4) using the determined one-sided tolerance limit and the known mean error.

#### 2.1.4. DNBR LSSS statistical methods

CPC setpoint is determined as the Analytical Limit. The overall uncertainty factors, which are cycle dependent, are applied directly in the CPC DNBR and LPD calculation.

# 1. INTRODUCTION

## 1.1 Purpose

The purpose of this report is to describe Core Protection Calculator (CPC) setpoint analysis methodology for APR1400. The methodology is applied statistically by combining uncertainties involved in the determination of the Local Power Density (LPD) and Departure from Nucleate Boiling Ratio (DNBR) Limiting Safety System Settings (LSSS). ~~CPC setpoint is determined by subtracting the overall uncertainty factors from the Safety Limit, but the overall uncertainty factors are cycle dependent.~~ Therefore, the overall uncertainty factors analyzed in the methods presented in this report are applied in the CPC DNBR and LPD calculation for every cycle. The overall uncertainty factors assigned to LPD and DNBR, establish that the adjusted LPD and DNBR are conservative at 95/95 probability/confidence throughout the core cycle with respect to actual core conditions.

This report describes the statistical combination of state parameter and modeling uncertainty for the determination of the LSSS overall uncertainty factors.

The methods described here are the same as those reviewed and approved earlier for C-E System 80 plants in references 1&2.

## 1.2 Background

The plant protection system in operation on APR1400 is composed of two sub-systems:

1. Engineered Safety Features Actuation System (ESFAS)

2. Reactor Protection System (RPS)

The CPC initiates two of the ten trips in the Reactor Protection System, the low DNBR trip and the high local power density trip. The RPS assesses the LPD and DNBR LSSS as a function of monitored reactor plant parameters. The CPC uses these monitored parameters as input data and calculates the on-line LPD and DNBR margin to trip limits. A list of variables that affect the CPC calculation of LPD and DNBR( in terms of the LPD and DNBR LSSS) is given in Table 1-1.

These two protective functions assure safe operation of a reactor in accordance with the criteria established in 10 CFR 50 Appendix A (Criteria Number 10, 20, and 25). The LSSS, combined with the Limiting Conditions for Operation (LCO), establishes the thresholds for automatic protection system actions to prevent the reactor core from exceeding the Specified Acceptable Fuel Design Limits (SAFDL) on center line fuel melting and Departure from Nucleate Boiling (DNB).

## 1.3 Report Scope

The scope of this report encompasses the following objectives:

- Describe CPC setpoint analysis methods applied statistically to combine uncertainties.

- Evaluate the aggregate uncertainties as they are applied in the calculation of LPD and DNBR.

The probability density functions associated with the uncertainties defined in Section 2.1 are analyzed to obtain the LPD and DNBR overall uncertainty factors based on a 95/95 (probability/confidence) level tolerance limit. The methods used for the determination of uncertainty on the power measurement, the core average Axial Shape Index (ASI), and the hot-pin ASI are also described.

## APPENDIX A CPC LPD SETPOINT CALCULATION

~~CPC LPD Setpoint (LSSS) = Safety Limit for LPD - Overall Uncertainty Factor (BERR3, BERR4)~~

~~CPC LPD Setpoint (LSSS) - LPD calculated by CPC = Margin~~

~~Safety Limit for LPD - (LPD calculated by CPC + Overall Uncertainty Factor (BERR3, BERR4)) = Margin~~

1. CPC LPD overall uncertainty factor (BERR3)

(1) Composite Fq modeling penalty factor ($PM_F$)

1) The mean of the composite Fq modeling uncertainty ($\overline{X}_{FM}$)

$$\overline{X}_{FM} = \sum^{N} \left( \frac{C_i - F_i}{\ } \right)$$

CPC LPD Setpoint (LSSS) = Analytical Limit for LPD
Margin = CPC LPD Setpoint (LSSS) - LPD calculated by CPC including Overall Uncertainty Factor

$F_i$ : reactor core simulator calculated Fq

$C_i$ : CPC calculated Fq

N : sample size

2) kσ of the composite Fq modeling uncertainty $(k\sigma)_{FT}$

A. CPC power distribution synthesis uncertainty ($k\sigma_{FM}$)

$$(k\sigma)_{FM} = -(TL)_{FM} + \overline{X}_{FM}$$

B. engineering factor ($k\sigma_{FE}$)

C. rod bow penalties ($k\sigma_{FF}, k\sigma_{PP}$)

D. computer processing uncertainty ($k\sigma_{CP}$)

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$$PM_F = \frac{1}{1 + TL_F}$$

where

$TL_F$ = 1) – 2)

(2) Axial fuel densification uncertainty (PA)

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2. Core power measurement uncertainty factor (BERR4)

## APPENDIX B CPC DNBR SETPOINT CALCULATION

~~CPC DNBR Setpoint (LSSS) = Safety Limit for DNBR - Overall Uncertainty Factor(BERR0,BERR1,BERR2)~~

~~CPC DNBR Setpoint (LSSS) - DNBR calculated by CPC = Margin~~

~~Safety Limit for DNBR - (DNBR calculated by CPC + Overall Uncertainty Factor (BERR0,BERR1,BERR2)) = Margin~~

1. CPC DNB-OPM uncertainty factor (BERR1)

   (1) The composite DNB-OPM modeling penalty factor ($PM_D$)

      1) The mean of the DNB-OPM modeling error ($\bar{X}_{DM}$)

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CPC DNBR Setpoint (LSSS) = Analytical Limit for DNBR
Margin = CPC DNBR Setpoint (LSSS) - DNBR calculated by CPC including Overall Uncertainty Factor

      2) kσ of the composite Fq modeling uncertainty $(k\sigma)_{DT}$

         A. DNB-OPM modeling algorithm uncertainties ($k\sigma_{DM}$)

$$(k\sigma)_{DM} = (TL)_{DM} - \bar{X}_{DM}$$

         B. rod and poison bow penalties ($k\sigma_{PF}$, $k\sigma_{PP}$)

         C. DNBR computer processing ($k\sigma_{CP}$)

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$$TL_D = 1) + 2)$$

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2. Thermal power measurement uncertainty factor for the CPC DNBR calculation (BERR0)

   (1) Calibration Allowance

Table 1.9-1 (12 of 35)

| NRC Regulatory Guide | Revision / Issue Date | Conformance or Summary Description of Deviation | DCD Tier 2 Section |
|---|---|---|---|
| 1.100 Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants | Rev. 3 09/2009 | The APR1400 conforms with this NRC RG. | 3.9.2.2.1, 3.9.3.3.1.1, 3.9.3.3.1.2, 3.9.3.3.1.3, 3.9.3.3.1.3.1, 3.9.3.3.2.2, 3.9.6.1, 3.10.1.1, 3.10.1, 3.10.2, 3.10.2.1, 3.10.2.2, 3.10.2.3, 3.11.2, 5.2.2.1.1, 5.4.12.2.1, 5.4.12.2.2, Table 6.5-2, 8.3.2.2.2 |
| 1.101 Emergency Planning and Preparedness for Nuclear Power Reactors | Rev. 5 06/2005 | Not applicable (COL) | N/A |
| 1.102 Flood Protection for Nuclear Power Plants | Rev. 1 09/1976 | The APR1400 conforms with this NRC RG. | 3.4.1.1, 3.4.1.2 |
| 1.105 Setpoints for Safety-Related Instrumentation | Rev. 3 12/1999 | The APR1400 conforms with this NRC RG. | 7.1.2.44, Table 7.1-1, 7.2.2.7, 7.3.2.7, 15.0.0.9 |
| 1.106 Thermal Overload Protection for Electric Motors on Motor-Operated Valves | Rev. 2 02/2012 | The APR1400 conforms with this NRC RG. | Table 8.1-2, 8.3.1.1.3.11, 8.3.1.2.2, 8.3.2.2.2 |

except for the setpoints calculation using CPC Setpoint Analysis Methodology Technical Report (Reference 8).

6. SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” U.S. Nuclear Regulatory Commission, July 1993

7. 10 CFR Part 21, “Reporting of Defects and Noncompliance,” U.S. Nuclear Regulatory Commission.

8. APR1400-F-C-NR-14001-P, "CPC Setpoint Analysis Methodology for APR1400," Rev.1, KHNP, February 2017.

The independence and separation of redundant Class 1E circuits within and between the PPS assemblies or ESF-CCS assemblies are accomplished primarily by using fiber-optic technology. The optical technology provides reasonable assurance that no single credible electrical fault in a PPS division will prevent the circuitry in any other redundant division from performing its safety function.

The ESF-CCS cabinets provide separation and independence for the 2-out-of-4 actuation and component control logic of the divisions in the redundant ESF systems. The component control logic for each division is contained in a separate cabinet. The redundant cabinets are physically separated from each other by locating them in separate zones.

The RTSS consists of two sets of four reactor trip switchgears (RTSGs). Each RTSG, along with the associated switches, contacts, and relays, is contained in a separate cabinet. Each cabinet is physically separated from the other cabinets. This method of construction provides reasonable assurance that a single credible failure in one RTSG will not cause malfunction or failure in another cabinet.

The separation and independence of the power supplies are described further in Subsection 8.3.1.

The digital data sent from the safety system to non-safety systems (e.g., IPS, QIAS-N) for status monitoring, alarm, and display are isolated from the safety system. Fiber-optic isolation and other techniques are used to provide reasonable assurance that no credible failures on the non-Class 1E side of the isolation device will affect the PPS side and that the independence of the PPS will not be jeopardized.

7.1.2.42 Conformance with NRC RG 1.97

The design of the accident monitoring instrumentation system (the QIAS-P, QIAS-N, and IPS) is described in Subsection 7.5.1.1. The design complies with NRC RG 1.97.

7.1.2.43 Conformance with NRC RG 1.105

The setpoint methodology (Reference 72) follows the methodology in ISA-S67.04 (Reference 34) as endorsed by NRC RG 1.105 (Reference 35).

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

except for the setpoints calculation using CPC Setpoint Analysis Methodology Technical Report (Reference 82).

72. APR1400-Z-J-NR-14005-P, “Setpoint Methodology for Plant Protection System,” Rev 1, KHNP, February 2017.

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, KHNP, February 2017.

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.

82. APR1400-F-C-NR-14001-P, "CPC Setpoint Analysis Methodology for APR1400," Rev.1, KHNP, February 2017.

Table 7.1-1 (3 of 6)

| Applicable Criteria | | Title | I&C System: RTS | I&C System: ESF System | I&C System: QIAS-P | I&C System: QIAS-N | I&C System: PCS | I&C System: P-CCS | I&C System: DAS | Section in APR1400 DCD |
|---|---|---|---|---|---|---|---|---|---|---|
| Staff Requirements Memoranda | | | | | | | | | | |
| 34 | SRM on SECY-93-087, Item II.Q | Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems | × | × | | | | | × | 7.2, 7.3, 7.8, 7.9 |
| 35 | SRM on SECY-93-087, Item II.T | Control Room Annunciator (Alarm) Reliability | | | | × | | | | 7.5, 7.9 |
| NRC Regulatory Guides | | | | | | | | | | |
| 36 | NRC RG 1.22 | Periodic Testing of Protection System Actuation Functions | × | × | | | | | | 7.2, 7.3,, 7.9 |
| 37 | NRC RG 1.47 | Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems | × | × | | × | | | | 7.2, 7.3, 7.5, 7.6, 7.9 |
| 38 | NRC RG 1.53 | Application of the Single-Failure Criterion to Safety Systems | × | × | × | | | | | 7.2, 7.3, 7.4, 7.5, 7.6, 7.9 |
| 39 | NRC RG 1.62 | Manual Initiation of Protective Actions | × | × | | | | | × | 7.2, 7.3, 7.8 |
| 40 | NRC RG 1.75 | Criteria for Independence of Electrical Safety Systems | × | × | × | × | × | × | × | 7.2, 7.3, 7.4, 7.5, 7.6, 7.7, 7.8, 7.9 |
| 41 | NRC RG 1.97 | Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants | | | × | × | | | | 7.5 |
| 42 | NRC RG 1.105 | Setpoints for Safety-Related Instrumentation (1) | × | × | × | × | | | | 7.2, 7.3, 7.4, 7.5, 7.6, 7.9 |
| 43 | NRC RG 1.118 | Periodic Testing of Electric Power and Protection Systems | × | × | × | × | | | | 7.2, 7.3, 7.4, 7.5, 7.6, 7.9 |
| 44 | NRC RG 1.151 | Instrument Sensing Lines | × | × | × | | | | | 7.2, 7.3, 7.5, |
| 45 | NRC RG 1.152 | Criteria for Digital Computers in Safety Systems of Nuclear Power Plants | × | × | × | | | | | 7.2, 7.3, 7.5, 7.9 |

(1) Except for the setpoints calculation using CPC Setpoint Analysis Methodology Technical Report (Reference 82).

### 7.2.2.7 Setpoint Determination

The RPS nominal trip setpoints are determined based on the analysis setpoints in the Chapter 15 safety analysis, in which analysis setpoints exist for the parameters.

CPC setpoint is determined as the analytical limit. The overall uncertainty factors, which are the combined uncertainty in a CPC channel, are applied directly in the CPC DNBR and LPD calculation.

When determining uncertainties, the wors actuation is assumed based on the bou calculating uncertainty is provided in the Uncertainty Methodology and Application for Instrumentation Technical Report (Reference 13).

The methodology for combining uncertainty in a channel and determining the final trip setpoint is provided in the Setpoint Methodology for Plant Protection System Technical Report and CPC Setpoint Analysis Methodology Technical Report (References 14 and 29).

The setpoint methodology includes the relationship between the analytical limit, setpoint, and channel uncertainty. The setpoint methodology provides the channel uncertainty calculations associated with the setpoints used for the RT and ESF actuation functions.

The setpoint methodology meets the guidance of ANSI/ISA-S67.04 (Reference 15), as endorsed by NRC RG 1.105 (Reference 16).

The instrumentation channel response time is the signal propagation time from the process sensor to the final actuation device. The response time for the RPS meets the response time assumed in Chapter 15. The reactor protective instrumentation response times assumed in the safety analysis in Chapter 15

except for the CPC setpoints which conforms with CPC Setpoint Analysis Methodology Technical Report (Reference 29).

The methodology for calculating system response time is provided in the Response Time Analysis of Safety I&C System Technical Report (Reference 17).

### 7.2.2.8 Equipment Qualification

The RPS meets the requirements of IEEE Std. 323 (References 18) for environmental qualification, IEEE Std. 344 (Reference 19) for seismic qualification, NRC RG 1.89 (Reference 20), and NRC RG 1.209 (Reference 21).

The RPS that is designed and tested to minimize both the emission and susceptibility of EMI and RFI meets the guidance of NRC RG 1.180 (Reference 22).

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.

except for the CPC setpoints which conforms with CPC Setpoint Analysis Methodology Technical Report (Reference 82).

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)

78. APR1400-F-A-NR-14002-P, “The Effect of Thermal Conductivity Degradation on APR1400 Design and Safety Analyses,” Rev. 0, KHNP, September 2014.

79. NUREG-1462, “Final Safety Evaluation Report Related to the Certification of the System 80+ Design Docket No. 52-002,” Rev. 0, U.S. Nuclear Regulatory Commission, 1994.

80. APR1400-Z-A-NR-14014-P, “ATWS Evaluation,” Rev. 0, KHNP, November 2014.

81. APR1400-F-M-TR-13001-P (Proprietary), “PLUS7 Fuel Design for the APR1400,” Rev. 0, KHNP, August 2013.

82. APR1400-F-C-NR-14001-P, "CPC Setpoint Analysis Methodology for APR1400," Rev.1, KHNP, February 2017.