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**
Application Section:	**07.01**
Date of RAI Issue:	**11/10/2015**

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.2)

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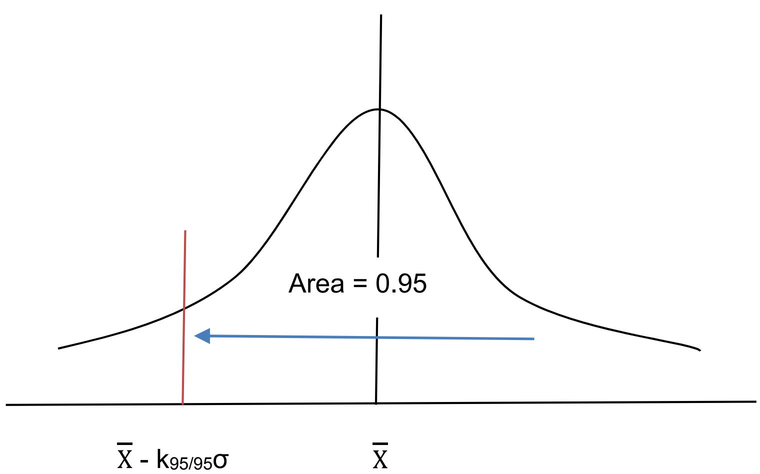
Figure 1. Probability Density Function (PDF) for Fq Modeling Errors



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_α = 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_α = 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 2 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 1.

The revised response (Rev.2) to RAI 301-8280 Q07.01-45 is submitted for the following reasons:

1) To respond APR1400 I&C Public meeting action item held on May 2 and 3, 2016
2) To apply the response to RAI 301-8280 Q07.01-45 Rev.1 in the DCD and Technical Report (APR1400-F-C-NR-14001) that have not yet been applied.

The action item is to add the Topical Report CEN-308-P-A, “CPC/CEAC Software Modifications for the CPC Improvement Program” in the Technical Report (APR1400-F-C-NR-14001) as a reference. Therefore, technical report (APR1400-F-C-NR-14001) will be revised to add the topical report (CEN-308-P-A) by reference as shown in the attachment 2.

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.

Impact on DCD

The Subsection 1.9.9, Table 1.9-1, Subsection 7.1.2.43, Subsection 7.1.5, Table 7.1-1, Subsection 7.2.2.7, Subsection 15.0.5 in the DCD Rev.1 and Table 15.0-12 in the DCD Rev.2 will be revised as shown in the attachment 1.

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 2.

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.

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).

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

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

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							Section in APR1400 DCD
			RTS	ESF System	QIAS-P	QIAS-N	PCS	P-CCS	DAS	
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[illegible] actuation is assumed based on the bou[illegible] 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 [illegible]

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).

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.

Table 15.0-12 (2 of 4)

Issue #	Subject	Disposition for APR1400
USI-C-6	LOCA Heat Source	The methodologies for evaluating large-break and small-break LOCA (References 63 and 67 in Subsection 15.0.5) account for effects of power density, decay heat, stored energy, fission power decay, and their associated uncertainties as required.
USI-C-10	Effective Operation of Containment Spray	An automatically actuated containment spray system is conservatively assumed to be activated at time zero in LOCA minimum containment pressure analysis, as presented in Subsection 6.2.1.5, Minimum Containment Pressure Analysis for Performance Capability Studies of the Emergency Core Cooling System.
GSI-3	Instrumentation Setpoint Drift	The APR1400 includes safety-related instrumentation and controls with established setpoints to actuate safety functions (Chapter 7). Setpoints for safety-related systems and components (e.g., the Plant Protection System), are established and maintained in accordance with the guidance given in NRC RG 1.105, Rev. 3, and conform to the criteria identified in ISA-S67.04-1994. (1)
GSI-22	Detection of boron dilution events during shutdown and refueling	This requirement is satisfied through a safety-related system that monitors boron concentration in the RCS and isolates the CVCS if boron dilution is detected as described in Subsection 15.4.6, Inadvertent Decrease in Boron Concentration in the Reactor Coolant System
GSI-23	Reactor Coolant Pump Seal Failure	The APR1400 minimizes the possibility of core damage resulting from a small-break LOCA event caused by an RCP shaft seal failure by assuring seal integrity. Reasonable assurance of seal integrity is provided by seal and support systems design, which address susceptibility to station blackout. RCP seal integrity can be maintained by either of two independent sources of cooling water: the seal injection flow from the Chemical and Volume Control System (CVCS) or Component Cooling Water (CCW). In the event of a loss of offsite AC power or during a complete loss of AC power, power can be supplied to the charging pumps, auxiliary charging pump and CCW pumps, or auxiliary charging pump, respectively, as presented in: • Subsection 5.4.1, Reactor Coolant Pumps • Subsection 9.2.2, Component Cooling Water System • Subsection 9.3.4, Chemical and Volume Control System

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

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.

where

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

$$b = k_{1-p}^{2} - \frac{k_\alpha^{2}}{N} \quad \text{(2-3c)}$$

p

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

k_α = percentiles of a normal distribution for the confidence coefficient

(1.645 for 95% confidence)

N = sample size.

(1-α)

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

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

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

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

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 ($\bar{X}_{FM}$)

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

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 ($k\sigma_{FM}$)

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

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

ABSTRACT

This report 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). ~~CPC setpoint is determined by subtracting the overall uncertainty factors from the Safety Limit, but the overall uncertainty factors are cycle dependent.~~ Therefore, overall uncertainty factors analyzed in the methods presented in this report are applied in CPC DNBR and LPD calculations for every cycle. 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.

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

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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. The algorithms and inputs of CPC are described in the Functional Design Requirements (Reference 3) which includes the CPCS Improvement program (Reference 4) 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.

elements are then determined from a regression analysis of the ex-core signals and the corresponding bottom, middle and top third integrals of the core peripheral power.

In the MSCU methodology, the Barrel of SAMs is applied as SAM uncertainty. The Barrel of SAMs is a family of SAMs calculated with noisy ex-core signal and peripheral powers.

2.2.1.5 Boundary Point Power Correlation Coefficient (BPPCC)

The CPC Boundary Point Power Correlation Coefficient (BPPCC) values used in the power synthesis algorithm are verified during power ascension testing. The predicted BPPCC values are calculated by simulating a free unrodded xenon oscillation similar to the SAM measurement procedure. The predicted BPPCC values are then determined from a regression analysis of the top and bottom one-third, of the core average power integrals and the boundary point powers at the top and bottom of the core.

In the MSCU methodology, the BPPCC uncertainty is randomly selected from a uniformly uncertainty distribution. This uncertainty is then multiplied to the BPPCC generated by the reactor core simulator then used as input to the CPC power distribution algorithm.

2.2.1.6 Other Uncertainty Factors

Axial Fuel Densification Uncertainty

The axial fuel densification uncertainty factor considers the global effect of the shrinkage (due to heating and irradiation) of the fuel pellet stack, on Fq, because the CPC Fq calculation does not account for it directly. This uncertainty factor, calculated based on the methodology described in ~~Reference 3~~ Reference 5, will be used as a multiplier on the net Fq uncertainty.

Fuel and Poison Rod Bow Uncertainties

The fuel and poison rod bow uncertainty considers the effect of "bowing" of the fuel and poison rods, due to heating and irradiation, on Fq, because the CPC Fq calculation does not account for it directly. These factors, calculated based on the methodology described in ~~Reference 4~~ Reference 6, will be part of the composite Fq modeling uncertainty.

Computer Processing Uncertainty

The computer processing uncertainty considers the effect of the computer machine precision of the off-line computer and the on-site computer on the CPC Fq calculations. The computer processing uncertainty will be part of the composite Fq modeling uncertainty.

Engineering Factor Uncertainty

The engineering factor uncertainty accounts for the effect of variations in the fuel pellet and clad manufacturing process. Variations in fuel pellet diameter and enrichment are included in this allowance, as are variations in clad diameter and thickness. These result in variations in the quantity of fissile material and introduce variations in the gap conductance. This factor, calculated based on methodology described in Reference 3, will be part of the composite Fq modeling uncertainty.

2.2.1.7 Overall LPD LSSS Uncertainty Factor

An overall CPC Fq uncertainty factor is determined by combining 95/95 probability/confidence tolerance limits of the error components. This overall uncertainty factor includes Fq modeling uncertainty, CECOR Fxy measurement uncertainty, startup test acceptance criteria uncertainty, axial fuel densification uncertainty fuel and poison rod bow uncertainties, computer processing uncertainty, engineering factor

4. REFERENCES

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2. CEN-356(V)-P-A, "Modified Statistical Combination of Uncertainties," Rev.1-P-A, Combustion Engineering, Inc., May 1988.

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4. CEN-308-P-A, "CPC/CEAC Software Modifications for the CPC Improvement Program," Combustion Engineering, Inc., April 1986.
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