FINAL SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS (REVISED) Topical Report APR1400-F-A-TR-12004-P "Realistic Evaluation Methodology for Large-Break Loss of Coolant Accident of the APR1400"

Korea Hydro & Nuclear Power Co. Ltd

TOPICAL REPORT APR1400-F-A-TR-12004-P
"REALISTIC EVALUATION METHODOLOGY FOR LARGE-BREAK LOSS OF COOLANT ACCIDENT OF THE APR1400"

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ACRONYMS AND ABBREVIATIONS

Acronym	Definition
ADAMS	Agency-wide Documents Access and Management System
AI	audit issue
APR	Advanced Power Reactor
ARO	all rod out
ASME	American Society of Mechanical Engineers
ATLAS	Advanced Thermal-hydraulic Test Loop for Accident Simulation
B&PV	Boiler and Pressure Vessel
BEMUSE	Best Estimate Methods - Uncertainty and Sensitivity Evaluation
BETHSY	Boucle d'Etudes Thermo-Hydraulique Système
BOC	beginning of cycle
CAMP	Code Applications and Maintenance Program
CAREM	code-accuracy-based realistic evaluation methodology
CCF	counter current flow
CCFL	counter current flow limitation
CCTF	Cylindrical Core Test Facility
CFR	Code of Federal Regulations
CHF	critical heat flux
CLI	cold leg injection
CLS	cold leg split
CSAU	Code Scaling Applicability and Uncertainty
CWO	core wide oxidation
DC	Design Certification
DCD	Design Control Document
DECLG	double ended cold leg guillotine
DOBO	downcomer boiling
DVI	direct vessel injection
ECC	emergency core cooling
ECCS	emergency core cooling system
ECCW	emergency core cooling water
EDG	emergency diesel generator
EM	evaluation model
EMDAP	evaluation model development and assessment process
FCT	fuel centerline temperature
FLECHT	Full Length Emergency Core Heat Transfer
FOM	figure of merit
HFP	hot full power
HTC	heat transfer coefficient
IET	integral effects test
INEEL	Idaho National Engineering and Environmental Laboratory
IRWST	in-containment refueling water storage tank
KHNP	Korea Hydro and Nuclear Power
KNGR	Korea Next Generation Reactor
L/D	length to diameter
LBLOCA	Large-break loss-of-coolant accident
LHGR	linear heat generation rate
LOBI	LWR Off-Normal Behaviour Investigation
LOCA	loss of coolant accident

Acronym	Definition
LOFT	Loss of Fluid Test
LOOP	loss of offsite power
LTCC	long term core cooling
MCO	maximum cladding oxidation
MIDAS	Multi-dimensional Investigation in Downcomer Annulus Simulation
MLO	maximum local oxidation
NA	not applicable
NEPTUN	Swiss LWR THTF facility for reflood experiments – see NUREG/IA-0040
NFI	Nuclear Fuel Industries
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
PCT	peak cladding temperature
PIRT	phenomena identification and ranking table
PKL	PWR Integral System Test Facility
PWR	pressurized water reactor
QA	quality assurance
QAPD	Quality Assurance Program Description
RAI	request for additional information
RCP	reactor coolant pump
RCS	reactor coolant system
Ref.	Reference
RES	NRC Office of Regulatory Research
RG	Regulatory Guide
RPV	reactor pressure vessel
SEASET	Separate Effects and Systems Effects Tests
SER	Safety Evaluation Report
SET	separate effect test
SG	steam generator
SGS	steam generator simulator
SI	safety injection
SIP	safety injection pump
SIT	safety injection tank
SIT-FD	SIT Fluidic Device
SKN	Shin-Kori Nuclear Power Plant
SRP	Standard Review Plan
SRS	simple random sampling
TCD	thermal conductivity degradation
THTF	Thermal Hydraulic Test Facility
TR	Topical Report APR1400-F-A-TR-12004
UGS	upper guide structure
UO_2	uranium dioxide
UPTF	Upper Plenum Test Facility
VAPER	Valve Performance Evaluation Rig

1. INTRODUCTION

By letter dated January 7, 2013, Korea Hydro and Nuclear Power Company, Ltd. (the applicant) requested to the United States Nuclear Regulatory Commission (NRC) review and approval of Topical Report APR1400-F-A-TR-12004-P, Revision 0 (TR [1]) [Ref. 1] (see NRC's Agency-wide Documents Access and Management System (ADAMS)). By letter dated August 11, 2017 (Reference 13), the applicant submitted a revision to the TR which included updates to address the U.S. Nuclear Regulatory Commission (NRC) staff's requests for additional information (RAIs) [Refs. 7, 8, 9, 10] to which KHNP responded [Ref. 11]. This TR describes the applicant's evaluation methodology (EM) for the analysis of a large-break loss-of-coolant accident (LBLOCA) for the Advanced Power Reactor 1400 (APR1400) design. The applicant will reference the TR in future licensing actions, including the APR1400 Design Certification (DC) application.

The TR describes the applicant's calculation methodology used to evaluate the LBLOCA event for the KHNP APR1400 nuclear steam supply system (NSSS) during the blowdown, refill, and reflood stages. This TR does not address long term core cooling following such an event. The KHNP methodology for long-term core cooling is addressed separately in Technical Report APR1400-F-A-NR-14003-P, "Post-LOCA Long Term Cooling Evaluation Model" [Ref. 2].

As part of the NRC evaluation of the TR, the NRC staff identified a number of technical issues that were discussed with the applicant as Audit Issues (AI) [Ref. 3] to which KHNP responded [Ref. 4, 5, 6] .[2] The NRC staff and KHNP subsequently discussed these RAIs and AIs during and following several NRC Audit Meetings [Ref. 12]. These discussions ultimately resulted in the issuance of TR Revision 1 [Ref. 13] by the applicant.

The objective of this Safety Evaluation Report (SER) is to provide an assessment of the technical adequacy of the methodology described in the TR for use in LBLOCA licensing calculations. The determination of technical adequacy is based on the NRC guidance and regulations as outlined in:

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Compiled 2007, (the SRP) - specifically Chapter 15 [Ref. 14]
- Title 10 *Code of Federal Regulations* (10 CFR) 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors" [Ref. 15]
- Regulatory Guide (RG) 1.157 "Best-Estimate Calculations of Emergency Core Cooling System Performance," [Ref. 17]
- RG 1.203 "Transient and Accident Analysis Methods" [Ref. 18]
- NUREG/CR-5249, "Code Scaling, Applicability and Uncertainty" (CSAU) [Ref. 19]

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[1] In this SER, TR refers to Topical Report APR1400-F-A-TR-12004, and is not used simply as an abbreviation.

[2] The attachments to Reference [3] contain material in the Korean language; however, the information contained in the also-attached English language documents was found to be adequate by the staff, and were relied upon for the purpose of the present review and assessment.

2. REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, "Domestic Licensing Of Production And Utilization Facilities," Section 46, Paragraph (a) specifies that each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical Zircaloy or ZIRLO cladding must be provided with an Emergency Core Cooling System (ECCS) designed so that the calculated cooling performance following a postulated LOCA conforms to the criteria set forth in 10 CFR 50.46(b). The regulations in 10 CFR 50.46(a) also stipulate that the requirements can be met through an evaluation model (EM) for which an uncertainty analysis has been performed:

> *...the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded.*

The regulation 10 CFR 50.46(b) requires that:

- the Peak Cladding Temperature (PCT) shall not exceed 2200°F
- the calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation
- the calculated amount of hydrogen generated from chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react
- calculated changes in core geometry shall be such that the core remains amenable to cooling
- calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The NRC has provided guidance on how these requirements can be met. RG 1.157 and NUREG/CR-5249 describe acceptable approaches that may be used to determine the uncertainty in the calculated parameters.

Additional guidance is provided in Standard Review Plan (SRP) Section 15.0.2, "Review of Transient and Accident Analysis Methods." Also, the guidance in RG 1.157 applies to the review of best-estimate methodology.

The information in the SRP is complementary to RG 1.203, which provides guidance to applicants on the standards expected by the NRC regarding transient and accident methods submittals. In SRP Section 15.0.2 and RG 1.203, the review of methods for the analysis of

transients and accidents includes a review of the submitted (or modified) EM consisting not only of the computer program(s) used in the analysis, but also of all associated aspects of the calculation framework, including mathematical models, assumptions, assignment of parameter values, and the treatment of input and output data. Both documents clearly delineate the items to be included in the submittal and the acceptance criteria upon which the applicant's documentation is evaluated.

RG 1.157 provides specific guidance for applicants regarding best-estimate accident analysis of emergency core cooling (ECC)[3] systems, which is clearly applicable to this TR. This review is conducted in accordance with the details in RG 1.157 and SRP 15.0.2.

As stated in SRP Section 15.0.2:

> *…licensees must analyze transients and accidents in accordance with the requirements of 10 CFR 50.34, 10 CFR 50.46, and where applicable, per NUREG-0737, "Clarification of TMI Action Plan Requirements." ... This section of the SRP describes the review process and acceptance criteria for analytical models and computer codes used by licensees to analyze accident and transient behavior… This review is independent of the type of application submitted (e.g., license amendment, topical report, design certification, or combined license application).*

This review documents the NRC staff evaluation of the applicant's TR, which describes a best-estimate ECCS Code-Accuracy-Based Realistic Evaluation Methodology (CAREM) and its associated EM.

SRP Section 15.0.2 lists six individual areas of review for transient and accident analysis methods. Discussed in each area are the criteria to be used to evaluate the submittal. These review areas are:

- Documentation
- Evaluation Model
- Accident Scenario Identification Process
- Code Assessment
- Uncertainty Analysis
- Quality Assurance Plan

The following sections provide an overview of how these six individual topics were considered in this evaluation.

2.1 Documentation

The SRP Section 15.0.2 states that the documentation must be scrutable, complete, unambiguous, accurate and reasonably self-contained in how it describes the codes and methods present in the EM. The SRP recommends that certain topics be specifically addressed:

- An overview of the methodology and associated EM

[3] The terms 'ECC', 'ECCS', and 'ECCW' are used interchangeably throughout this document, as well as in the applicant's documentation. All refer to the emergency core cooling system.

- A complete description of the accident scenario
- A complete description of the code assessment
- A determination of the code uncertainty
- A theory manual
- A user manual
- A quality assurance plan

Each of the applicable elements of the documentation must be accurate, complete, and consistent. The users' manuals in particular must provide clear and unambiguous guidance (on a scenario-specific basis, if applicable) for selecting or calculating all input parameters and code options.

Evaluation Model

The EM includes all computational and non-computational elements, including not only the computer program at the core of the methodology but also field equations, constitutive and closure relations, and simplifying assumptions used to perform the accident analyses. All of these parts of the methodology should be reviewed to determine their applicability, adequacy, and correctness.

2.2 Accident Scenario Identification Process

This review criterion recommends that applicants use a structured process for the identification and ranking of physical phenomena relevant to the accident scenarios to which the EM will be applied. This process is used to determine the importance of the phenomena by examining their effect upon the figures-of-merit (FOM) calculated. The guidance states that each accident scenario should include a complete and accurate description of its initial and boundary conditions. It should also be confirmed that phenomenological models present in the EM are consistent with the levels of importance attached to the phenomena.

2.3 Code Assessment

The guidance states that the assessment of the codes and code models present in the EM should be conducted over the full range of conditions to which they will be applied and use the same code options that would be employed in actual plant calculations. This assessment process should include separate effects testing, integral effects testing, code-to-code comparisons, and/or comparisons against the results of analytical models. Where applicable, the assessment process should also include scaling analysis to identify any non-dimensional parameters of the analysis, and should evaluate the impact of scaling upon the assessment. Comparisons should be sufficiently thorough such that the possibility of compensating errors is eliminated.

In the code assessment process, the review should confirm that the numerical solution conserves all important quantities and that code options are used appropriately.

Any closure relationships based on experimental data or more detailed calculations must have been assessed by the applicant over the full range of conditions expected to be encountered in the accident scenarios for which the EM will be used.

2.4 Uncertainty Analysis

The applicant should demonstrate in the submittal that all of the important sources of uncertainty in the codes, mathematical models, and experimental data comprising the EM are addressed. In addition, the combined uncertainty of the EM should be less than the design margin for safety of the resulting figures-of-merit. The process used should ensure that the resulting uncertainties in the EM do not exceed the design margins for the safety parameters of interest. Important sources of uncertainty should be identified using the accident scenario identification process.

2.5 Quality Assurance Plan

The guidance states that the applicant shall maintain the EM under a quality assurance program that meets the requirements of 10 CFR Part 50, Appendix B. The SRP guidance includes provisions to ensure that the documentation of the EM includes consideration of the issues of design control, document control, software configuration control and testing, and corrective actions and that independent peer reviews should be conducted at key steps in the development of the EM.

3. SUMMARY OF THE APPLICATION

A brief summary of the content of the TR is provided in this section.

3.1 Overview

In TR Section 1, the applicant introduced their LBLOCA methodology for APR1400, which is referred to as the code-accuracy-based realistic evaluation methodology (CAREM). According to the applicant, CAREM conforms to the guidance in RG 1.157, and consists of three elements and 14 steps as outlined in NUREG/CR-5249. An overall flow diagram of CAREM is shown in Figure 3-1, which is reproduced from TR Figure 1-1.

The TR states that one of the purposes of CAREM is to allow the applicant to quantify the uncertainties in the calculation of the acceptance criteria. This is accomplished in CAREM by propagating the uncertainty of each parameter deemed important for predicting the acceptance criteria. The TR states that CAREM utilizes the PCT, the maximum local oxidation (MLO), and the core-wide hydrogen generation as the LBLOCA acceptance criteria, which is consistent with the requirements of 10 CFR 50.46(b).

The TR states that CAREM is applied using modified versions of the RELAP5/MOD3.3 thermal-hydraulics code [Ref. 20] and the CONTEMPT4/MOD5 containment analysis code [Ref. 21] for the calculation of the APR1400 system thermal-hydraulics and containment back-pressure, respectively. The applicant performed and reported a demonstration calculation for the reference APR1400 plants, Shin-Kori Nuclear Power Plant (SKN) Units 3 and 4 using CAREM with these codes. In addition, the TR includes references to analyses performed by the applicant with previous RELAP5 versions, MOD2 [Ref. 22] and MOD3 [Ref. 23], which have their own unique characteristics, as described in the TR.

TR Section 1 provides a brief overview of the APR1400. It is a pressurized water reactor (PWR) design utilizing a direct vessel injection (DVI) system, a safety injection tank equipped with a fluidic device (SIT-FD), and an in-containment refueling water storage tank (IRWST). Otherwise, it retains the principal features of conventional 2-loop PWRs. Each loop consists of a hot leg, a steam generator (SG), two pump suction legs, two reactor coolant pumps (RCP), and two cold legs.

TR Table 1-1 lists the major system parameters of the APR1400. The core thermal power is 3,983 MW(t), and mass flow rates of the reactor coolant system (RCS) and core are 20,991 kg/s, and 20,361 kg/s respectively (with 630 kg/s through the core bypass region). The active core is 3.81 m long and consists of 241 fuel assemblies. Each fuel assembly utilizes the applicant's proprietary PLUS7 fuel design [Ref. 25], which employs a 16×16 lattice.

The TR states that certain key features of the APR1400 ECCS components and their design are directly relevant to the LBLOCA analysis. They include:

- Direct Vessel Injection (DVI) for all trains of ECCS components,
- Four independent trains of high-pressure safety injection pumps (SIPs),
- Four SIT-FDs,
- Elimination of low-pressure SIP due to the use of the fluidic device, and
- An IRWST which provides the water inventory (source) for the SIPs.

According to the TR, each of the four SIT-FDs and four trains of SIPs inject all the emergency core cooling water (ECCW) into the upper annulus of the reactor pressure vessel (RPV) through the respective DVI nozzles for each combination of SIT-FD and SIP. The four trains of the SIPs are designed to be mechanically and electrically independent. Four emergency diesel generators (EDG) independently provide the power to each SIP. According to the single failure analysis supplied in the TR, three SIPs are assumed operable. However, the analysis presented in the TR assumes that only two SIPs are operational for additional conservatism.

The TR describes the SIT-FD. The fluidic device controls the injection flow as a function of the water level inside the SIT. TR Figure 1-4 shows a schematic diagram of the fluidic device. It is a combination of a standpipe and a vortex chamber. The applicant states in the TR that the SIT-FD provides high injection flow, equivalent to that of a traditional SIT (i.e., accumulator) when the water level is higher than the top of the SIT standpipe. Once the water level falls at or below the top of the SIT standpipe, the SIT-FD provides low injection flow via the vortex chamber with performance that is similar to a conventional low-pressure SIP.

The TR states that refueling water, which is the source of the safety injection, is stored within a lower peripheral region of the containment building. This design is intended to eliminate the need for the traditional switchover from injection to recirculation, and eliminates the potential for breaks outside of containment through the ECCS pathways.

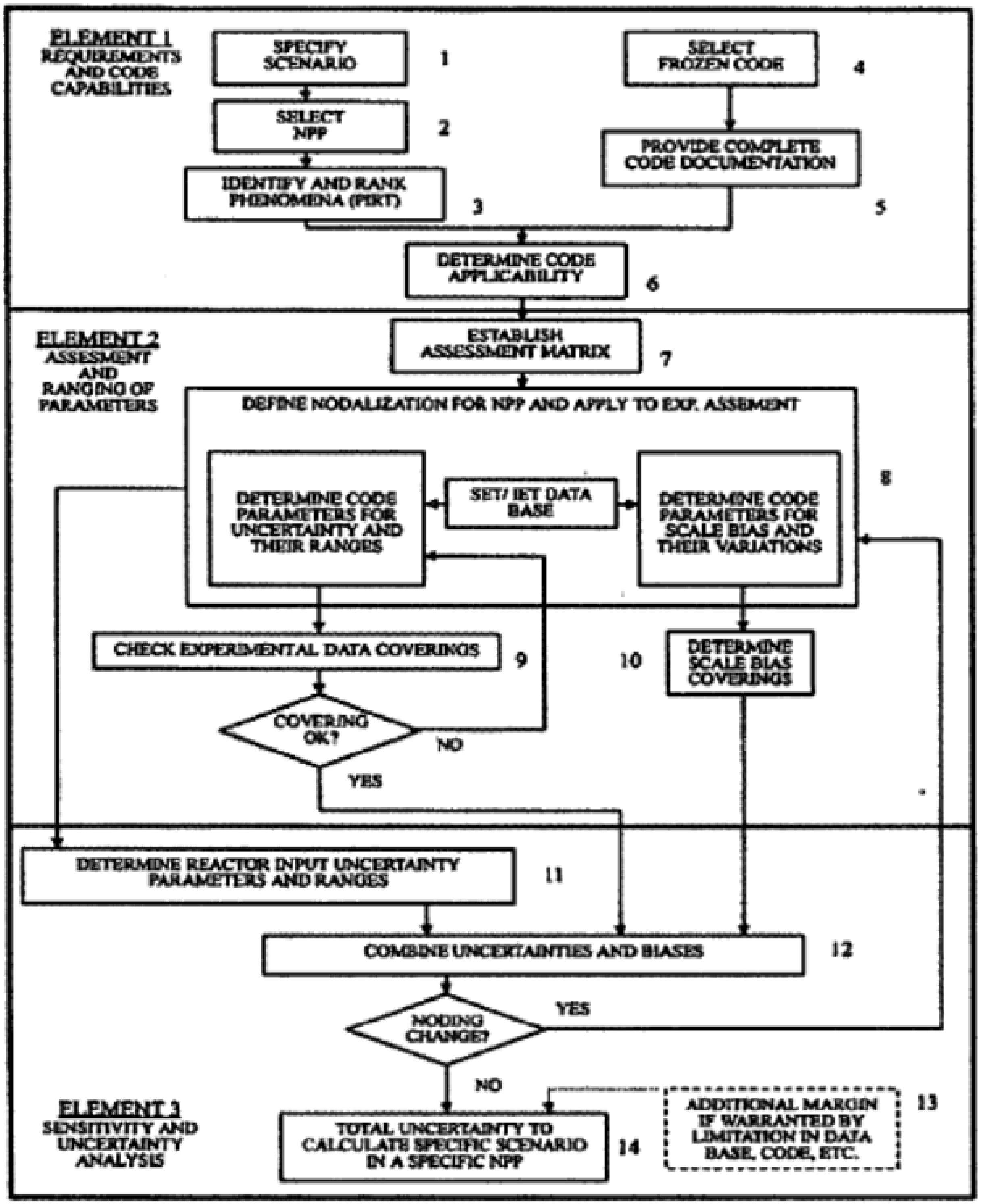

Figure 3-1 CAREM Flow Diagram

3.2 Method Roadmap

TR Section 2 describes the organization of the TR, as well as various features and steps associated with the CAREM methodology. This description states that the three elements of CAREM are described in subsequent sections. TR Sections 3 through 5 address each element of CAREM that is reproduced in Figure 3-1.

3.3 Requirements and Capabilities

TR Section 3 describes in detail the first element of CAREM, which is the specification of the requirements of the code capabilities and the determination of the code applicability. The first element consists of six steps and each one of these steps is discussed sequentially.

3.3.1 Scenario Specification (Step 1)

TR Section 3.1 discusses the scenario specification, which constitutes the first step in CAREM (see Figure 3-1). In order to determine the limiting scenario, the scenario that maximizes the impact on the FOMs, the applicant performed several pipe break sensitivity studies. The sensitivity studies covered the types of pipe breaks (double-ended guillotine breaks and split breaks), the locations of the breaks (i.e., cold leg, hot leg, and pump-suction leg), the break areas, and the assumptions employed. The applicant assumed that the loss-of-offsite power (LOOP) occurs coincident with the break, the RCPs coast down at the initiation of the accident due to LOOP, an emergency diesel generator (EDG) fails, and only credits the contribution of two of the four SIPs. The applicant uniformly used these assumptions in the sensitivity studies to determine the limiting break.

The applicant concluded that the highest peak cladding temperature (PCT) occurs in the case of the double ended guillotine cold leg break. This defines the limiting break type, location, and area. TR Sections 3.1.2 and 5.2.1.1 and Appendix K provide additional details.

The TR describes the scenario for the limiting break as being subdivided into four time-periods that characterize events during the accident. These time periods are (a) blowdown, (b) refill, (c) early reflood, and (d) the late reflood phases. These periods are defined by the water inventory in the reactor pressure vessel and the injection flow condition of the SIT-FD.

The TR defines the blowdown period as being from the occurrence of the break to the start time of the SIT-FD injection.

The TR defines the refill period as being from the start time of the SIT-FD injection to the time when the mixture level in the reactor vessel lower plenum reaches the bottom of the active core.

The TR defines the reflood period as being subdivided into early and late periods based on the time at which the SIT-FD depletes. This is to address the possibility of core heatup during the late reflood period since the APR1400 design does not include low pressure SIPs.

3.3.2 Nuclear Power Plant Selection (Step 2)

The applicant selected Shin-Kori (SKN) Units 3 and 4 as the reference plant in TR Section 3.2.

3.3.3 Phenomena Identification and Ranking Table (Step 3)

TR Section 3.3 introduces the phenomena identification and ranking table (PIRT) that is used to identify the major phenomena or processes that occur during the accident and to prioritize them according to their effects on the FOMs. The applicant stated that the PIRT for CAREM is derived from the Korea Next Generation Reactor (KNGR) PIRT. The KNGR PIRT was prepared by the Idaho National Engineering and Environmental Laboratory (INEEL) [Ref. 52], and KNGR is the name given during the developmental stage to what eventually became the APR1400 design.

TR Section 3.3 states that the KNGR and APR1400 PIRTs are both based upon the double-ended guillotine break of a cold leg with the same conditions as described above. These PIRTs consist of four characteristic time periods, however the definitions of time periods are different. APR1400 PIRT defines four characteristic time periods consisting of the blowdown, refill, early reflood, and late reflood. Each phenomenon or process considered in the KNGR and APR1400 PIRTs is ranked during each of the accident phases using 1 (low) to 5 (high) importance levels [Ref. 17]. The applicant modified some of the rankings in the KNGR PIRT when constructing the APR1400 PIRT to reflect the design finalization of APR1400 and new experimental data.

TR Table 3-2 provides the APR1400 PIRT. TR Appendix A provides details of the KNGR PIRT and the changes made to it resulting in the APR1400 PIRT.

As stated in TR Section 3.3, the distribution free statistical approach used in this methodology imposes no limitation on the number of uncertainty parameters. The ranking technique discussed in the TR ensures that those parameters that have a numerical rank of [[ ]] are used in the applicability and uncertainty analysis.

3.3.4 Frozen Code Selection (Step 4)

TR Section 3.4 states that the RELAP5/MOD3.3/K code used as the basis for the application is a modified version of RELAP5/MOD3.3 that is dynamically linked to the CONTEMPT4/MOD5 computer code by the applicant. This system of codes calculates the reactor system thermal-hydraulic conditions and the containment backpressure, respectively.

TR Appendix B documents several areas where the applicant modified the code to achieve certain improvements which include:

- [[
-
-
-]]
- and code failure due to [[ ]]

TR Appendix B displays the results from several cases demonstrating the degree of improvement achieved.

TR Appendix B also documents comparisons of RELAP5/MOD3.3/K calculations and Full-Length Emergency Core Heat Transfer for the Separate Effects and Systems Effects Tests (FLECHT-SEASET) measurements. This evidence of RELAP5/MOD3.3/K acceptability adds to the already substantial volume of comparisons from other RELAP5/MOD3.3/K assessments.

The TR continues by describing some of these assessments in the original RELAP5/MOD3.3 supporting documents [Ref. 20].

Volume III [Ref. d] documents "Developmental Assessment Problems" used to characterize RELAP5/MOD3.3 performance, and several classes of comparisons are documented therein. This assessment includes identification of some problem areas and limitations:

- Developmental Assessment Problems
- Phenomenological Problems
- Separate Effects Problems
- Integral Effects Problems

Volume 6 [Ref. g] documents the validation of RELAP5/MOD3.3 numerical techniques via a comprehensive battery of tests. Volume 6 comprehensively demonstrates the acceptability of the models employed, the domain of applicability, and the regions of solution stability. In addition, Volume 6 provides comparisons with several analytical solutions to support this validation.

Volume 7 [Ref. h] documents the wide variety of assessments that were accomplished independent of the NRC development and validation effort. A very diverse group of organizations, under the banner of the Organization for Economic Co-operation and Development and Nuclear Energy Agency (OECD/NEA), performed these assessments, and included:

- Code Applications and Maintenance Program (CAMP) members

- International Laboratories (e.g., Korea Institute of Nuclear Safety, Kurchatov Institute, etc.)

- Reactor vendors and licensees (e.g., Westinghouse, TRACTEBEL, STUDSVIK, Kraftwerk Union AG, etc.)

- Experimental facilities (e.g., BETHSY [Boucle d'Etudes Thermo-Hydraulique Système], LOFT [Loss of Fluid Test], FLECHT-SEASET, etc.)

- Independent researchers (e.g., Paul Scherrer Institute)

The Volume 7 assessment [Ref. h] pertains to an earlier version (i.e., version 3.0) of RELAP5 and it identified a number of deficiencies and areas for improvement. RELAP5/MOD3.3/K used by the applicant implements those improvements.

CONTEMPT4/MOD5 is a containment analysis code. As implemented by the applicant, it is used for calculating the LBLOCA containment backpressure utilizing input conditions supplied by RELAP5. This code includes models for the fan and spray cooling systems and the passive heat sinks (i.e., heat transfer to various containment structures) which are essential to the calculation of containment backpressure during a LBLOCA.

The application describes the merger of these two codes so that they may exchange information about the containment backpressure and the mass and energy release rate at every time step after a specific set of conditions are achieved.

3.3.5 Code Documentation (Step 5)

TR Section 3.5 provides a list of code manuals for RELAP5/MOD3.3 and CONTEMPT4/MOD5. The applicant asserts that because RELAP5/MOD3.3/K only includes relatively simple modifications to the original RELAP5/MOD3.3 code, the code manuals for RELAP5/MOD3.3 remain applicable [Ref. 20].

3.3.6 Code Applicability Determination (Step 6)

TR Section 3.6 discusses the analytical requirements of the code system. This section compares the features of RELAP5/MOD3.3/K and CONTEMPT4/MOD5 to these requirements by the applicant, where they conclude the following:

- *"The code system has the capability to simulate the four phases of blowdown, refill, early reflood, and late reflood of an APR1400 LBLOCA.*

- *The code system can approximate the multidimensional phenomena in the reactor pressure vessel by way of modeling multiple channels and their inter-connecting cross-flow junctions.*

- *The nonhomogeneous and non-equilibrium governing equations and the non-condensable gas transport equation, together with the constitutive relations of the code system can calculate such major phenomena as the critical flow at the break, dynamic characteristics of the centrifugal pump including the two-phase performance degradation, blowdown core heat transfer, penetration or bypass of ECC water in the downcomer, reflood core heat transfer, fuel rod quenching, droplet entrainment, two-phase pressure drop, containment backpressure, and so on.*

- *The code system contains a series of constitutive relations for all flow regimes.*

- *The code system contains important special component models for components such as the safety injection tank and separator.*

- *The code system can simulate plant geometry and control functions.*

- *The code system shows reasonable agreement qualitatively and quantitatively with various experimental data.*

- *Code assessment calculations performed against experiments at various scales can assure the scale-up capability of the code system to full-scale power plants.*

- *The frozen code system can be applied to the quantification of the calculational uncertainty, and is supported by a complete set of code documentation.*

- *Assessment calculations of the code system and their databases are available."*

3.4 Assessment and Ranging of Parameters

TR Section 4 states that the CAREM methodology evaluates the uncertainties of the models or correlations that it uses to simulate the major phenomena in PIRT. Further, uncertainty ranges

of individual parameters can be determined from information such as experimental data, reference documents, engineering judgments, code calculations, etc. The TR states that in some cases the uncertainty may be treated as a separate bias, bounding approach, or nodalization and for phenomena that are governed by the same model or correlation, the uncertainty is applied so as to ensure conservative predictions of PCT.

The TR defines code accuracy as a statistical difference (i.e., bias) and deviation obtained by comparing the calculated cladding temperature to that measured in experiments.

3.4.1 Establishment of Assessment Matrix (Step 7)

TR Section 4.1 describes the matrix of objectives used to select the separate effects tests (SET) and integral effects tests (IET) that establish the accuracy of various phenomena models, component performance attributes, and system behaviors. TR Section 4.1 states that the assessment matrix used for the APR1400 includes a combination of tests for traditional cold leg injection (CLI) plants, as well as those more specific to the APR1400 design. The tests for CLI plants are relevant because several phenomena that are expected to be important for the APR1400 LBLOCA are commonly important to CLI plants as well. However, it goes on to explain that since the APR1400 design also includes DVI, phenomena that occur in the downcomer region, such as ECC water bypass and void development in the downcomer region (i.e., downcomer boiling), can be significantly different from those in CLI plants. For this reason, the assessment matrix includes tests that investigate ECC water bypass and downcomer boiling phenomena:

- Multi-dimensional Investigation in Downcomer Annulus Simulation (MIDAS), Upper Plenum Test Facility (UPTF) 21D, which addresses the ECC water bypass phenomenon;

- Downcomer Boiling (DOBO), which addresses the downcomer boiling phenomenon;

- Advanced Thermal-hydraulic Test Loop for Accident Simulation (ATLAS), which addresses the general LBLOCA reflood behavior of the APR1400;

- Valve Performance Evaluation Rig (VAPER), which addresses the SIT-FD injection flow behavior.

3.4.2 Plant Nodalization and Experimental Assessment (Step 8)

3.4.2.1 Plant Nodalization (Step 8.1)

TR Section 4.2.1 discusses the nodalization of the APR1400 plant used in the CAREM for demonstration calculations and shows it graphically in TR Figure 4-1. The two hot legs and the four cold legs are modeled separately. The two loops are modeled identically except that the pressurizer is attached to one of the loops. The other loop contains the cold leg break between the pump and the reactor vessel which is the limiting break location. The noding for each loop describes the hot leg piping, U-tubes, and fluid volumes of the steam generator primary and secondary sides, two pump suction legs, two pumps, and two pump discharge cold legs.

The TR states that the reactor vessel nodalization of the APR1400 includes the downcomer, upper and lower plena, and the core regions. It states that the nodalization follows the typical

PWR approach except for the core and the downcomer components which include additional detail in the nodalization to allow calculations to capture the various thermal-hydraulic phenomena and multidimensional behaviors.

In TR Section 4.2.1 the reactor vessel downcomer is divided into [[ ]]. The applicant selected [[ ]] through sensitivity calculations that determined the number of channels required to prevent distortion of the phenomena, and to coincide with the loop piping connections. The [[ ]] are interconnected by cross-flow junctions. The channel form loss coefficient was modeled to reflect the geometric shapes as a function of the downcomer width, roughness, radius of curvature, and friction coefficients. The height of the upper annulus node to which the DVI nozzle is connected was selected to locate the node center at the DVI nozzle elevation. The TR states that the hot wall effect is acceptably simulated by modeling heat structures of the downcomer wall.

The TR states that the core is modeled with [[ ]]. The applicant determined that the active length of the fuel rod was acceptably modeled using [[ ]] axial nodes through the code assessment against the Thermal Hydraulic Test Facility (THTF) and FLECHT-SEASET tests.

3.4.2.2 Determination of Code Uncertainty Parameters and their Ranges (Step 8.2)

TR Section 4.2.2 describes the determination of RELAP5/MOD3.3/K input parameters that are selected for use under CAREM, what uncertainties and distributions the applicant believes are applicable to them, and why. This information is used to support the evaluation of overall calculation uncertainty for the plant sensitivity calculations. Plant input parameters and their uncertainties are discussed separately in TR Section 5.1.

TR Subsection 4.2.2.1.1 discusses how the fuel pellet thermal conductivity affects the initial stored energy. Using the CSAU approach, the applicant determined that the four dominant parameters affecting the fuel stored energy were the gap conductance, fuel rod peaking factors, fuel thermal conductivity and convective heat transfer coefficients. They determined that the gap conductance is dependent on the initial and transient fuel rod performance parameters of geometry, gap gas composition, burnup, etc. To address the effects of thermal conductivity degradation (TCD) in CAREM, the gap conductance is not treated as an uncertainty variable since the steady state fuel centerline temperature is adjusted to match fuel rod design information when the TCD effect is considered. The applicant determined a penalty which they apply to the fuel centerline temperature in the fuel performance design interface to account for TCD.

TR Subsection 4.2.2.1.1 discusses how the fuel pellet heat transfer is affected by the thermal conductivity of UO_2 pellets. The thermal conductivity of the fuel pellet is modeled by the modified Nuclear Fuel Industries (NFI) correlation to consider thermal conductivity degradation (TCD) as a function of burnup. The modified NFI correlation and related uncertainty is discussed in the FRAPCON-4.0 code document [Ref. 24].

TR Subsection 4.2.2.1.1 discusses how RELAP5/MOD3.3/K uses the Dittus-Boelter correlation to calculate turbulent convection heat transfer to single-phase liquid and vapor. The applicant referred to discussions in various references where 95 percent of the data have been

encompassed in an uncertainty range of [[ ]] percent. Consequently, for CAREM the applicant applied that as the uncertainty in the Dittus-Boelter correlation for prediction of single-phase liquid and vapor heat transfer and assumed the applicability of a normal distribution function for this uncertainty component.

TR Subsection 4.2.2.1.2 discusses how RELAP5/MOD3.3/K calculates the nucleate boiling heat transfer coefficient using the Chen correlation. According to the RELAP5 code manual, the average uncertainty is ± 21.1 percent in saturation nucleate boiling, and ± 40 percent in subcooled nucleate boiling. For CAREM, the applicant assumed a normal distribution with a standard deviation of [[ ]] percent for this parameter.

TR Subsection 4.2.2.1.2 discusses how RELAP5/MOD3.3/K calculates the critical heat flux (CHF) in the high-pressure and high-flow region using the Groeneveld lookup table. In TR Figure 4-2, a comparison of 1993 data points covering various test conditions, including those of high-pressure and high-flow, shows that the Groeneveld lookup table estimates the critical heat fluxes lower than the experimental data by 4.9 percent on average with a root-mean-squared error of 37.9 percent. The TR goes on to state that no comparison between the prediction from the lookup table and the data measured in only the high-pressure and high-flow condition is available. Therefore, for CAREM the applicant assigns the root-mean-squared error of [[ ]] percent as the standard deviation of the uncertainty of CHF in the high-pressure and high-flow region with a normal distribution.

TR Subsection 4.2.2.1.2 discusses how the CHF model of the RELAP5/MOD3.3/K code for the low-pressure and low-flow region is the modified Zuber correlation. The applicant demonstrated in TR Figure 4-3 that the modified Zuber correlation conservatively predicts 75 experimental data points. Furthermore, the correlation predicts 95 percent of the data in a range from [[ ]] percent. However, CAREM uses an uncertainty range for this correlation of [[ ]] percent, which is more conservative, and assumes a normal distribution.

TR Subsection 4.2.2.1.2 discusses how the reflood model in RELAP5/MOD3.3/K uses the maximum of the film boiling heat transfer coefficients calculated by the Bromley and the Forslund-Rohsenow correlations. Analysis using CAREM will select these two correlations as uncertainty parameters for film boiling heat transfer. The uncertainty in the Forslund-Rohsenow correlation in [Ref. 19] and code assessment results documented in TR Appendix C show that the uncertainty range for the Forslund-Rohsenow correlation is [[ ]]. The RELAP5/MOD3.3 manual states that the calculated film boiling heat transfer coefficients using the Bromley correlation covers all data within ±18 percent. The applicant cited a different study wherein the experimental data - which includes conduction, convection, and radiation heat transfer - are 12.1 percent higher than the RELAP5 predicted values on average, and the data are distributed within 0.37 to ~ 2.61 times of the predicted values. Based on these results, the applicant determined that for CAREM, the standard deviation of the Bromley correlation is [[ ]].

TR Subsection 4.2.2.6.1 discusses how the RELAP5/MOD3/K code uses the Ransom-Trapp critical flow model to predict critical flow rate. The applicant determined the uncertainty of this model using an assessment against the nine MARVIKEN tests. The assessment showed that the Ransom-Trapp critical flow model over-predicts the data by [[ ]] percent for the subcooled single-phase critical flow and under-predicts the data by [[ ]] percent for the two-phase critical flow. The predicted values agreed well with the data when the discharge coefficients of [[ ]] were applied to the subcooled single-phase and to the two-phase critical flow, respectively. As a result, the applicant selected these values of the

discharge coefficient for CAREM as the mean values of the normal distribution. The standard deviations of the discharge coefficients of the subcooled single-phase and the two-phase flow are [[ ]], respectively.

TR Subsection 4.2.2.6.2 discusses how the containment temperature and pressure history is calculated conservatively. [[

]]. The TR further states that, due to the conservatively chosen CONTEMPT4/MOD5 input parameters, the uncertainties in various containment-related parameters or phenomena are not subjected to sampling using uncertainty distributions. Since the APR1400 CAREM employs a mixture of best-estimate and conservative assumptions, for the post-blowdown containment backpressure calculation, the input parameters to the CONTEMPT4/MOD5 computer code are selected conservatively to result in a conservatively low containment backpressure.

TR Subsection 4.2.3.1, the applicant discusses how several phenomena of importance in the downcomer region that they attribute to the ECC water bypass behavior, such as condensation, multidimensional flow, steam velocity, entrainment, de-entrainment, inter-phase momentum transfer, and countercurrent flow. Since it is difficult to determine one specific code parameter that governs ECC water bypass, CAREM captures the combined uncertainties related to the ECC water bypasses as a bias.

TR Subsection 4.2.3.1 discusses how the applicant used code assessments to determine the ECC water bypass bias during the refill period. To accomplish this, the applicant conducted tests at the Upper Plenum Test Facility (UPTF) and the ATLAS. TR Appendix F describes the UPTF tests, while TR Appendix E describes the ATLAS tests. The applicant performed an assessment against a full-scale UPTF-4A test, and compared the predicted lower plenum inventory with the measured data. The UPTF-4A test simulated the bypass phenomena during the refill period of a LBLOCA in a plant with CLI configuration. According to the applicant, the thermal-hydraulic behavior that occurred in the reactor pressure vessel downcomer during the refill period of the APR1400 LBLOCA is similar to that experienced in a conventional CLI PWR. This justified the use of the UPTF-4A test data. The code assessment revealed that the water inventory in the lower plenum predicted at the end of the test exceeded the measured amount by several hundred kilograms. The applicant conservatively interpreted [[ ]] as a measure of the ECCW bypass during the refill period. For CAREM, the applicant chose to evaluate the bias in the prediction of the PCT due to ECCW bypass [[ ]].

TR Subsection 4.2.3.1 discusses how the applicant evaluated the code's ability to predict ECC bypass phenomena during the reflood period through code assessments against tests in the Multi-dimensional Investigation in Downcomer Annulus Simulation (MIDAS) facility and the UPTF-21 D, and ATLAS Tests 9, 11, and 15. ATLAS is an integral test facility simulating the APR1400, whereas MIDAS and UPTF-21 D tests are separate effects tests designed to investigate ECCW bypass during the reflood phase for DVI configurations. TR Appendices E and F provide details of the code assessment against the ATLAS tests, and the MIDAS and UPTF 21-D tests, respectively.

TR Subsection 4.2.2.2.1 states that the results of the code assessment studies against the above-mentioned tests showed that the RELAP5/MOD3.3/K code conservatively calculates ECC bypass during reflood. The TR asserts that, since the tests used in the code assessment

include a full-scale UPTF test, the conservative prediction of RELAP5/MOD3.3/K can be considered to be valid for full-scale APR1400 LOCA events. The conservative or favorable bias for ECCW bypass during the reflood phase is not credited by the applicant in the CAREM evaluation of the PCT bias of the plant sensitivity calculations. Therefore, the applicant did not determine the code parameters or measures necessary to evaluate the bias of the ECCW bypass during reflood. As a result, the applicant applies the ECCW bypass bias only during the refill phase of the accident.

TR Subsection 4.2.3.2 discusses the importance of accounting for the steam binding bias and the source of this concern. Steam binding results due to the vaporization of droplets entrained into the steam generator U-tubes. The droplets are vaporized by the heat transfer from the secondary side to the primary side during the reflood period. The amount of water droplets that enter the U-tubes is determined by the entrainment and de-entrainment in the upper plenum and hot legs. Water droplets entrained from the upper plenum by the steam exiting the core can be de-entrained by the mechanical structures in the upper plenum. This entrainment and de-entrainment phenomena occurring across the upper plenum is inherently multidimensional. RELAP5/MOD3.3/K does not include a specific model to predict the mechanical de-entrainment by the structures. The applicant assessed the two distinct phenomena of entrainment and vaporization of the droplets independently in order to determine the relevant biases in CAREM analyses.

TR Subsection 4.2.3.2 discusses the entrainment and de-entrainment in the upper plenum as assessed by the UPTF-10B test. The applicant cited a previous assessment of RELAP5/MOD3.1 against UPTF-10B, which showed that mass inventory of the upper plenum can be [[

]]. Based on these results, the applicant determined that [[

]] for CAREM analyses.

TR Subsection 4.2.3.2 discusses the decision for CAREM analyses to maximize the effect of the vaporization of the droplets in the U-tubes by [[

]] at the same time.

3.4.3 Check Experimental Data Coverings (Step 9)

In TR Section 4.3, the applicant discusses the experimental data used to ensure the validity of the selected code uncertainty parameters. Step 9 examines whether the simulation calculations using the uncertainty parameters from the previous steps can successfully encompass the acceptance criteria of the experiments. The applicant performed these calculations for all the experiments in which the cladding temperatures were measured.

3.4.3.1 Code Accuracy Evaluation (Step 9.1)

In TR Section 4.3.1, the applicant defines code accuracy as the statistical difference between calculated and measured PCTs. Even if the locations and the occurrence times of the PCTs are different from each other, the code accuracy is defined simply as the PCT difference regardless of the location and when it occurs (i.e., code accuracy is defined in terms of PCT predictability).

In TR Section 4.3.1 the applicant uses standard definitions of parameters such as the mean and standard deviation to determine the statistical code accuracy based on the difference between the predicted and measured PCT. They state that the code accuracy, or the ability of the code to predict the PCT, needs to be reproducible, even if the uncertainties in the models used to predict PCT are propagated. As a result, the applicant uses the code accuracy as a reference value. If the resulting PCT from simple random sampling (SRS) calculations using the parameter uncertainties is higher than the maximum value of the measurements, then the number and probability distribution functions of the uncertainty parameters are sufficient. If the resulting PCT of SRS calculations cannot cover the maximum value of the measurements, the applicant will then need to examine further whether the resulting PCT covers the 95 percent one-sided limit.

TR Section 4.3.1.1 discusses four separate effects tests (THTF-105, 151, 160, and 162) and four integral effects tests (LOBI [LWR Off-Normal Behaviour Investigation] A1-66, LOFT L2-2, L2-3, and LP-02-6) to determine the code accuracy for the blowdown PCT. The TR states that blowdown experiments in facilities with DVI configuration are unavailable, partly due to the fact that the blowdown phenomena are almost independent of the injection system and therefore the necessity of DVI blowdown experiments is low. As a consequence, the applicant evaluated the blowdown code accuracy against the above mentioned blowdown experiments, which are in facilities with a CLI configuration. TR Appendices C and D describe code assessment calculations against separate effects tests and integral effects tests.

TR Section 4.3.1.2 discusses the evaluation of the code accuracy for reflood PCT. The applicant performed this evaluation using data from 27 separate effects tests (i.e., 17 FLECHT-SEASET tests, 7 NEPTUN tests, and 3 ATLAS tests) and 7 integral effects tests (i.e., LOFT L2-2, L2-3, L2-5, LP-02-6, CCTF C2-4, PKL-IIB5 [PWR Integral System Test Facility], and Semiscale S-06-3). TR Appendices C and D describe code assessment calculations against separate effects tests and integral effects tests.

3.4.3.2 Check Data Covering (Step 9.2)

TR Section 4.3.2 discusses the need to ensure that the spectrum of data employed in the evaluation is sufficient to justify the analysis objectives and that distribution-free statistics are utilized. The applicant showed that using distribution-free statistics requires 181 simulations in order to obtain at least five samples that exceed the 95th percentile at a 95 percent confidence level. In subsequent discussions in this SER this is then termed the 95/95 criteria, condition, limit, objective, or value depending on usage. In the context of this SER it means that the three most common results, the highest maximum cladding oxidation (MCO), the highest core wide oxidation (CWO), and the third ordered peak cladding temperature (PCT) are drawn from the domain of 181 LBLOCA calculations and they are 95/95 values.

TR Section 4.3.2.1 states that SRS calculations are performed against the test data obtained from THTF, LOFT, and LOBI in order to determine whether the calculations encompass the measured blowdown PCTs. To accomplish the coverage check, the applicant sampled eight code parameters and their associated uncertainties, from a larger potential list, to examine the blowdown test results:

- the [[ ]],
- the [[ ]],
- the [[ ]],

- the [[ ]],
- the [[ ]], and
- the [[ ]].

The applicant selected some additional system parameters that are considered for these tests since the LOFT and LOBI tests are integral effects tests that cover the reflood period as well. The applicant stated that the effects of these additional code and system parameters selected for the reflood period on the blowdown PCT calculation are minimal. The additional parameters used are listed in Appendices C and D for the THTF, and LOBI and LOFT, respectively.

The applicant confirmed that the third highest value out of the 181 calculations encompasses the upper-bound of the blowdown PCT measurements in the abovementioned tests based on the 181 calculations and the use of a distribution-free statistical approach.

TR Section 4.3.2.1 states that the applicant evaluated reflood results against the ATLAS, FLECHT-SEASET, LOFT, NEPTUN, Semiscale, PKL, and CCTF tests. The applicant employed eight code uncertainty parameters:

- the [[ ]],
- the [[ ]],
- the [[ ]],
- the [[ ]],
- the [[ ]], and
- the [[ ]].

The applicant also employed the use of system parameters specific to each test facility. TR Appendices C lists the additional parameters used (for FLECHT-SEASET, NEPTUN, PKL, and CCTF), D (for LOFT and Semiscale), and E (for ATLAS).

The applicant concluded that the third highest value out of the 181 calculations encompasses the upper-bound of the reflood PCT measurements in the above mentioned tests.

3.4.4 Determination of Scale Bias Covering (Step 10)

TR Section 4.4 states that the total amount of the ECC is conservatively estimated. TR Section 4.2.3 states that the effect of the downcomer water level depression in the reflood phase is not considered. TR Section 4.4.2 states that the RELAP5 code does not include a model to describe mechanical de-entrainment due to the internal structures in the upper plenum, the entrainment amount is [[ ]] the experimental measurements. The steam binding bias is evaluated by assuming the [[ ]] in the U-tubes. In this manner steam binding is maximized. TR Section 5.2 evaluates the effects of the steam binding bias on plant calculations.

3.5 Sensitivity and Uncertainty Analysis

TR Section 5 discusses the determination of the uncertainty in those parameters that are plant inputs, rather than phenomenological variables. The CAREM methodology provides the results of the 181 SRS calculations for the APR1400 plant to determine the limiting PCT.

3.5.1 Determination of Plant Input Uncertainty (Step 11)

TR Section 5.1 discusses the probability distribution functions and uncertainties of plant condition parameters. The overall uncertainty of the LBLOCA transient, including peak cladding temperature behavior, which is one of the major acceptance criteria, depends on the initial and boundary conditions of the plant, as well as the code uncertainty parameters. Plant initial and boundary conditions include core power distribution, fuel parameters, reactor coolant pump performance, safety injection system capacity and availability, system pressures and system flow rates.

TR Section 5.1.5 discusses the loss coefficient (K-factor) for the SIT-FD during high and low flow injection. The uncertainty in that parameter is not included based on the separate-effects test (VAPER) and sensitivity studies performed by the applicant. TR Appendix H provides the details of the VAPER test and the sensitivity studies.

3.5.2 Combine Uncertainties and Biases (Step 12)

The applicant performed a base case LBLOCA calculation to illustrate overall APR1400 system behavior in TR Section 5.2.1. The base case calculation of the plant applied the best-estimate operating conditions for the limiting break type, location, and size. The applicant defined the best-estimate conditions as the mean values of plant operating parameters and then performed a time step sensitivity study to determine that the variation of PCT is minimal and that the numerical stability is ensured if the maximum time step is smaller than [[ ]] seconds. Therefore, the applicant has committed to using a maximum time step restriction of [[ ]] seconds for CAREM calculations.

The applicant used the results of the base case calculation, along with the figures showing key system parameters during the accident, to describe the system behavior during the four accident phases.

The applicant performed demonstration calculations that included uncertainty variation in [[ ]] selected parameters that are listed in TR Table 5-1. The 181 demonstration calculations use SRS and distribution-free statistics. The applicant provided the third highest value of the calculated PCT from the 181 cases as the licensing value, and compared it against the 10 CFR 50.46(b)(1) acceptance criteria. TR Appendix J lists the 181 simple random samplings of [[ ]] uncertainty parameters utilized. In addition, the applicant also used several of the sampled cases to evaluate the scale bias due to ECCW and steam binding.

3.6 Quantification of Total Uncertainties (Steps 13 and 14)

TR Section 6 describes how CAREM is used to determine the limiting PCT, the maximum local oxidation, and the maximum hydrogen formation for the demonstration case which the applicant compared against the acceptance criteria. Therein, the applicant evaluated uncertainties from sources other than the code models or plant operational conditions, such as the automatic time-step control function and the data reading frequency of RELAP5/MOD3.3/K. The applicant added a fixed value of 10 K to the limiting PCT due to the difficulty in discerning the effects of these sources from the calculated cladding temperatures. This bias is based on the applicant's experience with using RELAP5 to support licensing applications in Korea. The applicant compared the final results against the acceptance criteria to demonstrate that there is sufficient margin in the calculated APR1400 behavior.

3.7 TR Appendix A: Identification and Ranking of Phenomena and Processes

TR Appendix A discusses the development of the APR1400 PIRT. The actual PIRT is provided therein as Table 5. This PIRT is a derivative taken from the KNGR PIRT prepared by INEEL in 2001 [52]. KNGR was the name given to what subsequently became the APR1400 design.

The KNGR PIRT was also based on the double-ended guillotine break of a cold leg. The PIRT considered the same four characteristic time periods of blowdown, refill, early reflood, and late reflood. The KNGR PIRT ranked each phenomenon or process considered during each of the accident phases using one of five importance levels. The KNGR PIRT is documented in TR Appendix A, Table 3.

As stated in TR Appendix A.2, the KNGR PIRT was developed on [[

]].

The applicant divided the LBLOCA scenario of the APR1400 into four temporal periods and described them as:

- The blowdown period starts when the break occurs and ends when SIT injection initiates;
- The refill period ends when the liquid level in the vessel lower plenum approaches the core inlet and remains full thereafter;
- The early reflood period ends when the SITs are emptied; and
- The late reflood period continues after the SITs are emptied.

The applicant also adjusted the relative importance of certain phenomena or processes after modifying the definition of temporal periods to account for the finalization of the APR1400 design and related experiments.

The applicant only considered the high ranked phenomena from the KNGR PIRT (with a numerical ranking of [[ ]] in any accident phase) to develop the APR1400 PIRT or modify the rankings. The rationale stated in TR Appendix A.3.2 is that the high ranked phenomena or processes are necessary for appropriate assessment of uncertainties. The uncertainty parameters that are identified for these parameters are then included in plant calculations. In cases where the identification of uncertainty parameters is not possible, those phenomena or processes are either treated conservatively or appropriate biases are determined and applied. The medium ranked phenomena (with a numerical ranking of [[’]] in any accident phase) can be modeled with moderate accuracy but corresponding uncertainty parameters are not determined. Phenomena or processes of low importance ranks (with a numerical ranking of [[ ]] in any accident phase) are not used to determine any uncertainties or biases.

Based on the above rationale, the applicant identified [[ ]] phenomena from the KNGR PIRT whose rankings were adjusted in response to the changes in the phase definitions. The result is the APR1400 PIRT listed in Table 5 of TR Appendix A.

3.8 Appendix B: Freezing of RELAP5/MOD3.3/K

TR Appendix B documents the changes made to the source code of RELAP5/MOD3.3 to develop RELAP5/MOD3.3/K. These include modifications for:

- the [[ ]],
- the [[ ]],
- the [[ ]],
- the [[ ]], and
- the [[ ]].

TR Appendix B.1 states that various assessments against SETs and IETs have revealed that the RELAP5/MOD3.3 code tends to predict that rod quenching occurs earlier than the experimental results. This tendency is mainly caused by the [[ ]].

TR Appendix B.1.1 describes changes to the RELAP5/MOD3.3 code to allow [[ ]]. The original implementation in RELAP5/MOD3.3 code allowed [[ ]], thereby resulting in earlier quenching. The modification is based on the [[ ]] in an older version (MOD3.1) of RELAP5. The predicted cladding temperature and quenching behavior after the modification show improved comparison against data from two FLECHT-SEASET tests.

TR Appendix B.1.1 describes how in RELAP5/MOD3.3 the calculation of the heat transfer coefficient to liquid during film boiling in bundles undergoing reflood is the maximum of the Forslund-Rohsenow correlation and the normal Bromley correlation. The final film boiling heat transfer coefficient is then obtained by adding the contributions from “the heat transfer coefficient to vapor” and “radiation to droplets” to the “heat transfer coefficient to liquid.” The applicant compared the heat transfer coefficients to liquid calculated by the three correlations (film coefficient, the Forslund-Rohsenow correlation, and the Bromley correlation) against corresponding data-derived values for two FLECHT-SEASET tests. The comparisons showed that the values predicted by [[ ]]. Thus, the film boiling heat transfer coefficient in RELAP5/MOD3.3/K is the maximum value between the [[ ]]. The predicted cladding temperature and quenching behavior predictions are improved using comparisons from two FLECHT-SEASET tests.

TR Appendix B.1.3 describes the detection of errors in the calculation of the [[ ]] in the RELAP5/MOD3.3 code. The applicant corrected these errors by modifying the source code.

TR Appendix B.1.5 describes modifications to RELAP5/MOD3.3/K to prevent calculational failures due to the manner in which RELAP5 models [[ ]].

TR Appendix B.2.1 describes the improvements to the reflood model due to modifications of the [[ ]] calculation via comparisons against SET data. The assessment calculations used FLECHT-SEASET tests, 31504 and 31108. These tests are representative of low and high reflood rate tests, respectively. The comparisons illustrate the improvements in heated rod cladding temperatures and the differential pressure. The applicant concluded that the modifications to the reflood model improve the prediction of heated rod quenching time, and the effects of the modifications on system hydraulic behavior are insignificant.

The applicant investigated the effect of the modification of [[ ]] calculation through sensitivity calculations. The applicant compared [[ ]] predicted by the modified and unmodified versions of RELAP5/MOD3.3 against the values calculated by a previous version, RELAP5/MOD3.1. Since the [[ ]] has remained unchanged from the previous version (MOD3.1), the [[ ]] calculated without the modification was expected to be the same as the results from RELAP5/MOD3.1. The calculated values, however, show a difference due to a coding error identified by the applicant. After modification by the applicant, the calculated values are restored to the RELAP5/MOD3.1 values. The applicant showed that after the modification to the [[ ]] calculation, the blowdown PCT decreased by about 6 K and the reflood PCT decreases by about 35 K with no significant difference in the heated rod quenching time.

3.9 Appendix C: Assessment of RELAP5/MOD3.3/K against Separate-Effect Tests

TR Appendix C documents the capability of the RELAP5/MOD3.3/K code to predict various thermal-hydraulic responses during blowdown and reflood phases using SETs. The objectives of the assessment are to:

- examine the code ability to accurately predict experimental data,
- determine code accuracy,
- evaluate probability distribution functions of the uncertainty parameters, and
- confirm the adequacy of the uncertainty parameters and their ranges.

As a result of the abovementioned objectives, TR Appendix C is heavily cited in the assessment and ranging of parameters in TR Section 4.

TR Appendix C.1 states that code accuracy is the difference between the calculated and the measured PCTs. The applicant confirmed the ability of the code to predict the thermal-hydraulic phenomena and determined the uncertainty parameters. By adopting distribution-free statistic techniques, the applicant performed 181 calculations with SRS for each experiment. The applicant then confirmed that the PCT of the experiment was covered by the third highest value of PCT calculated.

TR Appendix C.1 states that the following SETs for code assessment, code accuracy evaluation and the experimental coverage check are used:

- seventeen FLECHT-SEASET tests,
- one Cylindrical Core Test Facility (CCTF) test,
- seven PWR reflood in bundle geometry (NEPTUN) tests,
- four THTF tests, and
- one Primary Coolant Loop (PKL) test.

The four THTF tests assess the capability of the RELAP5/MOD3.3/K code to predict the blowdown progression. The other tests assess the capability of the code to predict the reflood progression and reflood PCT.

Appendix C provides extensive details regarding each SET facility, the tests that are chosen, and a discussion of the comparison of the results from the code assessment and accuracy against the experimental data.

The SRS calculations for the seventeen FLECHT-SEASET, CCTF, and NEPTUN tests used [[ ]] code uncertainty parameters that are all related to reflood core heat transfer. These parameters are:

- the [[ ]],
- the [[ ]],
- the [[ ]],
- the [[ ]],
- the [[ ]], and
- the [[ ]].

In addition to these parameters, the applicant also considered system parameters specific to the FLECHT-SEASET test facility, such as the [[ ]] in the SRS calculations. The applicant did not consider system parameters specific to the CCTF and NEPTUN facilities in the SRS calculations.

SRS calculations performed against four THTF tests used [[ ]] code uncertainty parameters related to the core heat transfer. These parameters are:

- the [[ ]],
- the [[ ]],
- the [[ ]],
- the [[ ]],
- the [[ ]], and
- two code uncertainty parameters related to [[ ]].

SRS calculations performed against the PKL-IIB5 test used [[ ]] code uncertainty parameters related to the core heat transfer and [[ ]].

In TR Appendix C, Sections 2.4, 4.4, and 5.4, the applicant determined comparisons of the predictions against the measured data to quantify the code accuracy for prediction of blowdown and reflood PCT. The applicant presented the accuracy in terms of a bias and standard deviation on the PCT.

TR Appendix C, Sections 2.5, 3.4, 4.5, 5.5, and 6.4, state that the SRS calculations support the conclusion that the measured PCTs are covered by the calculated values and that the number and the probability distribution functions of the selected code uncertainty parameters used for the SRS calculations in TR Appendix C are sufficient to cover the measured PCT.

3.10 Appendix D: Assessment of RELAP5/MOD3.3/K Against Integral-Effect Tests

TR Appendix D documents the capability of the RELAP5/MOD3.3/K code to predict thermal-hydraulic responses during blowdown and reflood phases when assessed against IETs. The purpose and approach for the assessment against the IETs is identical to that used for the assessments against the SETs in TR Appendix C and is not repeated here.

The IETs used for the code assessment are:

- four LOFT tests (L2-2, L2-3, L2-5 and LP-02-6),
- one Semiscale test (S-06-3), and
- one LOBI test (AI-66).

TR Appendix D, Sections 2, 3, and 4 provide extensive details regarding each IET facility, the tests that the applicant chose, and a discussion of the comparison of the results from the code assessment and accuracy against the experimental data.

TR Appendix D, Section 2 discusses the 181 SRS calculations performed by the applicant for each LOFT experiment assessment. The assessment includes all [[ ]] uncertainty parameters listed in TR Appendix D, Table 2-2. Since the fuel rods are used in the LOFT tests, the parameters include those related to both blowdown and reflood phenomena as well as the nuclear fuel-related parameters, such as [[ ]]. In addition, the assessment also included three parameters specific to LOFT tests [[ ]]. The applicant used test-specific uncertainty ranges of these facility-related parameters in the SRS calculations for each test because the applicable uncertainty ranges of the [[ ]] for each test.

SRS calculations performed against the Semiscale Test S-06-3 used [[ ]] uncertainty parameters listed in TR Appendix D, Table 3-3 that are related to [[ ]]. SRS calculations performed against the LOBI test used [[ ]] uncertainty parameters listed in TR Appendix D, Table 4-3 that are related to [[ ]].

TR Appendix D, Section 5 states that the comparisons of the predictions against the measured data from the selected IETs allow the determination of the code accuracy for prediction of blowdown and reflood PCT. The applicant determined the accuracy in terms of a bias and

standard deviation based on the difference in the predicted and measured PCT. The SRS calculations support the conclusion that the calculated PCTs are sufficient to cover the measured PCTs.

3.11 Appendix E: Assessment of RELAP5/MOD3.3/K Against APR1400 Reflood Tests

TR Appendix E describes the assessment of RELAP5/MOD3.3/K code against the APR1400 LBLOCA tests performed in the ATLAS (Advanced Thermal-hydraulic Test Loop for Accident Simulation) test facility. Three tests are used to simulate a LBLOCA in the APR1400 design:

- LB-CL-09 (Test 09),
- LB-CL-11 (Test 11), and
- LB-CL-15 (Test 15).

The applicant used these tests to assess major phenomena related to the DVI of the ECCW, such as boiling of the downcomer inventory and ECCW bypass.

According to TR Appendix E.2.1, the ATLAS facility is an integral effects thermal-hydraulic test facility that uses ½ reduced height and 1/288 volume scaled test facility based on the design features of the APR1400. It also provides details about the design and configuration of the ATLAS facility, along with important scaling relations used in the facility design. TR Appendix E.5 provides a description of the results.

TR Appendix E.3 describes one aspect of the assessment which indicates that the ATLAS tests showed [[ ]] during the APR1400 LBLOCA. The code simulations [[ ]].

In the assessment, the applicant performed 181 SRS calculations for ATLAS Test 09 and Test 15 using the selected uncertainty parameters. The applicant did not perform SRS calculations for Test 11 because all fuel rods are quenched at the very beginning stage of the time period considered for the code assessment. The applicant used a total of [[ ]] uncertainty parameters in the SRS calculations. The uncertainty parameters considered in the SRS calculations are [[ ]].

TR Appendix E.5 states that the RELAP5/MOD3.3/K code assessment against the ATLAS Test 09, 11, and 15 shows that the code correctly calculated the reflood thermal hydraulic behaviors, including the fuel rod cladding temperatures, and downcomer boiling phenomenon. Based on the results of these SRS calculations using the distribution free statistics approach, the applicant concluded that the selected uncertainty parameters, including the parameters related to [[ ]], are sufficient to cover the measured cladding temperatures.

3.12 Appendix F: Assessment of RELAP5/MOD3.3/K Against ECCW Bypass Tests

In TR Appendix F, the applicant addressed ECCW bypass during the late reflood period when only the SIP flow is available and examined the ability of the RELAP5/MOD3.3/K code to predict

this behavior. The assessment used the data obtained from tests conducted using the Multidimensional Investigation in Downcomer Annulus Simulation (MIDAS) and the Upper Plenum Test Facility (UPTF).

TR Appendix F.2 describes the MIDAS facility having a scale which is 1/5 in height, 1/24 in flow area, and 1/54 in flow rate as compared to APR1400. The temperature and pressure conditions of the MIDAS tests are prototypic to the APR1400 during the period of interest for the simulation. ECCW bypass tests allow the assessment of that phenomenon during the late reflood period. Only SIPs supply the ECCS injection during the late reflood period.

TR Appendix F.2.2.2 describes the assessment of the RELAP5/MOD3.3/K predictions of the ECCW bypass against MIDAS tests. This assessment shows that the code conservatively over-predicts the ECCW bypass fraction over the entire range. The degree of the over-prediction increases as the measured value decreases. RELAP5/MOD3.3/K also over-predicts the ECCW bypass fraction over the entire range of the steam flow rate. The over-prediction is stronger for the low steam injection flow rate condition. When the steam flow rate is high, the degree of over-prediction is reduced. The code-predicted condensation ratio is generally lower than measured, except for one test for which the code over-predicts the condensation ratio. TR Appendix F.2.2.2 states that even though RELAP5/MOD3.3/K typically over-predicts the ECCW bypass and under-predicts the condensation ratio, the code predicts the general trend of the bypass and condensation acceptably. It goes on to assert that the code prediction is conservative with respect to the availability of ECCS injection regardless of the number of available SIPs and their injection locations.

TR Appendix F.3 describes the assessment of RELAP5/MOD3.3/K predictions of ECCS bypass against UPTF Test 21-D where ECCW injection is through the two DVI nozzles in the downcomer. TR Appendix F.3.3 states that comparisons of key parameters of interest predicted by the code against the corresponding experimental data shows that RELAP5/MOD3.3/K reasonably predicts the major phenomena observed during the UPTF Test 21-D and conservatively over-predicts the ECCW bypass.

3.13 Appendix G: Assessment of RELAP5/MOD3.3/K Against Downcomer Boiling Tests

TR Appendix G describes the ability of RELAP5/MOD3.3/K to simulate downcomer boiling using assessments from the downcomer boiling (DOBO) test facility. TR Appendix G.2 describes the test facility which simulates the high temperature wall of the reactor pressure vessel annulus (downcomer) between the elevations of the cold leg and flow skirt. The test section has a heated rectangular cross-section with prototypic APR1400 dimensions. The circumferential length of the test section is 1/47.08 of the APR1400 under the assumption that there is no preference of downcomer boiling in the azimuthal direction. The simulated azimuthal angle of the test section corresponds to 7.65 degrees. Due to the small curvature of the simulated angle, the applicant neglected curvature, and used a rectangular channel for the test section. The downcomer wall is heated by the electrical heaters to achieve a steady-state constant heat flux condition. The test facility operates at prototypic APR1400 pressures during the period of interest. The injection nozzle of the emergency core cooling water is located at the upper part of the test section and the major parameters during the test were the wall heat flux, the water temperature, and the void fraction.

TR Appendix G.2.2 describes the results of the assessment of RELAP5/MOD3.3/K against experimental data from the DOBO facility. The code over-predicts the void fractions in all of the

axial nodes for all test cases. In addition, the measurements show subcooled water at the bottom of the test section, whereas the code calculations show almost no subcooling.

TR Appendix G. 3 concludes that RELAP5/MOD3.3/K conservatively predicts downcomer boiling using a single radial channel. Accordingly, in the TR, the applicant commits to using this model for APR1400.

3.14 Appendix H: Assessment of RELAP5/MOD3.3/K Against Fluidic-Device-Installed Safety Injection Tank Tests

TR Appendix H describes the assessment of the ability of RELAP5/MOD3.3/K to simulate the performance of safety injection tanks equipped with fluidic devices (SIT-FD). It states that the fluidic device is installed inside the SIT and passively controls the water injection flow rate from the SIT which injects water at a high flow rate during the blowdown and refill periods, and at a low flow rate during the reflood period.

TR Appendix H.1.1.1 describes the assessment facility, Valve Performance Evaluation Rig (VAPER) which consists of the SIT-FD, compressed air supply system, safety injection (SI) water discharge pipe line, SI water supply and recirculation system, stock tank, and a data acquisition and control system. The fluidic device is installed at the bottom of the tank. The fluidic device of the VAPER facility has a similar geometric shape and size as the in APR1400.

TR Appendix H.1.1.1 states that the major parameters measured in the tests are SIT water level, gas pressure and temperature, water level in the standpipe, and pressure drop across the fluidic device. The details of how the measurements are taken are detailed there.

The applicant uses four different modeling approaches to represent the SIT-FD. These models are:

- Model A: Modeling of SIT-FD with [[ ]]
- Model B: Modeling of SIT-FD with [[ ]]
- Model C: Modeling of SIT-FD with [[ ]]
- Model D: Modeling of SIT-FD with [[ ]]

Details regarding each of these modeling approaches are provided in TR Appendix H.1.2. The model assessment results show that models A, B, and C reasonably predict the injection flow rates, pressure drop in the tank, and water level behaviors during both the large and small flow injection periods. Air release behaviors predicted by these three models, however, show discrepancies from the test data. [[ ]].

TR Appendix H1.3 provides the assessment results. The assessment determines that the uncertainty range for the K-factors of large and small flow delivery can be determined using the results of VAPER tests and Model C is selected as the final SIT-FD model for the APR1400 LBLOCA analysis. The results also show that the K-factor selection shows little variation in downcomer/core level. The blowdown PCTs are the same in all calculations, because the blowdown PCTs occur before the SIT-FD injection. The maximum variation of reflood PCT and fuel rod quenching time, according to injection characteristics of the SIT-FD, is within 10 K and 75 s, respectively. The highest refllod PCT was predicted with [[

]].

3.15 Appendix I: Coupling RELAP5/MOD3.3/K and CONTEMPT4/MOD5

The RG 1.157 guidance states that the containment pressure should be used for evaluating cooling effectiveness during the post-blowdown phase of a LOCA. It goes on to state that the pressure should be calculated in a best-estimate manner and should include the effects of containment heat sinks and the operation of all pressure reducing equipment assumed to be available. However, TR Appendix I.1 states that the RELAP5/MOD3.3 code does not have the capability and thus, CAREM adopts the use of CONTEMPT4/MOD5 for this calculation. TR Appendix I.1.1 describes the coupling of CONTEMPT4/MOD5 to RELAP5/MOD3/K by utilizing the dynamic link library.

TR Appendix I.2 illustrates the use of this coupled code for calculation of a LBLOCA in the APR1400 and discusses the verification of the results. It states that the containment pressure calculated using the coupled code shows good agreement with the pressure calculated through iterative calculations exchanging the results of the separate calculations of RELAP5/MOD3.3 and CONTEMPT4/MOD5.

3.16 Appendix J: Sampled Uncertainty Parameter Values for 181 Plant Calculations

TR Appendix J provides the list of the sampled values for all the [[ ]] uncertainty parameters for each of the 181 plant calculations performed to determine the limiting PCT. The results of the calculations are described in TR Section 5.

3.17 Appendix K: Break and Burnup Sensitivity Study

TR Revision 1 provides Appendix K. This appendix provides the results of sensitivity studies for the break size, type, and location, in order to establish the limiting break for the nominal set of conditions. Since the effect of fuel TCD due to burnup affects the core stored energy, Appendix K also describes the results of sensitivity studies to determine the limiting time in life, i.e., the limiting burnup condition. Since the core stored energy can affect the limiting break size, the [[ ]] are performed for each of the three selected break discharge coefficients of 1.0, 0.8, and 0.6, for a total of [[ ]] cases. TR Appendix K also describes the process used to initialize the core stored energy that was specifically developed to address the TCD effect and that process is used to perform the 181 SRS calculations.

4. TECHNICAL EVALUATION

This review considers the compliance of the TR with the NRC regulatory requirements and applicable guidance documents for LBLOCA. It considers the technical soundness and merits of the statements made by the applicant in the TR and TR Revision 1.

Furthermore, this evaluation fully considers all of the material presented in responses to the AIs, information presented during the Audit Meetings, any supporting information addressing the AIs discussed during the Audit Meetings, as well as responses by KHNP to the RAIs. The NRC staff evaluated and satisfactorily resolved the issues raised in the AIs and RAIs, as described in this section.

During this review, the NRC staff identified certain Conditions and Limitations which must be met by the applicant for future licensing submittals that utilize the CAREM methodology. These Conditions and Limitations are discussed in this section and then are summarized in Section 5 of this SER.

4.1 Code Management

As discussed in SRP 15.0.2 Section 2, an EM consists of:

> *"one or more computer programs and other information necessary for application of the calculation framework to a specific transient or accident, such as mathematical models used, assumptions included in the programs, a procedure for treating the program input and output information, specification of those portions of the analysis not included in the computer programs, values of parameters, and other information necessary to specify the calculation procedure."*

The TR commits to using specific versions of RELAP and CONTEMPT that are dynamically coupled by the applicant. As discussed in TR Appendix B, the applicant obtained RELAP5/MOD3.3 through an international agreement and it has been rigorously maintained. During an inspection [Ref. 53], NRC staff verified that the codes are appropriately maintained under the KHNP QA program. In the EM, the applicant designated the modified code as RELAP5/MOD3.3/K.

KHNP's QA program is described in approved topical report APR1400-K-Q-TR-11005-NP-A, Revision 5, "KHNP Quality Assurance Program Description (QAPD) for the APR1400 Design Certification," April 2016, ADAMS Accession No. ML16123A406, [Ref. 54]. KHNP commits to follow ASME NQA-1-2008 and NQA-1a-2009 Addenda, "Quality Assurance Requirements for Nuclear Facility Applications," [Ref. 55], as endorsed by the NRC in RG 1.28, Revision 4. Subpart 2.7 of NQA-1, "Quality Assurance Requirements for Computer Software for Nuclear Facility Applications" provides requirements for the acquisition, development, operation, maintenance, and retirement of software. The KHNP QAPD governs the treatment of computer applications and digital equipment software when used in safety-related applications and designated non-safety-related applications. KHNP and its suppliers are responsible for developing, approving, and issuing procedures, as necessary, to control the use of such computer applications and digital equipment software.

In AI-26 [Ref. 3], the staff requested details of the quality assurance program applied to the LBLOCA Evaluation Model for the RELAP5/MOD3.3/K code, specifically for the changes made to the RELAP5 program. In response to AI-26, KHNP submitted their Design Control Procedure

DP-10-10, “Design Computer Code Control” [Reference 4]. The staff determined that the KHNP internal design control procedure is consistent with Subpart 2.7 of NQA-1.

Code Configuration Management

The guidance in SRP Section 15.0.2 states that the EM should be placed under a quality assurance program that meets the requirements of 10 CFR Part 50, Appendix B. The NRC staff conducted a separate limited scope software quality assurance inspection from February 29 through March 4, 2016 [Ref. 53], which included the RELAP5/MOD3.3/K code and other selected software. In the inspection, the NRC staff also examined the applicant’s implementation of a 10 CFR Part 21 [Ref. 15] program for reporting of defects and noncompliance. Since the applicant contracts design analysis activities to the various entities comprising KHNP, the inspection focused on the applicant’s oversight of these activities. The NRC inspection team found that, for the samples assessed, the applicant’s design controls are in compliance with the requirements of Criterion III, “Design Control” of Appendix B to 10 CFR Part 50. The oversight of contracted activities is consistent with the requirements of Criterion IV, “Procurement Document Control,” of Appendix B to 10 CFR Part 50.

On the basis of the applicant’s conformance with 10 CFR 50, Appendix B, the NRC staff finds that the applicant’s configuration management process for this EM is acceptable.

Code Changes

TR Appendix B documents the changes made to the source code of RELAP5/MOD3.3 to develop the RELAP5/MOD3.3/K code. Appendix B also includes excerpts showing the original source code and the modified version. The applicant provided the RELAP5/MOD3.3/K source code files [Ref. 26]. The NRC staff compared the changes to the source code described and shown in Appendix B against the actual source code provided in [Ref. 26].

In AI-75a [Ref. 3], the NRC staff requested the applicant to explain several code changes to the RELAP5/MOD3.3/K code and whether those changes were made under an acceptable system of controls that is consistent with Subpart 2.7 of ASME NQA-1. In response to AI-75a [Ref. 4] , the applicant explained that they implemented changes to allow improvements to the calculation of gas gap heat transfer and to edit the gap conductance data as well as the pellet radius as major edits to identify the current heat structure number. Similarly, the applicant modified the code to remove [[

]].

In AI-21 [Ref. 3], the NRC staff requested the applicant to characterize all of the changes made to RELAP5/MOD3.3 and include these in TR Appendix B. In response to AI-21 [Ref. 4], the applicant stated that TR Section 3.4 and Appendix B describe the model and error changes to RELAP5/MOD3.3/K which would affect the calculation results. The applicant stated that they transcribed and managed any other modifications to RELAP5/MOD3.3/K and CONTEMPT4/MOD5 in a separate report for each code controlled by the KHNP software management system. The NRC staff reviewed the documentation of code changes and the qualification records of persons making the changes during the software quality assurance inspection. The staff finds that the documentation is complete and in compliance with the requirements of 10 CFR 50, Appendix B, and that the personnel making the changes are qualified.

Based on the discussions above, including the response to AI-75, the staff considers the process used by the applicant in making modifications to the code and in maintaining code configuration acceptable, and therefore concludes that this issue is resolved.

<u>CONTEMPT-RELAP Coupling</u>

According to [Ref. 26] (page 7 of 23) for the APR1400 LBLOCA analysis,

> *"...TPEAK is set to 0 (zero) and CONTEMPT4/MOD5 dynamic-link library (contl.dll) is loaded when RELAP5/MOD3.3 starts reflood calculation (reflood heat structures enter to reflood mode). The coupled code sets TPEAK to the time reflood calculation starts."*

In AI-44 [Ref. 3], the NRC staff requested the applicant to explain how the containment backpressure is calculated and passed to RELAP5/MOD3.3/K for the accident period prior to the early reflood phase.

The applicant responded that [[

]] as described in Volume II, Section A10.1 [Ref. 20] of the code manual. [[
]]. The loading time is about 22 seconds after the break in the nominal calculation of the APR1400. Because the early reflood period begins at around 38 seconds, as discussed in TR Section 3.1.2.3, the applicant confirmed that CONTEMPT4/ MOD5 is loaded during the refill period. A representative containment pressure is used for LBLOCA analysis of the APR1400 before the start of the CONTEMPT4/MOD5 activation. The TR states that since the break flow remains critical to about 20 s, there should be no impact from containment pressure on break flow and thus reactor coolant system condition. Therefore, the NRC staff finds that the use of a "representative containment pressure" prior to activation of CONTEMPT4/MOD5 calculations is acceptable. Figure 2 in the response to AI-44 [Ref. 5] provides the value of the representative containment pressure used in the APR1400 analysis calculation and shows that it is lower than the Nominal Case containment pressure calculation.

The NRC staff finds that the applicant's treatment of CONTEMPT-RELAP5 coupling as described above is acceptable since it provides a conservative representation of the containment pressure response due to LBLOCA during the early reflood period, and therefore concludes that this issue is resolved.

<u>RELAP/MOD3.1 Applicability</u>

TR Section 4.2.3.2 cites results from a past assessment by the applicant using RELAP5/MOD3.1. That is not the code version that will be utilized in subsequent licensing submittals. In AI-48 Part a [Ref. 3], the NRC staff requested the applicant to indicate whether RELAP5/MOD3.3/K, the version that will be used in subsequent license submittals, has been used to confirm that the results cited are applicable. In response [Ref. 4], the applicant stated that the cited analysis has not been confirmed directly using RELAP5/MOD3.3/K; however, it is expected to be acceptable because of similar code predictability inferred from the code accuracy results, which were described in [Ref. 43] and [Ref. 44], where the RELAP5/MOD3.3/K predictions show similar or more conservative results. The applicant further stated that in order

to confirm the code applicability of RELAP5/MOD3.3/K, they performed an assessment against the UPTF-10B test. The results of the recent simulation are consistent with the results from RELAP5/MOD3.1, as shown in Figure 1 of the response.

The applicant stated that the RELAP5/MOD3.3/K has not been tested against data, except for UPTF-10B. They further stated that results using the RELAP5/MOD3.3/K code version have been shown to be "similar" or "more conservative" results.

The NRC staff finds that the applicant's assessment of the applicability of RELAP5/MOD3.1 to APR1400 is acceptable in this instance, since it provides similar or more conservative results as compared to experimental data, and therefore concludes that this issue is resolved.

Code Accuracy

The applicant defined code accuracy as the statistical difference between calculated and measured PCTs. The applicant used standard definitions of parameters, such as the mean and standard deviation, to determine the statistical code accuracy based on the difference in the code-predicted and the measured PCT. In AI-49 [Ref. 3], the NRC staff requested the applicant to clarify the one-sided 95 percent limit of blowdown PCT of [[ ]]. The applicant's response to AI-49 [Ref. 4] stated that the equation in the TR (i.e., [[ ]]) is correct as stated in the TR. However, the applicant corrected the numerical value to [[ ]] in TR Revision 1.

The NRC staff finds that the applicant's assessment of code accuracy is adequately justified in the response to AI-49, and on the basis of that information concludes that this issue is resolved.

4.2 Code Scaling, Applicability, and Uncertainty

TR Section 1 states that the CAREM methodology follows the Code Scaling Applicability and Uncertainty (CSAU) methodology [Ref. 19]. In the NRC staff assessment of the TR, the NRC staff raised a number of issues, which are further discussed in this section of the SER.

PIRT and Uncertainties

TR Section 3.3 states that the distribution-free statistical approach that is used in CAREM is superior to that of CSAU (NUREG/CR-5249). The TR states that, in contrast to the CSAU methodology, CAREM provides more freedom in the ranking of the phenomena in a PIRT, as evidenced by a large number (i.e., [[ ]]) of phenomena or processes that are ranked as [[ ]]. However, the application of the methodology, described in TR Section 5 states that only [[ ]] uncertainty parameters are utilized. In addition, the process used to arrive at the [[ ]]uncertainty parameters is not described. Therefore, in AI-14, Parts a and b [Ref. 3], the NRC staff requested the applicant to clearly describe the process that is followed to determine the number of uncertainty parameters that can be used. Further, the NRC staff also requested the applicant to explain and justify the approach used (i.e., conservative, or best-estimate values) for the remaining parameters not included as part of the uncertainty analysis.

The applicant responded [Ref. 5] that they described the entire process to determine the [[ ]] uncertainty parameters from the [[ ]] process/phenomena in the APR1400 PIRT in the presentation material provided during the January 2016 meeting [Ref. 1.a] and that they provided a revised PIRT in response to AI-15 [Ref. 3]. In the response to AI-14, Part a, Tables 1 through Table 3 describe the PIRT component, the related process/phenomena, and provides

a reference to a section in the TR or other sources of information. This information provides an additional roadmap to the location of the discussion of the process/phenomena. The response then describes the general process used to determine whether a parameter is treated through an uncertainty evaluation or is treated conservatively.

In response to Part b of AI-14, the applicant provided Table 1, Table 2, and Table 3 to justify excluding some parameters from the uncertainty analysis. In addition, the applicant noted that two phenomena were not included in the TR:

- [[ ]],
- [[ ]].

The applicant stated that they did not consider [[ ]]. The applicant's statements at the June 2016 Audit Meeting argued that it is highly unlikely that [[ ]]. Based on the results of the VAPER tests, consideration was given to various SIT parameters already included for uncertainty quantification, and the expanded geometry of the SI piping connection to the reactor coolant system (i.e., minimizing the potential for any impact from the pump flow on SIT injection). Therefore, the staff concludes that the applicant has provided sufficient justification and bases for the conservative treatment of both [[ ]], such that further uncertainty considerations are not required. The staff considers this issue resolved.

In response to RAI 7-8567, Question No. APR1400-4, (ADAMS Accession No. ML16153A454), the applicant provided the same information also provided in response to AI-14, Part a [Ref. 5], but with a clear description of the process used to determine the phenomena that were treated by other uncertainty parameters, the parameters treated conservatively, and the phenomena treated as biases. The response to RAI 7-8567 discussed the phenomena treated conservatively in the response to AI-5 [Ref. 5], and was augmented by an additional discussion of phenomena treated by other uncertainty parameters and phenomena treated as biases. Table 1 in the response to RAI 7-8567 [Ref. 1.h] describes the PIRT component, the related process/phenomena, and provides a reference to a section in the TR or other sources of information. Table 2 in the response to RAI 7-8567 provides a list of the phenomena treated conservatively by CAREM. Table 3 provides a list of the modified APR1400 PIRT process/phenomena treated as biases. While not a clear and concise description of the process used to determine when a parameter is treated through uncertainty evaluation or is treated conservatively, the information in the response to RAI 7-8567 provides sufficient explanation to understand how each PIRT process/phenomena is treated.

The NRC staff finds that the applicant's RAI and AI responses and modification of the PIRT are acceptable and comply with the guidance as outlined in RG 1.203, and on the basis of that information concludes that this issue is resolved.

PIRT Source Information

TR Table 3-2 provides the PIRT used by the applicant as part of the CAREM methodology. The PIRT is used to identify the major phenomena or processes that occur during the accident and to prioritize the major phenomena according to their impact on the acceptance criteria. The PIRT is based on the KNGR PIRT [Ref. 52]. The applicant adjusted the relative importance of certain phenomena or processes in the KNGR PIRT after modifying the definition of temporal periods between the LBLOCA used for the KNGR PIRT and that for the APR1400 PIRT.

TR Appendix A provides details of the KNGR PIRT, but only shows the phenomena that have a numerical ranking of [[ ]] during any phase of the accident, i.e., only the phenomena determined to have a significant influence on the transient. A complete PIRT, as described in the guidance of NUREG/CR-5249 [Ref. 19] includes all the phenomena and their corresponding ranks. In addition, both the KNGR and the APR1400 PIRTs lack a ranking for the "state-of-knowledge" that is used to determine whether a particular phenomenon is well understood and can, therefore, be simulated accurately by the EM. In addition, according to RG 1.203 [Ref. 18], the PIRT generated to guide the EM development process should be adequately documented. Such documentation includes the discussion of the steps followed to generate the PIRT and the definition of each phenomenon that is ranked in the PIRT. Neither the original TR Section 3.3, nor its Appendix A provides a complete PIRT. In AI-15 [Ref. 3], the NRC staff requested the applicant to provide a complete PIRT that addresses the above-mentioned issues.

In response [Ref. 5], the applicant stated that it would revise the APR1400 PIRT to the document contained in Attachment 2 of the Revision 1 response to AI-15 [Ref. 6], which includes a brief rationale description for the assigned process/phenomena rankings and a brief state-of-knowledge column. In addition, TR Revision 1, Appendix A, documents that six experts in nuclear reactor research and operation peer reviewed the adjustments to the KNGR PIRT to arrive at the APR1400 PIRT.

The NRC staff finds that the information supplied satisfies the regulatory guidance in RG 1.203 [Ref. 18] and the approach originally documented in NUREG/CR-5249 [Ref. 19] for PIRT development, and is therefore acceptable. Since the additional information is included in the TR revision [Reference 11], the staff concludes that this issue is resolved.

Containment Pressure Analysis

Section 3.12.1 of RG 1.157 [Ref. 17] states that "...the containment pressure used for evaluating cooling effectiveness during the post-blowdown phase of a LOCA should be calculated in a best-estimate manner and should include the effects of containment heat sinks..." According to TR Section 3.4, the APR1400 best-estimate LBLOCA methodology calls for the coupling of RELAP5/MOD3.3/K with CONTEMPT4/MOD5. Consequently, containment phenomena need to be included in the PIRT, so as to determine the adequacy of, and uncertainty in, CONTEMPT4/MOD5. Table 3-2 contains and ranks generic processes such as "pressure history" and "temperature history" for the containment component. However, the NRC staff considers the ranking of such generic processes insufficient for adequacy determination. The NRC staff position is that individual phenomena that impact these general processes (e.g., "condensation heat transfer," "impact of non-condensable gases," "droplet heat and mass

transfer," etc.) should be included and ranked in the PIRT to determine code adequacy and uncertainty following the SRP Section 15.0.2, and NUREG/CR-5249. Therefore, in AI-16 [Ref. 3], the NRC staff requested the applicant to provide information regarding the PIRT, containment parameters and the process used for the statistical treatment of the containment calculations.

In response [Ref. 4], the applicant stated that CAREM does not treat the containment pressure analysis statistically through uncertainty variables and assessments, but performs a conservative analysis using conservative assumptions. The applicant provided the initial conditions, active heat removal systems assumptions, and the passive heat removal assumptions used to perform a minimum containment pressure analysis in response to AI-16 [Ref. 4]. A conservatively low containment pressure calculation based upon the initial conditions and heat removal assumptions will result in a conservative transient analysis in terms of the PCT.

The NRC staff finds that the explanation supplied by the applicant is acceptable, since an assumed low containment pressure will result in a conservative PCT prediction, so the pressure uncertainty need not be considered. The staff concludes that this issue is resolved.

Importance Adjustments

In Parts (a) through (i) of AI-17 [Ref. 3], the NRC staff requested the applicant to justify the rationale for the assignment of the importance ranking for several phenomena in the APR1400 PIRT presented in Table 3-2. The NRC staff requested the applicant to provide the basis for the stated importance rankings for the following phenomena:

- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]],
- [[ ]], and
- [[ ]].

In response to AI-17 [Ref. 5], the applicant stated that modification of the KNGR PIRT to the APR1400 PIRT resulted in some errors, in large part due to the change in the definitions of various time periods. The applicant revised the PIRT in Revision 1 of TR Appendix A to correct the errors consistent with the definition of the time periods and provided a basis for the importance rankings.

The NRC staff finds that the applicant acceptably updated revisions to the importance rankings consistent with the guidance in RG 1.203 [Ref. 18] and the approach originally documented in NUREG/CR-5249 [Ref. 19]. Therefore, on the basis of that information concludes that this issue is resolved.

Applicable Phenomena

TR Table 3-5 lists the constitutive equations and models in the RELAP5/MOD3.3/K and CONTEMPT4/MOD5 computer codes that are used to predict important phenomena or processes for the APR1400 LBLOCA. Table 3-5 identifies the components, the key phenomena, or processes in each component based on the PIRT rankings and the models or correlations in the codes that participate in the prediction of each phenomenon.

TR Table 3-5 does not provide the entire picture of the presence of models and correlations in RELAP5/MOD3.3/K and CONTEMPT4/MOD5 that are necessary for phenomena prediction. The model for the phenomenon of “entrainment/de-entrainment” and “level” in the upper plenum component is specified as “interfacial drag” in TR Table 3-5. However, as pointed out in TR Section 3.6.1, RELAP5/MOD3.3/K cannot model de-entrainment on internal structures. As a result, the information in TR Revision 0, Table 3-5, can be confusing because the reader is left with the impression that the interfacial drag correlations in RELAP5/MOD3.3/K can predict de-entrainment, whereas in reality such a correlation does not exist in the code. The applicant did not mention any other biases that will be included in the uncertainty analysis in TR Section 3.6.1 or Table 3-5.

In AI-19 Part a [Ref. 3], the NRC staff requested the applicant to provide a clear statement of all the phenomena that are treated by biases in TR Section 3.6.1. In the Revision 1 response to AI-19, Part a [Ref. 4], the applicant stated that there are two types of biases in CAREM; (1) Steam Binding bias, described in TR Section 4.2.3.2, which consisted of de-entrainment bias in the upper plenum and a droplet evaporation bias in the steam generator U-tubes; and (2) ECC Bypass bias, described in TR Section 4.2.3.1, which the applicant divided into a ECC bypass bias during the refill and ECC bypass bias during reflood. The applicant provided Table 1, “Modified APR1400 PIRT treated by biases,” in the response to AI-19, Part a, which summarized the processes / phenomena that are treated as biases. The applicant’s discussion of the steam binding and the ECCS bypass bias along with the table that lists the PIRT phenomena that are treated by biases addresses the NRC staff’s question in AI-19, Part a.

The response to RAI 7-8567, Question APR1400-4, (ADAMS Accession No. ML16153A454), includes for component “Reactor vessel upper plenum” a process/phenomenon and level as part of the Steam binding bias. Table 1 in the response to AI-19, Part a does not include this process/phenomenon. In addition, both tables include the process/phenomena flashing in the ECC bypass bias, which the applicant agreed in the June 2016 Audit Meeting should not be included in a bias discussion.

In Part b of AI-19 [Ref. 3], the NRC staff requested the applicant to consider modifying TR Table 3-5 to include a column providing the status of existence of a model (“yes/no”) in

RELAP5/MOD3.3/K and CONTEMPT4/MOD5 for each of the listed phenomenon, following the approach utilized in NUREG/CR-5249.

In response [Ref. 4], the applicant stated that the second column of TR Table 3-5 shows important phenomena related to each component and that the third column shows the existing models in RELAP5/MOD3.3/K and CONTEMPT4/MOD5. Table 1 in the response to AI-19, Part a, provided information from the Modified APR1400 PIRT that showed which process/phenomena were treated as biases.

The NRC staff finds that the information supplied in response to AI-19 is acceptable, since it completes the expected PIRT information recommended in RG 1.203 [Ref. 18] consistent with the approach documented in NUREG/CR-5249 [Ref. 19]. The applicant incorporated this information in TR Revision 1, so the NRC staff concludes that this issue is resolved.

<u>Uncertainty Parameters</u>

In TR Section 4.3.1.2, the applicant demonstrates the variety, number, and ranges of the selected code uncertainty parameters that are appropriate for each licensing application of the CAREM methodology. In TR, Revision 1 [Ref. 13], Section 5.2.2, “Plant SRS Calculation,” the applicant confirms that in license submittals they will utilize distribution-free statistics and 181 simulations in order to obtain at least three samples for the PCT, one for the MCO and one for the CWO for a 95/95 confidence level. The applicant will then choose appropriate uncertainty parameters for variation during each of the 181 calculations for each SET used in what is called the “coverage check” as described in TR Section 4.3.2.

According to the applicant, the ‘coverage check’ compares the uncertainty of [[ ]] parameters against SET data. In AI-50, Part a [Ref. 3], the NRC staff requested the applicant to explain the process for choosing the limited number of parameters for uncertainty variation in the blowdown and reflood “coverage check” from the list of all phenomena with [[ ]] during these periods. Also, in AI-50, the NRC staff asked the applicant to explain how it is ensured that inclusion of the additional uncertainty for parameters does not change the results and alter the conclusions of the ‘coverage check’. Further, if such a demonstration has been accomplished, the NRC staff requested the applicant to describe the change in the limiting PCT in AI-50, Part b [Ref. 3].

In the response [Ref. 4], the applicant stated that all uncertainty parameters can be consistently considered for blowdown and reflood coverage check. The applicant further stated that it is not necessary to consider the uncertainty parameters related to reflood phenomena for blowdown coverage check, and vice versa.

The NRC finds that some uncertainty parameters involved with both blowdown and reflood phenomena are commonly used for the blowdown and reflood coverage check. In addition, both uncertainty parameters related to blowdown and reflood phenomena are considered for IETs. Furthermore, a larger number of uncertainty parameters results in wider uncertainty variation (i.e., higher cladding temperature).

However, in the view of the NRC staff, this explanation did not address the fact that many phenomena [[ ]], specifically for blowdown (APR1400 PIRT), are not used in the blowdown coverage check for the SET. This was discussed during the January 2016 Audit Meeting [Ref. 12], where the applicant confirmed that the number of parameters selected covers the applicable and appropriate phenomena [[ ]]. The applicant stated that the

phenomena that were not included in the coverage check for a specific SET were not applicable or appropriate for consideration based upon the APR1400 design and operating conditions, which precluded particular phenomena from occurring. Based upon the facility design detailed information in the TR, the NRC staff concurs that restricting the selection of parameters to those related to the blowdown phenomena in the specific SET coverage checks is appropriate. The NRC staff finds that the coverage check performed by the applicant is acceptable, and is consistent with the guidance in RG 1.203 [Ref. 18] and the approach documented in NUREG/CR-5249 [Ref. 19]; therefore, the determination of uncertainty parameters is also acceptable. On the basis of this information, the NRC staff concludes that this issue is resolved.

Simple Random Sampling Approach

TR Section 5.2.2 discusses the use of simple random sampling approach to perform 181 SRS calculations using the distributions of the uncertainty parameters listed in the TR Table 5-1. However, the TR does not discuss the model used to perform the random sampling. In AI-63 [Ref. 3], the NRC staff requested the applicant to describe the model or method that is utilized to perform the random selection of samples and to provide justification that the sampling process is unbiased.

The applicant's response [Ref. 4] referred to a report by Glaser, "BEMUSE Phase VI Report," [Ref. 40] that provides guidance which has been followed for CAREM:

> *"It is also important to note that the model outcome sample values $Y_1 \ldots Y_N$ from which the tolerance intervals/limits are determined must constitute a random sample of the model outcome Y in the statistical sense, i.e., they must be realizations of stochastically independent and identically distributed random variables $Y_1 \ldots Y_N$ from which the tolerance intervals/limits are determined must constitute a random sample of the model outcome Y in the statistical sense, i.e., they must be realizations of stochastically independent and identically distributed random variables $Y_1 \ldots Y_N$.. This is ensured if the underlying input parameter sample is generated according to the simple random sampling principle. Other types of parameter selection procedures like Latin Hypercube Sampling or Importance Sampling, etc. may therefore be appropriate for tolerance intervals or tolerance limits."*

The Glaser report also states:

> *"Simple random sampling should be used to derive tolerance intervals or one-sided tolerance limits."*

To get the random sample, the applicant used an open source computer program, LHS [Ref. 45]. Simple random sampling was performed using the option for "random sampling" in the LHS code that causes LHS to perform a pure Monte Carlo method to generate random samples.

The applicant demonstrated the use of distribution-free statistics for the APR1400 LBLOCA calculations in TR Section 5.2.2. The applicant chose the third highest PCT result of 181 calculations as the 95/95 value for PCT. In TR Section 5.2.2, the applicant presented the third highest result of 181 SRS calculations as the limiting PCT. In AI-64 [Ref. 3], the NRC staff requested the applicant to provide the values and plots for all three highest PCTs predicted in the SRS calculations. The applicant provided the requested information [Ref. 4], but the AI-64 response information was superseded by new calculations in TR Revision 1.

The NRC staff finds that the applicant's method for performing the random sampling is acceptable, as demonstrated by the revised calculations which utilize methods published in the peer-reviewed literature that meet Regulatory Position 4 (Estimation of Overall Calculational Uncertainty) of RG 1.157 [Ref. 17]; therefore, on the basis of that information, the staff concludes that this issue is resolved.

Basis for the Number of Cases

The applicant employs distribution-free statistics and uses 181 simulations to obtain at least five samples consisting of the highest maximum cladding oxidation (MCO), the highest core wide oxidation (CWO), and the third ordered peak cladding temperature (PCT) at a 95/95 confidence level. The applicant formally documented this approach in the response to AI-51, Parts a-d [Ref. 5]. In the response, the applicant states that the uncertainty evaluation method is third order with 181 runs based upon the theoretical background provided by the formulation of Guba, Makai, and Pál [Ref. 37], and the AI response details the parameter selection process and the number of cases that need to be run. The methodology continues to rely upon the work of Glaser [Ref. 40] to agree that more than 124 sample runs are necessary to meet the NRC's requirements on the interpretation for satisfying the 10 CFR 50.46 acceptance criteria under the order statistics approach. The response provides justification for the use of 181 cases and the selection of the third highest value of PCT, the highest value of the MCO, and the highest CWO selected from all of the SRS calculations. With this approach, the APR1400 CAREM methodology documented in TR Revision 1 remains largely the same as that in the original TR Revision 0, the only difference being that the number of SRS calculations would now be 181 instead of the previous 124.

The NRC staff finds that the applicant's use of distribution free statistics to determine that 181 cases satisfies Regulatory Position 4 (Estimation of Overall Calculational Uncertainty) of RG 1.157, since it is consistent with published peer reviewed statistical methods; therefore, on the basis of that information, the staff concludes that this issue is resolved.

Uncertainties in Initial and Boundary Conditions

As stated in TR Section 5, uncertainties in the initial and boundary conditions of the plant as well as those associated with modeling parameters govern the uncertainties associated with analysis of LBLOCA events. Plant initial and boundary conditions include [[

]]. TR Table 5-1 lists probability distribution functions and uncertainties of plant condition parameters, as well as code uncertainty parameters that are applicable.

The guidance in RG 1.157, Section 4, establishes acceptable controls for the estimation of uncertainties. In addition, NUREG/CR-5249 describes the process for formulation of uncertainty distributions. In CAREM, the uncertainties are propagated through the models using distribution-free (i.e., non-parametric) statistics to determine the limiting PCT based on the variation of key uncertainty parameters.

In AI-52 Part a [Ref. 3], the NRC staff requested the applicant to provide the justification for the use of the non-parametric distribution free statistical approach. The Revision 1 response to AI-51 [Ref. 6] provides the justification.

The NRC staff finds that the applicant's response to AI-51 provides an acceptable explanation to support the uncertainties employed to ascertain PCT, consistent with the guidance in Regulatory Position 4 of RG 1.157, and on the basis of that information concludes that this issue is resolved.

95/95 Methodology

The example of the CSAU process shown in NUREG/CR-5249 indicates that the mean value of the effect for each of the uncertainties on the blowdown and reflood portions is calculated. The sums of all the 95th percentiles, relative to the mean, are added to obtain the summed biases. The summed biases are then added to 95th percentile of the response surface produced from the calculation of the mean peak cladding temperatures from the uncertainty of high influence parameters. TR Revision 1, Tables 5-5 through 5-8, provide the determination of the biases for application in the CAREM process. In AI-52, Part b [Ref. 3], the NRC staff requested the applicant to describe how the CAREM process provides the 95th percentile for the bias determination.

In response [Ref. 4], the applicant stated that, although the bias evaluation in NUREG/CR-5249 appears to provide the 95th percentiles for the bias determination, it only considers the bias related to the hot channel. It further states that CAREM uses almost the same approach as NUREG/CR-5249. The applicant evaluated the biases in CAREM using [[ ]]. The applicant then observed the thermal-hydraulic conditions that can affect the PCT in the assessment of [[ ]]. The applicant considered the evaluated bias as shown in Tables 5-4 through 5-7 for practically all of the SRS cases. For example, [[ ]]. The 95th percentile of the bias does not need to be evaluated in CAREM because the bias is considered for practically all of the SRS cases. Therefore, for the final scaled bias evaluation, the applicant considers the maximum biases, which neglect the negative biases.

This response shows that the determination of biases in CAREM differs from the approach utilized in CSAU (NUREG/CR-5249), but the applicant argues that in the CAREM approach, the bias is considered conservatively, for a large number of SRS cases. Since the maximum impact of biases is in increasing the calculated 95th percentile of PCT, the applicant states that this approach is conservative.

RG 1.157 indicates that the uncertainty can vary over time during the transient, and it states that:

> *"In evaluating the code uncertainty, it will be necessary to evaluate the code's predictive ability over several time intervals, since different processes and phenomena occur at different intervals. For example, in large-break loss-of-coolant accident evaluations, separate code uncertainties may be required for the peak cladding temperature during the blowdown and post-blowdown periods. Justification for treating these uncertainties individually or methods for combining them should be provided."*

Since it was not clear to the NRC staff how the temporal differences in the bias and uncertainty are treated in the CAREM methodology, in AI-52, Part c [Ref. 3], the NRC staff requested the applicant to provide the basis and the process for determination and application of the bias and uncertainty during the blowdown and reflood (early and late) periods.

In response [Ref. 4], the applicant stated that the temporal difference in the biases is addressed by examining the importance rankings defined in the PIRT. If one bias is important for a certain period, the bias is activated for that period. The applicant concluded that the treatment of biases need not consider the uncertainties due to the application of the estimated biases. The estimated biases from test assessment are applied for practically all of the SRS cases, thus the methodology obtains the most conservative estimation of biases as the result of the final bias evaluation. The applicant restates that the assignment of biases in CAREM follows a conservative approach.

Furthermore, the response states that the temporal impact of biases considers the importance rankings in the PIRT. The applicant provided examples for ECC bypass, steam binding and steam generator U-tube heat transfer and vapor superheat to show the consideration of biases varies over intervals of time. The applicant provided these to show that the final bias evaluation is sufficiently conservative.

The NRC staff finds that the applicant's methodology for determining suitably conservative biases is acceptable, since it is consistent with Step 20 of RG 1.203 [Ref. 18] and the approach documented in NUREG/CR-5249 [Ref. 19]. Therefore, on the basis of that information, the staff concludes that this issue is resolved.

Uncertainty Distribution SRS for FLECHT-SEASET Tests

Table 2-2 in TR Appendix C shows the parameters and distribution functions used for SRS calculations for the FLECHT-SEASET tests. The system-specific parameters for the FLECHT-SEASET test facility include [[ ]]. However, the applicant did not provide an explanation for how the uncertainty in these parameters is determined. Therefore, in AI-77 [Ref. 3], the NRC staff requested information on the process followed to determine the uncertainty range and distribution for the flooding rate and power that are applicable to the FLECHT-SEASET facility. AI-77 also requested clarification regarding the importance of negative power and flooding rate and how they are handled in the calculations.

In response [Ref. 4], the applicant provided the measurement uncertainties of total power and volumetric flow rate at a one sigma level. Since the applicant's RELAP5 model uses the liquid phase velocity as a boundary condition using a time dependent junction, volumetric flow rate is changed to obtain velocity using the core area of 0.0156 m^2. Furthermore, the response stated that, in CAREM, the uncertainty ranges are calculated by [[ ]]. Therefore, the sample uncertainty ranges for power and liquid phase velocity (reflooding rate) are as follows:

[[ ]]

These uncertainties for power and reflooding rate are used in conjunction with normal distribution functions for power and reflooding rate in common, since the measurement uncertainty data described in [Ref. 2-1] of TR Appendix C are the standard deviations for each measurement.

In the response, the applicant stated that the fuel rod power in TR Appendix C, Table 2-1, is the linear heat generation rate for a fuel rod, while the uncertainty associated with power shown above is the uncertainty associated with the total power. The response also stated that the unit

of flooding rates in Table 2-1 in Appendix C was given as cm/sec, and it should have been in mm/sec. In TR Revision 1, Appendix C, the correct information is listed.

The sampled values for power and reflooding rate are varied from their nominal values, which are boundary conditions of each test. Therefore, the sampled values are added to the nominal values of the total power and reflood rate.

The NRC staff finds that the applicant has provided sufficient information on the process to determine the uncertainty range and distribution for flooding rate and power, and on the basis of that information concludes that this issue is resolved.

Limits of the Uncertainty Distribution

The NRC staff determined that the limits of the uncertainty distribution utilized in CAREM (e.g., twice the standard deviation) used for the parameters with normal distribution during the SRS sampling in TR Appendix C, are not clearly specified. In AI-78 [Ref. 3], the NRC staff requested the applicant to clarify the uncertainty limits for parameters with a normal distribution during SRS sampling

In response [Ref. 4], the applicant stated that, for the SRS sampling of the uncertainty variables with normal distribution, the uncertainty range of [[ ]] is used in order to include a broad range in the uncertainty. On the one hand, for sampling with uniform distribution, the uncertainty range of [[ ]] is used. In this case, σ is the standard deviation, but μ is the deviation of one-sided limit.

The NRC staff finds that the applicant has clarified the uncertainty limits acceptably, and on the basis of that information concludes that this issue is resolved.

Negligible Uncertainties

In TR Section 4.5 of Appendix C, the applicant states that "...the selected code parameters and distribution functions for the NEPTUN test are the same as those used in the SRS calculations against FLECHT-SEASET tests." However, Table 4-5 of Appendix C lists only [[ ]] parameters compared with the [[ ]] parameters used for the FLECHT-SEASET tests. In AI-79 [Ref. 3], the NRC staff requested the applicant to clarify and confirm any inconsistencies.

In response [Ref. 4], the applicant referred to the response to AI-50, Part a, which commits to considering the effects of test-specific uncertainty, if not negligible. The response provided specific examples of these uncertainty factors in the FLECHT-SEASET tests, such as the [[ ]].

The NRC staff finds that the applicant's information concerning non-negligible uncertainties is acceptable, and on the basis of that information concludes that this issue is resolved.

4.3 Evaluation Model Assessments

According to TR Section 4.1, the applicant performed the EM assessment using SETs that investigate individual phenomena, and IETs that incorporate many or all of the important phenomena and components. Table 4-1 restates the type of assessments, facilities used, and where the test details are documented.

Table 4-1 APR1400 Separate Effects Tests

Assessment	Facility Tests	Documented in
reflood, system pressure, clad temperature, power refill & reflood reflood energy transport/transfer reflood	FLECHT-SEASET 31805, 31504, 31203, 31302, 31701, 34209, 32013, 30518, 30817, 34420, 31021, 34524, 31108, 32235, 32333, 33338, 34006 CCTF C2-4 NEPTUN 5025, 5036, 5049, 5050, 5051, 5052, 5056 THTF 105, 151, 160, 162 PKL IIB5	TR Appendix C
integral effects tests	LOFT L2-2, L2-3, L2-5, LP-02-6 Semiscale S-06-3 LOBI A1-66	TR Appendix D
DVI of ECCW, downcomer boiling, ECCW bypass	ATLAS 09, 11, 15	TR Appendix E
ECCW bypass	MIDAS, UPTF 21-D	TR Appendix F
downcomer boiling	DOBO 1, 2, 3, 4	TR Appendix G
SIT-FD	VAPER 01, 02, 03, 04	TR Appendix H

Certain tests raised issues during the review and were the subject of information requests to the applicant.

FLECHT-SEASET Reflood Tests

TR Section 3.4 and Appendix B discuss two modifications which the applicant made to improve the predictability of the time of fuel rod quenching. Figures 3 and 4 in TR Appendix B show only a small difference from the change in the PCT for [[ ]]. In AI-22 [Ref. 3], the NRC staff requested the applicant to discuss the impact of the code modification that was used to improve the fuel rod quenching time predictions. In response, the applicant provided figures illustrating the effect of [[ ]] on the simulation of FLECHT-SEASET low flooding rate tests 31805 and 34006, which showed delayed quenching and better agreement with the test data.

Additionally, as part of AI-23, the applicant clarified that RELAP5/MOD3.3/K predicts [[ ]] measured in FLECHT-SEASET tests 31805 and 34006. However, in a letter dated April 25, 2018, the applicant provided analysis results to demonstrate that the application of RELAP5/MOD3.3/K in accordance with the CAREM methodology results in conservatively high estimates for fuel temperatures and quench times for FLECHT-SEASET tests at all reflood rates (Reference 57). Based on the analyses presented in Appendix C of the TR and the supplemental information provided in Reference 4 and Reference 57, NRC staff finds that the application of RELAP5/MOD3.3/K in accordance with the CAREM methodology provides suitably conservative estimates for fuel temperature during reflood.

An initial version of this SER imposed Condition (1) here. This condition has been removed based on the additional information presented by the applicant. However, due to external references, the condition numbering is being kept for consistency.

Condition (1): Deleted.

ATLAS Integral Effects Tests

In TR Appendix E, the applicant shows that [[ ]] than the ideal case using scaling relationships. This means that [[ ]]. Therefore, the outer surface temperature of the lower downcomer wall (i.e., reactor vessel-wall below the cold leg elevation) of [[ ]]. In spite of this, [[ ]]. As a result, [[ ]] in the APR1400. The initial temperature of the upper downcomer wall was [[ ]].

The applicant states that the results of three ATLAS tests showed [[ ]] during the APR1400 LBLOCA events. Moreover, RELAP5/MOD3.3/K code is [[ ]].

The scaling of the ATLAS facility is briefly discussed in TR Section 2.1 of Appendix E. It provides the major scaling relations used in the ATLAS facility design. Section 3.1 of Appendix E demonstrates that the downcomer wall heat transfer area of the ATLAS is about [[ ]]. However, the applicant did not discuss the approach used to scale the metal mass to fluid volume ratio in order to correctly capture the stored energy release and downcomer boiling phenomena. In AI-82 [Ref. 3], the NRC staff requested the applicant to describe how the metal mass to fluid volume ratio was scaled in order to correctly capture the stored energy release and the downcomer boiling phenomena.

In the response to AI-82 [Ref. 4], the applicant provided descriptions of how the metal mass to fluid volume ratio was scaled in order to correctly capture the stored energy release and the downcomer boiling phenomena. The response stated that for the reactor pressure vessel downcomer with an annular shape, the metal mass to fluid volume ratio is proportional to volume ratio, and can be expressed by:

[[ ]]

where D_i and D_o are the downcomer inner and outer diameter, respectively. L is the height, and t is the vessel thickness. The ratio of metal mass to fluid volume can be expressed by:

[[ ]]

Here, G is the downcomer gap, defined as (D_o-D_i), and the subscript R indicates a ratio of model to prototype.

In order to preserve the stored energy release from the downcomer wall to fluid in the scaled-down model, the ratio $(t/G)_R$, should be equal to 1.0 However, in the as-built design of ATLAS, the downcomer diameter was intentionally enlarged to have a gap large enough to simulate the cap bubble rising in the downcomer region and the wall thickness was increased to be able to withstand the high pressure conditions using the guidance from the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME) (B&PV) Code, Section VIII, Division II. Consequently, the ratio $(t/G)_R$ of ATLAS becomes [[ ]]. As a result, the ATLAS downcomer wall has a larger stored energy per unit fluid volume than the APR1400 design by a factor of [[ ]]. The applicant provided a summary of the comparison of the major scaling values in Table 1 of the response to AI-82. The applicant concluded that these scaling conditions provide more conservative conditions, particularly when the downcomer boiling phenomena are simulated in the ATLAS facility.

TR Section 2.2 in Appendix E states that the conditions at the start of the reflood period of the APR1400 LBLOCA are given as the initial and boundary conditions for ATLAS Tests 9, 11 and 15. Presumably, the applicant determines these conditions based on code calculations. Therefore, in AI-83 [Ref. 3], the NRC staff requested the applicant to confirm this understanding and clarify whether it uses RELAP5/MOD3.3/K to determine the initial conditions.

The applicant stated [Ref. 4] that RELAP5/MOD3.3 patch 03 obtained the initial and boundary conditions for the ATLAS tests at the start of the reflood period of the APR1400 LBLOCA. However, the code prediction during blowdown and refill periods is not different from RELAP5/MOD3.3/K because, [[ ]]. Therefore, the applicant considered the code assessment for blowdown period in TR Appendices C and D to be valid in so far as the adequacy of code used in the determination of the initial condition for the ATLAS tests is concerned.

TR Figures 2-9 and 2-15 in Appendix E show the SIT-FD injection rate (high- and low-flow) for ATLAS Tests 9 and 11. The tests start from the reflood period and therefore, as shown in Figure 2-6 in Appendix E, the applicant does not consider a portion of the SIT-FD flow. In AI-84 [Ref. 3], the NRC staff requested the applicant to explain how the time at which SIT-FD injection, and therefore, the injection rate in the tests, is determined. In addition, the NRC staff also requested the applicant to explain the reason for the difference in the injection rate between the four SITs as seen in Figure 2-9 and especially in TR Figure 2-15.

In the response, the applicant stated that it determined the initial condition of the reflood test and the injection rate of the SIT-FD in the ATLAS tests using the [[ ]], which it then verified against measurements.

The applicant provided an explanation of the test initialization procedure. In the drainage step, the applicant explained that the power is linearly increased from 0 to a specified value when the water levels in the core and downcomer reach a specified value. Subsequently the power is maintained until the maximum heater rod temperature reached 450°C, at which point the SIT injection signal is triggered. [[ ]] and the power is decreased to follow 120 percent of the ANS-79 decay heat curve. The applicant attributed the differences noted in above in TR Figure 2-9 and Figure 2-15 of TR Appendix E to small differences in the orifice diameter and the behavior of the flow control valves.

The applicant asserts in TR Appendix E that [[ ]] in ATLAS Tests 9, 11 and 15 are not expected to occur in the APR1400 plant. However, the purpose of integral

tests is to demonstrate the expected behavior of the plant. Therefore, stating that certain phenomena observed in the tests are not expected in the plant defeats the purpose of the tests. In addition, the applicant's rationale for not expecting [[ ]] in the plant was unconvincing. As an example, [[ ]] but, such flow will also exist in the test facility which does exhibit the temperature difference. In AI-85 [Ref. 3], the NRC staff requested the applicant to justify the use of the selected ATLAS tests for this assessment.

In response [Ref. 5], the applicant states that downcomer boiling can occur during the refill and reflood periods. The effects of downcomer boiling on the thermal-hydraulic behavior is relatively insignificant because a large amount of ECCW is injected during refill and early reflood periods, while thermal-hydraulic behavior during the late reflood period can be affected by downcomer boiling phenomena due to the small ECCW injection flow rate. [[ ]]. The applicant selected two uncertainty parameters for downcomer phenomena in CAREM, and determined uncertainty parameter ranges and distribution functions based on the assessment of ATLAS tests.

In addition, the applicant states that the methodology treats the ECCW bypass phenomenon by bias because it is related to many complex phenomena as described in TR Section 4.4.1. Discussions during the January 2016 Audit Meeting and the response to AI-45, Part f [Ref. 5], and AI-85 described [[ ]] in the ATLAS tests as discussed above. Even though ATLAS Test 15 had better initial conditions than Tests 9 and 11, the initial conditions of Test 15 were also [[ ]]. Since the ATLAS tests were used to justify, with other tests, the treatment of ECC water bypass as a bias, the contribution of the ATLAS tests to the bias determination, and specifically which other tests determined the bias, needed to be provided by the applicant. This issue was again discussed in conjunction with AI-45, Part f [Ref. 5], at the June 2016 Audit Meeting. As noted before, some non-typical aspects of the facility in connection with the test initialization procedure resulted in some portions of the tests being unsuitable for use in the determination of the ECCS bypass bias. For the periods when the test was prototypical, the applicant described the information available to perform the code simulation, and for the period when the condition was deemed non-prototypical, a restart of the code was performed with the appropriate temperature input conditions to simulate the prototypical period in the test. The NRC staff agrees that the process of restarting the test simulation with the appropriate temperature data is acceptable.

Based upon the guidance in RG 1.203, the applicant followed the EMDAP process in Step 8(a) by evaluating the integral effect test distortion and utilized the appropriate information from the ATLAS test to meet the test and code simulation objectives for the period in which the test data was prototypical. The NRC staff finds that the applicant's information regarding the metal mass to fluid volume scaling is acceptable, and on the basis of that information concludes that this issue is resolved.

4.4 Biases and Uncertainties

Entrainment [[ ]] and Droplet Evaporation Biases

In [Ref. 26] (Slide 18 of 29) the applicant illustrates the implementation of the entrainment [[ ]] bias in RELAP5/MOD3.3/K, and states that the bias is activated [[ ]] period. However, it is unclear when this bias is activated during the simulation of the

UPTF-10B test and whether the system conditions during the test at the time of activation are similar to those in the APR1400 simulation during [[ ]] period.

In AI-48, Part c [Ref. 3], the NRC staff requested the applicant to clarify whether the [[ ]] implemented in RELAP5/MOD3.3/K is consistent with that used for the UPTF-10B test. In the response to AI-48 Part c [Ref. 4], the applicant stated that the de-entrainment bias activation time in the APR1400 simulation is different from that of the simulation of the UPTF-10B test. The time at which the entrainment [[ ]] bias is activated in the APR1400 simulation is at the [[ ]] period. The applicant stated that the purpose of the UPTF-10B simulation was to investigate the de-entrainment phenomena in the upper plenum during the reflood period, and the start of the reflood in the simulation of the UPTF-10B test corresponds to the start of the liquid injection time. Therefore, both the APR1400 calculation and the simulation of the UPTF-10B test use [[ ]]. The applicant stated that the [[ ]] for de-entrainment bias in the EM. Hence, the NRC staff finds that applicant's position on the bias activation time is acceptable because the bias should be and is active in the specific period of interest, and concludes that this issue is resolved.

The applicant describes in TR Section 5.2.3 the droplet evaporation bias. Selection of the activation time for this bias is different from that for the entrainment [[ ]] activation. According to TR Table 3-5, the APR1400 PIRT assigns its highest importance rank of '5' to [[ ]] in periods 3 and 4. In AI-48, Part d [Ref. 3], the NRC staff requested the applicant to justify the selected activation timing for the droplet evaporation bias because the steam binding effect is expected to be more important during the early reflood period due to the higher rates of flow and entrainment. In response [Ref. 4], the applicant stated that, due to the reasons described in the response to AI-71 [Ref. 4], the activation timing for droplet evaluation bias is [[ ]] because of the high flooding rate during the early reflood and excessive steam binding bias evaluation methodology.

The NRC staff finds that the applicant's information clarifying the activation times for the entrainment [[ ]] bias and the droplet evaporation bias are acceptable, and on the basis of that information concludes that this issue is resolved.

Best-Estimate vs. Conservative Assumptions

The applicant presented the CAREM approach in the TR as a best-estimate methodology. However, the actual implementation of the methodology incorporates a mixture of best-estimate analysis that also uses conservative values for some parameters selectively. The reason that some items are treated conservatively is identified only for certain parameters. Therefore, in AI-5 [Ref. 3], the NRC staff requested the applicant to provide the basis for the determination of when an item is to be treated by using conservative values instead of in a best-estimate manner.

In response to AI-5 [Ref. 5], the applicant provided Table 1 "Phenomena treated conservatively in CAREM" and provided a basis and justification for conservative treatment. For each of the components in Table 1, the applicant showed the process/phenomena and PIRT ranking. The applicant cited the guidelines provided by RG 1.157 regarding justification for the introduction of conservatism in a best-estimate model. For each of the process/phenomena, the applicant provided a justification for the conservatism and the category of conservatism in accordance with RG 1.157.

In response to RAI 7-8567, Question No. APR1400-4, [Ref. 1.i], the applicant described the parameters that are treated statistically by the uncertainty analysis, the parameters that are assigned conservative values, and the parameters that are biased. This response provides the basis for the applicant's determination of when a process/phenomenon is to be treated by a conservative approach and not in a best-estimate manner within CAREM.

The NRC staff finds that the applicant's information clarifying the basis for the determination of when an item is to be treated conservatively instead of in a best-estimate manner is acceptable, and on the basis of that information concludes that this issue is resolved.

ECCS Bypass

NRC RAI No. 7-8567, Question No. APR1400-4 [Ref. 10] requested the applicant to identify which phenomena are treated by other uncertainty parameters, which are treated conservatively and as biases. In Table 3, "Modified APR1400 PIRT treated by biases" of its response [Ref. 1.h], the applicant clearly designates which processes/phenomena are treated by biases. [[ ]], which the applicant stated was an intrinsic and appropriately calculated code parameter, there are [[ ]] processes/phenomena which are combined into the steam binding bias, and [[ ]] processes/phenomena which are combined into the ECCS bypass bias, as discussed in the response above. While the CSAU process discusses the combination of some processes/phenomena into the ECCS bypass bias determination, the basis and justification for the number of processes/phenomena combined into a single bias by the applicant in the CAREM methodology is not clear to the NRC staff. This question was posed in AI-40 [Ref. 3].

In the response to AI-40 [Ref. 5], the applicant stated that the cited process/phenomena are not included as part of the SRS uncertainty assessment process, but are treated as biases, as discussed in Section 4.2.3 of the Topical Report. The applicant noted that test facilities designed to scale power to volume can result in a bias due to scale distortions and that phenomena affected by these scale distortions should be treated as biases which are evaluated based on assessments against full-scale test data. In response, the applicant provided Table 1, "PIRT items treated by biases." This table indicated that it considered [[ ]] part of the ECC bypass bias, [[ ]] as part of the ECC bypass bias, and [[ ]] as part of the ECC bypass bias.

Table 1 in the response to AI-40 corresponds to Table 1 in the response to AI-19 Part a [Ref. 4], which as noted earlier, does not completely agree with Table 3 in the response to RAI 7-8567, Question APR1400-4. The applicant's response to AI-40 [Ref. 5] stated that the processes/phenomena are all included in the ECCS bypass bias. While in a different text format, Section 4.2.3 of TR Revision 1 lists the various phenomena that are evaluated by the applicant as biases, which therefore resolves the inconsistencies noted above.

The NRC staff finds that the applicant's information justifying the combination of bias terms into a single bias does not render the analysis results unrealistic and is consistent with the guidance in RGs 1.203 [Ref. 18] and 1.157 [Ref. 17] to incorporate biases in the analyses in absence of best-estimate models. Therefore, on the basis of that information, the staff concludes that this issue is resolved.

Dissolved Nitrogen

NUREG/CR-5249 [Ref. 19] explicitly addresses the bias due to the presence of dissolved nitrogen in the injected coolant. The RELAP5/MOD3.3 code does not model any dissolved nitrogen in the liquid, and therefore, the predicted reflood peak cladding temperature does not reflect the effect of dissolved nitrogen coming from coolant injected from the SITs. In AI-20 [Ref. 3], the NRC staff requested the applicant to provide an assessment of the effect of dissolved nitrogen on the PCT and transient results.

In the Revision 1 response to AI-20 [Ref. 6], the applicant stated that Section 5.2 of [Ref. 28] evaluated the dissolved nitrogen release from the SIT-FD. The applicant states that this reference conservatively estimates the mass and volume flow rate of dissolved nitrogen from the SIT-FD, and analyzes the effects on the fluidic device K-factors as shown in Figure 5.2-1 and Figure 5.2-2 of that document. The applicant states that the estimated mass and volume flow rate effect due to dissolved nitrogen is small compared with contribution of ECCW from the SIT-FD.

Furthermore, the applicant performed a sensitivity calculation to evaluate the dissolved nitrogen effects on the refill phase using APR1400 LBLOCA EM. In that calculation the amount of dissolved nitrogen assumed was based upon the following: (1) nitrogen gas is fully saturated in the water of SIT-FDs, (2) the amount of dissolved nitrogen in the water is proportional to the partial pressure of the nitrogen gas, and (3) all of the dissolved nitrogen is released when the water from SIT-FDs is discharged through the DVI nozzle. In that calculation, the applicant did not consider the pressure change in the tank, and larger amounts of dissolved nitrogen were released than in the evaluation in [Ref. 28] to make the effects of dissolved nitrogen more apparent. The applicant stated that it used four time-dependent volumes and junctions to simulate the dissolved nitrogen gas release and it installed each set of time-dependent volumes and junctions at the same position as the DVI. A control variable in RELAP5/MOD3.3/K that depends upon the mass flow rates of SIT-FDs controlled the amount of released nitrogen gas. The applicant stated that most of the LBLOCA analysis shows that the refill period ends at about 33 seconds after the break; therefore, the study focused on the timeframe between the initiation of the break and 50 seconds after. The applicant provided a figure showing the total dissolved nitrogen gas release rate from the four SIT-FDs as a function of time. The peak mass flow rate of dissolved nitrogen was about 2.7 kg/s, which exceeds the maximum value in [Ref. 28]. However, the amount of dissolved nitrogen compared to the discharged water mass flow rate from SIT-FDs is insignificant.

Figure 2, provided as part of the response, shows the core collapsed water level. The bottom of the active core is about 2.5 m and the collapsed water level reaches that elevation at about 35 seconds in both the nominal case and the case with the dissolved nitrogen. Therefore, the applicant concluded that the dissolved nitrogen gas has no effect on the duration of the refill phase. Since the discharged water mass flow rate from SIT-FDs is large, the dissolved nitrogen effects are negligible, and this result is consistent with the description in [Ref. 28].

The NRC staff concurs with the applicant's assessment that the effect of dissolved nitrogen on the PCT and transient results is insignificant. Therefore, this issue is resolved.

Hot Channel

Section 4.3.1.1 of NUREG/CR-5249 discusses the use of different nodalizations in the core to evaluate the hot channel bias. The TR does not document or discuss such sensitivity studies. In AI-27 Part e [Ref. 3], the NRC staff requested the applicant to provide the basis for the selected radial nodalization of the core and upper plenum.

The applicant stated in the response to AI-76 [Ref. 5], that the [[ ]] utilized by CAREM, in which [[ ]] is adequate to model the core and also includes some conservatism coming from using the higher F_r value for the hot assembly. In addition, the applicant stated that various code assessments confirmed the acceptability of the [[ ]] as described in TR Appendices, and therefore, no additional sensitivity studies for the multi-channel model of the core need to be performed. Furthermore, the applicant noted that due to the issues related to the modeling of the upper plenum during the reflood period related to the entrainment of liquid droplets, CAREM incorporates a conservative model as part of the determination of the steam binding bias.

In the June 2016 Audit Meeting, the applicant reiterated the position that the [[ ]] nodalization is adequate due to the confirmation provided by the code assessment studies provide in TR Appendices C, D, and E. Since the analysis utilized [[ ]]. Then, the applicant proceeded to present the results of a [[ ]]. The applicant believes that [[ ]]. This is discussed in response to AI-27 and AI-47 [Ref. 5]. Despite the [[ ]].

The NRC staff finds the applicant's basis for the radial nodalization of the core and upper plenum is acceptable, and on the basis of that information concludes that this issue is resolved.

Reactor Vessel Downcomer

TR Section 4.2.2.2.1 discusses the uncertainty parameters for important phenomena or processes that occur in the reactor vessel downcomer. The applicant attributes all except three such phenomena to the ECCW bypass and states that the ECCW bypass bias accounts for them. Therefore, the applicant determines the uncertainty for only three processes or phenomena of [[ ]] important phenomena identified in the APR1400 PIRT. Among the phenomena that the applicant lumps together with the ECCW bypass bias is [[ ]].

The NRC staff and the applicant discussed the issue of flashing in the downcomer in the June 2016 Audit Meeting as part of the discussion of the number of process/phenomena that are included in the biases in CAREM. The applicant presented the following information at the meeting:

Flashing uncertainty should be considered directly

1. *In RELAP5, flashing is not considered as a correlation, but as a result of governing equations and state equations.*

2. *The uncertainty of flashing is considered in CAREM is considered as other parameters*

3. *[[ ]].*

4. *Initial system pressure may affect the subsequent flashing behavior*

The NRC staff notes the conflict between these facts and believes that flashing is a phenomenon associated with several components as stated in the TR PIRT. The applicant stated that flashing is a consequence of the solution of the mass and energy conservation equations through the equations of state, and therefore its treatment as one of the process/phenomenon lumped into a bias does not strictly correspond to the general approach for addressing biases in RG 1.157. The response to AI-39 described proposed changes to TR Section 4.2.2.2.1 to make the discussion of flashing in the downcomer consistent with the treatment in other components by stating that [[ ]]." The details are then documented in TR Revision 1, Sections 4.2.2.6.1 and 5.1.7.

The NRC staff finds that the applicant's decision to modify the TR to provide consistency in the treatment of the phenomenon of flashing follows the guidance in RG 1.203, and on the basis of the information provided concludes that this issue is resolved.

<u>*Bias Grouping*</u>

In response to RAI 7-8567, Question APR1400-4, the applicant provided information regarding which processes/phenomena are treated conservatively, through a bias, through uncertainty representation or are considered as part of other parameters. In the response to RAI 7-8567, Question APR1400-4, Table 2 lists the processes/phenomena which are treated conservatively in CAREM and Table 3 provides a list of the processes/phenomena which are treated by biases.

In the responses to AI-14, AI-17, and AI-19 [Refs. 3 and 4], the applicant identified several processes/phenomena that are combined into a single bias. The applicant did not describe or justify the process and basis for combining multiple processes/phenomena using the guidance in RG 1.157 or NUREG/CR-5249. The CSAU approach discussed in NUREG/CR-5249 indicates that combining biases may be appropriate:

(1) when the data base is insufficient to judge the nodalization and therefore a separate bias for plant noding may be added to determine uncertainty;

(2) due to the difference between code-calculated and experimental data for important phenomena, although a more powerful and successful technique determines the individual uncertainty contributions;

(3) when the effects of scale must be combined with other uncertainty contributions; or

(4) when the absence of data may necessitate a bounding analysis.

RG 1.157 states:

> *"The effects of all important variables should be considered. If it is not possible or practical to consider a particular phenomenon, the effect of ignoring this phenomenon should not normally be treated by including a bias in the analysis directly, but should be included as part of the model uncertainty."*

RG 1.157 recognizes that a bias may be introduced when a particular model does not require a totally best-estimate calculation. In general, the APR1400 CAREM does not provide a justification or methodology for determining the single bias that is to be used to represent a combination of processes or phenomena. The NRC staff believes that such a justification is important when the processes or /phenomena are of high importance.

During the June 2016 Audit Meeting discussion the applicant provided presentation material which described the process of bias treatment and identified that power-to-volume scaled test facilities have components that cause bias due to scaling distortions. The applicant further identified that the one-dimensional modeling capability of RELAP5 is known to be inadequate for a complete best-estimate depiction of all the phenomena that are combined into the biases. By determining a conservative bias in the code prediction of the test facility response, the phenomena are effectively combined into a single bias consideration.

Regulatory Guide 1.157 states that conservatism may be introduced for several reasons, including: "*…2. The uncertainty of a particular model is difficult to determine, and only an upper bound can be determined. 3. The particular application does not require a totally best estimate calculation, so a bias in the calculation is acceptable*." The CSAU methodology discussed in NUREG/CR-5249 provides some clarification, stating: "*In the absence of a data base sufficient to quantify scaling effects in the form of biases and deviations, it may become necessary to perform a bounding analysis and employ the results as a separate bias (i.e., a penalty or benefit) in the total uncertainty statement*."

Based upon the information the applicant provided in response to AI-40, the information in the TR regarding the test simulations, and the clarifications in the June 2016 audit meeting, the NRC finds that the applicant follows the guidance provided by Regulatory Guide 1.157 and NUREG/CR-5249, and provides a reasonable method for representing multiple phenomena with a single bias. The NRC staff finds that the applicant's method for combining biases into a single bias is acceptable, and on the basis of the information provided concludes that this issue is resolved.

Containment

TR Section 4.2.2.6.2 discusses the applicant's uncertainty evaluation approach for the containment related parameters. Containment pressure history during LBLOCA is especially important during the reflood phase since it acts as the backpressure for reflooding the core.

Break mass flow rate and system pressure drop are dependent on the containment backpressure during the reflood phase. In CAREM, containment backpressure is calculated by CONTEMPT4/MOD5 when choked flow is stopped at the end of blowdown. Prior to that, a representative containment backpressure is used.

TR Section 4.2.2.7.2 states that relevant containment parameters are selected to ensure conservative results and containment backpressure is calculated conservatively. However, the applicant did not provide any description of the CONTEMPT inputs (e.g., for heat transfer surface area, the wall condensation model, etc.), nor any results of sensitivity calculations that demonstrate the conservatism of the selected inputs relative to the LOCA backpressure calculations. Therefore, in AI-43 [Ref. 3], the NRC staff requested the applicant to provide justification for the statements regarding conservatism in modeling the containment response.

The applicant's response to AI-16 [Ref. 4] provided some of the containment calculation input parameters and the type of values selected for the calculation of a minimum containment backpressure calculation. The applicant's response concluded that sensitivity calculations demonstrating conservatism are unnecessary. The NRC staff and the applicant discussed treatment of containment backpressure during the January 2016 Audit Meeting and agreed that lower containment pressures tend to result in lower core reflooding rates since break flow is increased. This is conservative from a PCT perspective.

The NRC staff finds that the applicant's information regarding CONTEMPT input and the results of sensitivity calculations that demonstrate the conservatism of the selected inputs relative to the LOCA backpressure is acceptable, and on the basis of that information concludes that this issue is resolved.

ECCW Bypass

According to the TR, the applicant relied upon code assessments to determine the ECCW bypass bias during the refill period. The code assessments revealed that the water inventory in the lower plenum predicted at the end of the test exceeded the measured amount by several hundred kilograms. According to the TR, the applicant conservatively uses [[ ]] of the ECCW bypass during the refill period. The applicant then chooses to evaluate the bias in the prediction of the PCT due to ECCW bypass [[ ]] during the refill period. Similarly, the applicant also used code assessments to determine the ECCW bypass phenomena during the reflood period. Results of those code assessments showed that the RELAP5/MOD3.3/K code conservatively calculates ECCW bypass during reflood. The applicant does not credit this conservatism for ECCW bypass during the reflood phase in the evaluation of the PCT bias in the plant sensitivity calculations. As a result, the applicant applies the ECCW bypass bias only during the refill phase of the accident. In AI-45 [Ref. 3], the NRC staff requested the applicant to clarify several concerns regarding the determination of the ECCW bypass bias that are discussed in TR Section 4.2.3.1:

- In AI-45 Part a [Ref. 3], the NRC staff requested the applicant to clarify the approach used to incorporate the ECCW bypass bias in the code during the blowdown, early reflood, and late reflood periods. In response to AI-45, Part a [Ref. 4], the applicant stated that [[ ]] to the code-calculated-ECC
- bypass and it is only for the refill ECC bypass. Separately from this, the ECC bypass during the early and late reflood periods is assessed based on the code assessment studies against MIDAS, UPTF-21D, and ATLAS test 9, 11, and 15. Results of the

code assessment using these reflood tests showed that the RELAP5/MOD3.3/K code conservatively calculates ECC bypass during reflood. Therefore, these studies justify the conservatism of the code without the need to apply a bias for the reflood ECC bypass. Following the guidance in RG 1.157 regarding comparisons to different tests to ensure that a reasonable estimate of the bias has been obtained, the NRC staff finds that the applicant's application of the ECC bypass refill bias is appropriate for that period and the reflood ECC bypass bias was appropriately determined for the reflood period based upon test assessments.

- In AI-45 Part b [Ref. 3], the NRC staff requested the applicant to provide comparisons from the simulation of the UPTF-4A test with and without the ECCW bypass bias applied to show that the selected bias is appropriate. The applicant stated [Ref. 4] "*The purpose of the UPTF-4A test assessment is not to confirm the appropriateness of bias, but to determine the value of the bias.*" The applicant did not expect to obtain good agreement with the test data when the bias was activated, due to the impact of the bias on local predicted core and downcomer levels. However, the applicant believes that it is important to get agreement with test data for global parameters, such as vessel inventory. The applicant included results in the response to AI-45, Part b, which showed the liquid mass through the breaks and the liquid mass extracted before and after activation of the refill ECC bypass. The run with bias showed slightly smaller break flow mass than the run without bias, but the difference was not significant. In the run with the bias, the applicant confirmed that the total liquid out of the vessel is larger than the test data, which is [[ ]]. Therefore, the applicant believes that, for the vessel inventory, the activation of ECC bypass is logically correct. The NRC staff examined the results of the requested analysis with and without the bias and determined a slightly different mass was extracted through the break with and without the bias. The NRC staff concurs that the applicant correctly followed the guidance in RG 1.157 for the application of the bias and that the key parameters meet the expected response, which addresses the concern.

- In AI-45, Part c [Ref. 4], the NRC staff requested the applicant to comment on the applicability of UPTF-4A to determine the ECCW bypass bias it initiates from a system pressure of 1200 kPa (i.e., 1.2 MPa) while the pressure in the refill period in APR1400 is approximately 6 MPa. The applicant stated:

  "*The initial conditions of UPTF-4A test were determined from the TRAC analysis of a US PWR and starts at 1.2 MPa, while the SIT injection in APR1400 starts at 6 MPa. The beginning of SIT injection is only dependent on the setpoint of SIT, hence is different due to the design differences between reference plant of UPTF-4A and APR1400. Although the SIT injection starts at 6 MPa, the RCS pressure still drops rapidly after the injection of SIT, and at about 25 seconds the RCS pressure becomes below 1.2 MPa which is consistent with the initial condition of UPTF-4A. The period between 6 MPa and 1.5 MPa still has the strong characteristic of blowdown and therefore not adequate to be considered as the ECC bypass for refill."*

The applicant concluded that the difference in the initial conditions for UPTF-4 and the APR1400 minimally affected the ECCS refill period (e.g., period between 6 MPa and 1.5 MPa is considered to be representative of the blow-down period). Therefore, the applicant considers the UPTF-4A test is applicable for the selection of ECCS bypass

bias for the analysis. The NRC staff finds that the conditions of the APR1400 simulation and the UPTF simulation are somewhat different. The point at which the APR1400 SIT begins to inject at 6 MPa through the conditions at end of blowdown, with a pressure of approximately 1.5 MPa, are not appropriate for the calculation of ECC bypass during refill.

The appropriate conditions after the pressure has fallen below 1.5 MPa correspond approximately to the test conditions for the UPTF-4A test. Consequently the NRC staff concludes that the applicant correctly followed the guidance in RG 1.157 to utilize the appropriate test conditions to determine ECC bypass during the refill period, such that the concern is resolved.

- In AI-45, Part d [Ref. 3], the NRC staff requested the applicant to further justify the selection of UPTF-4A with cold leg injection for determination of the ECCW bypass bias over the tests with a configuration similar to that in APR1400. The applicant stated [Ref. 5] that the ECC bypass during the end-of-blowdown and refill period is mainly dependent on the flow mixing behavior below the bottom of the cold leg. Therefore, the ECC injection type is less important in comparison to the depressurization process. Also, the test configurations of UPTF-21A and -21B are not exactly the same as the LBLOCA conditions during the end-of-blowdown and refill period.

During the June 14, 2016, Audit Meeting [Ref. 12], the applicant presented information regarding the UPTF-21A, UPTF-21B and UPTF-4A tests to justify the selection of the UPTF-4A cold leg injection test for the determination of the ECCW bypass ratio for the APR1400 with DVI [Ref. 27]. As the applicant noted in prior responses:

- The ECC bypass during end of bypass and refill is mainly dependent on the flow mixing below the bottom of the cold leg and therefore the delivery process to that point is of less consequence,

- The depressurization process is more important to the bypass determination than the injection point,

- The cold leg injection configuration in UPTF-4A can be used to quantify the ECC bypass bias since it captures the important phenomena, and

- The large break LOCA conditions in tests UPTF-21A and UPTF-21B are not exactly the same as those expected in the APR1400.

In that same meeting, the applicant described the features of the UPTF-21A -21B tests that renders them less suitable for determination of the ECC bypass bias:

- The intentions of the tests were to investigate the steady state downcomer counter-current flow limitation (CCFL) and provide data only for certain flow rate conditions,

- The DVI configuration UPTF tests 21A and 21B were counter-part tests to the cold leg injection (CLI) configuration tests of UPTF-5, 6 and 7,

- The intact loops in the DVI configuration tests UPTF-21A and -21B were blocked to match the CLI configuration tests,

- The tests were at steady state conditions,

- The pressure was approximately 3 bar for the test UPTF-21A, and tests UPTF-21B-I, -II & -III were sensitivity tests,

- The DVI configuration test closest to the expected plant conditions was test UPTF-21A, which had steam injection through the core simulator and steam generator simulator (SGS),

- The DVI configuration test UPTF-21B-I had hot ECC fluid and test UPTF-21B-II and –III had no steam generator simulator.

The applicant stated that there were no other UPTF tests that are more representative of the end of bypass to refill period which had a representative DVI configuration. The closest DVI configuration test, UPTF-21A, shows only one state of the process and not the whole end of bypass to refill process. Consequently, the applicant concludes that this test cannot be used to quantify the bias due to the difficulty in converting the results into a bias. In contrast, the applicant stated that the CLI configuration test UPTF-4A can be used to determine the bias as the difference in the [[ ]] during the test. Since the DVI injection point in the APR1400 design is above the cold legs in the downcomer, the distance that the steam and ECC water are expected to be in contact until they go through the broken cold leg, is almost the same as that of the cold leg injection plants. Consequently, the applicant concluded that the ECC injection type was less important than the expected mixing flow behavior, for which the test UPTF-4A would be representative.

The NRC staff recognizes and acknowledges that there are no tests with the exact DVI configuration and test conditions that are appropriate to determine the ECC bypass bias. Based upon the information provided by the applicant regarding the inadequacy of the DVI configuration UPTF tests, and the fact that the UPTF-4a cold leg test conditions are appropriate to the conditions for which DVI ECC injection will occur in the APR1400, the NRC staff concurs that the test UPTF-4A provided the most representative conditions for assessment, and this resolved the concern.

- In AI-45, Part e [Ref. 3], the NRC staff requested the applicant to discuss the differences between the APR1400 and UPTF nodalizations and their impact on the conclusions drawn from the UPTF simulations. In response [Ref. 4], the applicant stated that, although the downcomer modeling of UPTF tests is not exactly the same at that of the AP1400, the principle idea employed in the modeling of the [[ ]]. The applicant explained that in modeling [[ ]]. In UPTF, there are four (4) hot legs and four (4) cold legs, while the APR1400 has two hot legs and four (4) cold legs. In order to preserve the [[ ]]. The difference in the

axial nodalization of the downcomer is mainly due to the geometrical difference between the APR1400 and the UPTF rig. The [[ ]] values are similar and also the connections to the cold legs are consistent in both nodalizations by having the same single-junction connections. RG 1.203 states that the modeling of the test facility should be consistent between the experimental facility and similar components in the nuclear power plant. Due to differences in the configuration of the UPTF test facility and the APR1400, the NRC staff finds that the modeling of the UPTF facility follows the modeling approach employed for the APR1400 and that the UPTF nodalization is acceptable, and therefore resolves the concerns.

The applicant stated that the upper plenum noding in UPTF in TR Appendix F is different from that of APR1400. The noding is based on the upper plenum noding scheme developed for the analysis of other nuclear power plants in Korea which have upper plenum injection. In this approach, every upper plenum structure is modeled using pipes and multi-junction components. The applicant stated that his noding scheme was employed because the upper plenum SI mixing phenomena were considered to be important in CAREM. However, in the APR1400 modeling, the detailed investigations on these phenomena were not necessary, and therefore, the multi-channel upper plenum noding scheme is not used. The UPTF nodalization is only used in the assessment of the test UPTF-21d. The scheme used in the assessment of the test UPTF-10B was performed using the simplified upper plenum noding, which is consistent with the APR1400 nodalization. See the response to AI-48, Part b [Ref. 4]. In the assessment of UPTF-21D, the impact of the modeling differences does not appear to be significant. The complicated upper plenum noding may only affect the flow behavior in the upper plenum, which would be very limited in the DVI injection test, such as UPTF-21D.

During the January 2016 Audit Meeting [Ref. 12], the NRC staff and the applicant agreed that it is important for KHNP to revise the UPTF nodalization and repeat the study. In TR Revision 1, the applicant revised the UPTF-21D nodalization to be consistent with the APR1400 nodalization and then re-ran that simulation. The applicant also performed an assessment of UPTF-21D with a simplified upper plenum noding. The results of the downcomer level and the break flow showed that the simulation results are slightly more conservative than the test results. The applicant determined that the conclusions from the assessment in the TR were unchanged. The revised simulation of UPTF-21D using the nodalization applied in the APR1400 showed slightly more conservative results and followed the guidance in RG 1.203 that there should be consistency between the modeling of the experimental facility and the modeling of the nuclear power plant. Therefore, the NRC staff finds the revised modeling of UPTF-21D acceptable, which resolves the concern.

In AI-45, Part f [Ref. 3], the NRC staff requested the applicant to provide justification for neglecting the non-conservatism shown in the first 200 seconds of ATLAS tests 9, 11 and 15 in the ECCS bypass prediction during the early reflood period. The applicant's response [Ref. 5] stated that, as discussed in TR Appendix E, the result of the initial period (the period before 200 seconds after the beginning of the test) of the ATLAS Tests 9 and 11 include [[ ]]. The applicant stated that the cited behaviors are due to [[ ]] during the reflood period of APR1400. During the January 2016 Audit Meeting, the applicant stated that the results of ATLAS Tests 9 and 11 could be [[ ]]. Since the potential for, and the effect of,

downcomer boiling is a function of the metal heat content of the vessel wall during the late reflood period of low safety injection flow, the NRC staff requested the applicant to describe the process by which the code metal heat content was adjusted, the process used to verify that the values corresponded to the metal heat content of the vessel wall in the ATLAS tests at the time of code restart, and how those values are prototypic of the expected behavior in the APR1400. In the June 2016 Audit Meeting, the applicant noted that there were sufficient measured data including wall temperatures to characterize the metal stored energy for setting the initial conditions for the ATLAS test simulations. In discussing the ATLAS IET earlier in this SER, the NRC staff found the restart of the simulation with the appropriate initial conditions for the simulation of the test data with prototypic plant conditions is an acceptable process, and concurred that this addressed the issue.

None of the descriptions of the models for APR1400, MIDAS, or UPTF used to determine the ECCW bypass bias mention the use of a CCFL correlation in the downcomer. The CCFL correlation can impact the prediction of ECCS bypass because such a correlation will reduce the amount of liquid penetrating downwards into the downcomer. In AI-45, Part g [Ref. 3], the NRC staff requested the applicant to clarify whether any such correlation was used, and if so, to describe the resulting impact on the ECCS bypass bias. The applicant's response [Ref. 4] indicated that a CCFL model was not used in any calculations since the model has a strong tendency to amplify numerical oscillations during reflood. The applicant noted that although it does not apply a CCFL model it conservatively calculated the results for test assessment (MIDAS and UPTF) without applying a CCFL model in terms of reflood ECC bypass; therefore, such a bias is not included. CCFL is discussed in detail later in this SER. The NRC staff finds that the applicant's information regarding CCFL on the tie plate and the observed calculated hydraulic phenomena are acceptable, and on the basis of that information concludes that CCFL does not need to be included, which resolves this issue.

TR Appendix F documents the assessment of the RELAP5/MOD3.3/K code against SETs in the MIDAS and UPTF facilities for determination of the ECCW bypass bias in the late reflood period. The expression for the bypass fraction defines that parameter as the ratio of the liquid flow rate discharged though the lower plenum to the total inflow from the safety injection and the condensate and the expression in Equation (1) was incorrect. TR Revision 1 [Ref. 13] incorporates the correct expression for the bypass fraction. The applicant also confirmed that they correctly calculated and compared Figures 2-4, 2-7, and 2-8.

Several concerns were raised regarding the determination of the ECCW bypass bias. The NRC staff has the following findings:

- The ECCW bypass bias determination during refill is appropriate,
- The ECCW bypass bias determination during reflood is appropriate,
- The application of the ECCW bypass bias resulted in the expected response in key parameters,
- The test conditions used to determine the ECCW bypass bias were appropriate, and
- The test nodalizations used to determine the ECCW bias were appropriate.

Therefore, NRC staff finds that the applicant's information regarding the several issues discussed above is acceptable, and on the basis of that information concludes that this issue is resolved.

Steam Binding

In AI-50, Part c [Ref. 3], the NRC staff requested the applicant to clarify whether the steam binding bias is included in the performance of the 'coverage checks' for the reflood period when compared to integral test data. In response [Ref. 4], the applicant stated that the purpose of the "coverage check" was to confirm the adequacy of the number of uncertainty parameters and the uncertainty range of each parameter. The purpose of the bias determination was to evaluate scale distortions, code deficiencies in prediction, or model deficiencies. The bias evaluations were only included in the plant calculations to quantify total uncertainty. The applicant justifies not including steam binding bias in coverage checks, believing that the proper place to examine biases is in the plant calculations that evaluate the total uncertainty, rather than in coverage checks of parameter uncertainties. The NRC staff confirms that this statement is correct according to CAREM, Step 10.

Following the guidance in RG 1.157, the applicant has justified the treatment of the uncertainties and methods for combining the uncertainties include the appropriate point for application of the bias. The NRC staff finds that the applicant's explanation of the interactions between "coverage check" and steam binding bias is acceptable, and on the basis of that information concludes that this issue is resolved.

Previous Experience and Scaling

The applicant, after performing the 181 demonstration calculations, evaluated scale biases for all cases that are expected to be limiting.

> *"Based upon previous experiences, [[ ]] are selected for scale bias calculation."*

In TR Revision 1, [[ ]] cases were selected for the evaluation of scale biases. In AI-65 [Ref. 3], the NRC staff requested the applicant to provide details of the referenced "previous experiences" and the basis for the selection of the cases. The NRC staff also requested the applicant to confirm that the case with the highest second peak is captured in the set that is used for the evaluation of the scale bias.

In response [Ref. 4], the applicant stated that 181 combinations of the uncertainty parameters are used to quantify the uncertainty of the PCT prediction. In those combinations, two cases with the highest PCT are not considered valid results and are therefore not considered for scale bias application. [[ ]] for the selection of the scale bias cases is based on "the previous experiences," for which the applicant explained that it means experience in exercising the code in development, sensitivity studies, and from licensing applications in Korea. The applicant stated that the potential for application of the bias to make a case more limiting is generally lower than [[ ]] as indicated in Table 5-8. In other words, once the limiting reflood PCT case has been selected, no other reflood PCT case has typically been found to become limiting with the application of the bias if the other cases are more than [[ ]] lower than the limiting reflood case. Consequently the applicant selected a conservatively high cutoff of [[ ]]. All reflood PCT cases within [[ ]] of the limiting reflood PCT case are selected for bias application to determine if they become the limiting reflood PCT case for comparison to the licensing limit. The first and second highest cases are neglected even if they are reflood peak PCT cases, since their use would exceed the 95/95 criteria. The NRC examined the reflood PCT cases in TR Revision 0 and TR Revision 1, and observed that a [[ ]] is

sufficiently conservative for the identification of reflood PCT cases for which the bias should be applied for evaluation.

The [[ ]] cases selected for bias evaluation are documented in TR Revision 1, Table 5-5. Four of the [[ ]] cases in TR Revision 1, namely cases 35, 43,142 and 162, do not show any influence of either of the [[ ]]. In AI-66 [Ref. 3], the NRC staff requested the applicant to elaborate on the reasons for the observed behavior. For the revised analyses in TR Revision 1 [Ref. 13], the zero bias values are the result of negative biases that are conservatively limited to a value of zero or are small such that round-off lists it as zero.

Based upon the applicant's response and information in the TR, the NRC staff finds that the applicant's information regarding "previous experiences" and the basis for the selection of the cases is acceptable, and on the basis of that information concludes that this issue is resolved.

4.5 Nodalization

Changes

The applicant states in TR Section 1 that the CAREM methodology described in the TR strictly follows the concept and philosophy of CSAU as documented in NUREG/CR-5249. Comparing Figure 1-1 in the TR to Figure 1 of NUREG/CR-5249, the NRC staff noted that the step that determines whether a noding change is necessary is different between the two figures. In the TR, the determination is made after all the uncertainties and biases have been determined. On the other hand, in NUREG/CR-5249, nodalization is established as part of the second element of the approach (i.e., assessment and ranging of parameters), while biases and uncertainties are determined as part of the third and last element of the process. In AI-4 [Ref. 3], the NRC staff requested the applicant to explain this discrepancy. During the January 2016 Audit Meeting, KHNP provided a presentation on the CAREM process [Ref. 27] describing how and when noding changes occur, stating that prior practice established the original nodalization, which was based upon separate effects tests. KHNP stated [Ref. 4] that they would revise the noding and repeat all of the process steps if changes were warranted to simulate test results or to capture important phenomena. This approach, although not strictly the same as that outlined in the CSAU (NUREG/CR-5249), is considered reasonable, since it provides a mechanism to revise the nodalization if important phenomena could not be captured.

The NRC staff finds that the applicant's information regarding how and when noding changes occur and that this process is based upon prior practice and the results from separate effects tests is acceptable and, on the basis of that information, concludes that this issue is resolved.

Core and Upper Plenum

The NRC staff expects that the predicted entrainment from the upper plenum will be a strong function of the nodalization of the core and the upper plenum. Therefore, in AI-48 [Ref. 3], the NRC staff requested the applicant to describe the nodalization of these components for comparison against UPTF-10B. In response [Ref. 4], the applicant provided the noding for the upper plenum in the APR1400 model and the upper plenum for the UPTF-10B test simulation. The model for the APR1400 upper plenum uses three volumes:

(1) the region above the [[ ]];

(2) the region above the [[ ]]; and

(3) the region above [[ ]].

The applicant explained that in UPTF, there is no such thing as the fuel alignment plate or UGS bottom plate. Therefore, the hot leg bottom and top line substitute for those plates because the vertical locations of those plates are similar with hot leg bottom and top lines. Consequently, the EM uses three volumes for the UPTF:

(1) the region above [[ ]];
(2) the region above [[ ]]; and
(3) the region above [[ ]].

The applicant stated that this makes the APR1400 upper plenum modeling consistent with UPTF-10B upper plenum model. The model does not include details of the upper plenum structures, [[ ]]. These are left as empty volumes in both the APR1400 and UPTF-10B models. The applicant explained that [[ ]]. The applicant provided the rationale for using the UPTF-10B nodalization for APR1400 during the January 2016 Audit meeting [Ref. 12]. The applicant explained that [[ ]].

The NRC staff finds that the applicant's information regarding the nodalization of the APR1400 and the UPTF-10B followed the guidelines of RG 1.203, step 14 which states: "In particular, nodalization and option selections should be consistent between the experimental facility and similar components in the nuclear power plant" and concludes that this issue is resolved.

TR Section 4.2 discusses the applicant's nodalization of the APR1400 plant for use in the CAREM methodology. The nodalization separately models the two hot legs and four cold legs, which are spaced equally, 60-degrees apart in the circumferential direction. The model includes two identically modeled loops, except that one of the loops includes the pressurizer and the other loop (i.e., the loop without the pressurizer) models the cold leg break between the pump and the reactor vessel. The nodalization for each loop contains the hot leg piping, the steam generator U-tubes, and the fluid volumes of the steam generator primary and secondary sides, two pump suction legs, two pumps, and two pump discharge cold legs.

The reactor vessel nodalization consists of the downcomer, upper and lower plenum, and the core regions. The reactor vessel downcomer is divided into six azimuthal downcomer channels according to the loop models and each channel has 10 axial volumes. The [[ ]].

The NRC staff developed several questions regarding the nodalization selected by the applicant. In AI-27 [Ref. 3], the NRC staff requested the applicant to respond to these questions. The NRC staff and the applicant discussed these questions during the January 2016

Audit Meeting [Ref. 12] and the discussion below characterizes the applicant's rationale behind each question.

TR Section 4.2.1 states that the APR1400 nodalization of the reactor vessel "...follows the typical pressurized water reactor nodalization..." However, the applicant did not provide any references or details to support this assertion or to demonstrate how the applicant determined "typical pressurized water reactor nodalization." Tables 7 and 8 in NUREG/CR-5249 provide the basis for the nodalization chosen in that study and compare the chosen nodalization to that used to represent several integral test facilities. The NRC staff requested the applicant to provide additional information supporting the assertion that the selected nodalization for APR1400 is "typical" using the content of Tables 7 and 8 in NUREG/CR-5249 as a guide. In AI-27, Part a [Ref. 3], the NRC staff requested the applicant to provide a reference for the documentation of the nodalization sensitivity studies that were performed to determine the CAREM nodalization that will be used to support license submittals.

In the response to AI-27, Part a [Ref. 4], the applicant stated that the APR1400 plant nodalization was based upon code assessments related to the nodalization, code assessments using SET and IET of the assessment matrix and nodalization examples and code user guidelines described in RELAP5/MOD3.3 Code Manual, Volume III and Volume V [Ref. 20]. Specific nodalization details for the upper guide structures are in [Ref. 30]. The applicant stated that the experimental assessments described in the appendices of the TR also confirmed the applicability of the selected nodalization.

The NRC staff finds that the applicant's information regarding the documentation of the nodalization sensitivity studies and the SETs and IETs follows the guidance provided in RG 1.203 and is consistent with industry practice, and on that basis concludes that this issue is resolved.

In the response to AI-27, Part c [Ref. 4], the applicant stated that [[ ]] were chosen to represent the downcomer based on [[ ]] However, the applicant did not provide any information as to what phenomena are investigated. Moreover, the applicant introduced the ECCW bypass as a bias that is not dependent on the nodalization of the downcomer. Therefore, the NRC staff requested the applicant to list the applicable phenomena considered. In AI-27, Part b, the NRC staff requested the applicant to provide information regarding the potential for distortion due to nodalization.

In the response to AI-27, Part b [Ref. 4], the applicant stated that the CAREM model uses a multi-channel downcomer modeling technique where [[ ]]. The NRC staff finds this approach is appropriate and consistent with industry practice. The applicant stated that the distortion of the phenomena that are addressed in the TR include ECCS bypass, the ECC bypass ratio, level at the downcomer, break flow, etc., and are the phenomena of concern. Since the applicant evaluated the ECC bypass bias against the UPTF and MIDAS tests, the bypass itself is somewhat dependent on the nodalization. The applicant stated that the downcomer nodalization in the UPTF and MIDAS test are consistent with that of the APR1400. The applicant stated that it is necessary to model the downcomer by azimuthally [[ ]] for the APR1400. The applicant provided a figure depicting the nodalization of the reactor vessel downcomer in the circumferential direction. The

applicant stated that the applicable phenomena considered relate to the ECC bypass. The applicant stated that the APR1400 modeling approach was based on the determination of biases using the assessment from the UPTF and MIDAS tests, which utilized a specific nodalization and therefore had some nodalization dependence. Therefore, it is reasonable to extend the bias determination to the APR1400 simulations. Based upon the information in the TR and the applicant's description of the phenomena investigated, the NRC staff finds that the ECCW bypass bias is dependent upon the nodalization inherent in the consistent use between the UPTF and MIDAS tests and the APR1400. Because of this, a condition is imposed upon the application of the APR1400 downcomer modeling.

The NRC staff imposes the following CAREM methodology application Condition (2):

> *Changes to the downcomer modeling shall require additional justification and NRC approval.*

The NRC staff finds that the applicant's information regarding the nodalization of the downcomer is acceptable, since it is justified by comparison to experimental benchmark tests with similar model nodalization, and on the basis of that information concludes that these issues are resolved, subject to Condition (2).

<u>*Plant Nodalization Using Code Assessment Studies*</u>

According to NUREG/CR-5249, the plant nodalization should be informed by, and to the extent possible, follow that used for the EM code assessment studies. The APR1400 plant shown in TR Figure 4-1, and the ATLAS facility shown in Figure 3-3 in TR Appendix E, show differences in the nodalization of the vessel. These differences include the nodalization of the downcomer, lower plenum, upper plenum, and the upper plenum to dome connection. In AI-27 Part m [Ref. 3], the NRC staff requested the applicant to explain the reasons for the differences.

In response [Ref. 5], the applicant explained that the lower downcomer to lower plenum noding differences resulted from geometric differences between the facility design for the APR1400 and design of the ATLAS test facility. Similarly, geometric differences in the lower plenum account for the differences in the nodalization in the lower plenum. The ATLAS test facility [[ ]] prototypical of the APR1400 design. The NRC staff determined that the deviation in the RELAP5 noding should not be the same, but reflect the actual designs for the test facility simulation and for the plant analyses.

For the upper plenum, the applicant noted that a [[ ]] for the APR1400 is utilized based upon sensitivity studies which indicate that it is conservative for blowdown quenching phenomena. Since the ATLAS facility was designed to assess the behavior during reflood, the detail desired for blowdown phenomena is not necessary for a reflood test. Consequently, a [[ ]] is utilized for the ATLAS facility. The applicant concluded that the thermal-hydraulic effects of the [[ ]]. However, the response to AI-5 [Ref. 5] showed a difference between the [[ ]]. This was discussed during the June 14, 2016, Audit Meeting where the applicant explained that the difference between the post-blowdown peak PCT response between the [[ ]]. Since the ATLAS facility would not experience the phenomena predominant during blowdown, the effects driving the difference during reflood would not be present. The NRC staff concludes that the lack of the blowdown forces would

result in an insignificant difference in the response of the ATLAS facility whether a [[ ]] is used. Consequently, the NRC staff finds that the [[ ]] ATLAS facility is sufficient for assessment purposes.

Furthermore, the NRC staff finds that the applicant followed the guidance in RG 1.203, which in step 14 states:

"In particular, nodalization and option selections should be consistent between the experimental facility and similar components in the nuclear power plant."

The differences between the noding utilized for the ATLAS facility and the APR1400 plant were determined to be due to facility design differences and differences in the prototypic conditions expected during reflood. The NRC staff finds that the applicant's information regarding the differences in nodalization between the APR1400 model and the model employed for the ATLAS facility for the downcomer, lower plenum, upper plenum, and the upper plenum to dome connection is acceptable, and on the basis of that information concludes that this issue is resolved.

Mixture Level

The NRC staff recognizes that the froth or mixture level is typically important in determining the quench front progression. The APR1400 nodalization, as described in the TR, uses [[ ]] control volumes in the core in the axial direction. Consequently, changing the axial nodalization may change the quench front uncertainty. As a result, in AI-27 Part g [Ref. 3], the NRC staff requested the applicant to explain the basis for the selected core axial nodalization and the level of accuracy that it provides.

The applicant's response to AI-27, Part g [Ref. 4], stated that the nodalization of the plant should be modeled in detail to simulate the design characteristic and the major phenomena. The modeling of core axial nodalization of CAREM is based on assessment against THTF and FLECHT-SEASET tests, of which the rod lengths are the same as the fuel rod of the conventional plant.

According to the applicant, the THTF and FLECHT-SEASET test facilities have the same core length as the plant and the assumption that the power-to-flow area ratio is similar to the value of the fuel assembly prevents the distortion of the thermal-hydraulic behavior in the core due to the scale down. Therefore, the measuring positions of the test facilities against axial direction of the core is an important reference in determining the noding for the plant. In CAREM, the active length of the fuel rod is modeled using [[ ]] axial nodes based upon the [[ ]] of FLECHT-SEASET. Therefore, in case of [[ ]]. CSAU modeling in NUREG/CR-5149 assumed 15 axial core nodes representing a nodal length of 9.6 in. The NRC staff does not consider the small difference between the FLECHT-SEASET tests and the APR1400 CAREM significant. As a test temperature probe is quenched, it is indicative that the fluid or two-phase mixture level has reached or surpassed the elevation of the temperature probe. The NRC staff finds that the APR1400 CAREM utilizes the same nodalization as the test simulations and the simulations of the THTF and FLECHT-SEASET temperature response and quenching times shows that the nodalization is sufficient to capture the mixture level response properly.

The applicant also provided results showing the comparison of the Courant limit and the time step for the plant calculation. This confirmed that the time step used in the plant calculation is always lower than the Courant limit for CAREM.

Based upon the test simulations and predictions of quenching times, the NRC staff finds that the applicant's model follows the guidance of RG 1.157 that the model should be "capable of calculating the two-phase level in the reactor during a postulated accident." The NRC staff finds that the applicant's information regarding the adequacy of the nodalization for mixture level tracking is acceptable, and on the basis of that information concludes that this issue is resolved.

Lower Plenum Nodalization

According to NUREG/CR-5249 (see Figure 18), analysis results from the simulation of a PWR LBLOCA showed that the lower plenum below the downcomer skirt requires at least two nodes to adequately model the sweep-out effect, which carries an [[']] in the APR1400 PIRT (TR Table 3-2). It appears that the APR1400 nodalization models [[ ]]. As part of the January 2016 Audit Meeting, in AI-27, Part h [Ref. 3] the NRC staff requested the applicant to justify the selected nodalization for the lower plenum.

The applicant provided a figure in the response to AI-27h [Ref. 4] showing the liquid temperature and the saturation temperature in the lower plenum as an indication that the vapor generation from flashing is limited to a short period of time, less than 30 seconds, after the break. In addition, the applicant performed a sensitivity study in which it utilized two-fluid nodes to represent the lower plenum below the flow skirt [[ ]]. It compared the vapor generation rate in the lower plenum for the EM noding and the vapor generation rate in the lower plenum for the two-node sensitivity study and showed very small differences in the calculated results. It also noted similar observations in comparing the temperature of the hottest fuel rod. Since the sensitivity study demonstrated only very small differences between the results obtained for [[ ]] lower plenum model, the NRC staff finds the applicant's response acceptable and considers the issue of noding of the lower plenum resolved.

Break Nodalization

According to Section 3.5 of RG 1.157, the break location and ECCS injection point are areas of high fluid velocity and complex fluid flow, and contain phenomena that are often difficult to calculate. The results of these calculations are often highly dependent on the nodalization in the vicinity of such points. As part of the January 2016 Audit Meeting, in AI-27, Part i [Ref. 3], the NRC staff requested the applicant to justify the nodalization selected for the APR1400 broken cold leg and provide results from relevant sensitivity studies to assess the impact of based on the Marviken data.

In the TR, the applicant provided a schematic illustrating the Marviken test nodalization. To determine the break nozzle nodalization, the applicant performed sensitivity studies, varying the number of volumes in the supplied report [Ref. 50]. The applicant stated that the two-phase critical flow rates are [[ ]]. Therefore, the [[ ]] was adopted for the nozzle. Using this noding, it performed an evaluation and determined the best-estimate discharge coefficients of [[ ]].

As noted in [Ref. 31], the length to diameter ratio of the nozzle was limited within the range of [[ ]] to minimize the impact on the critical mass flux under subcooled conditions in the Marviken tests. The applicant stated that the Marviken test nozzle is represented by [[and an L/D ratio of each volume is [[ ]].

To assess the noding in the vicinity of the break, the applicant performed a sensitivity study varying the L/D ratio from [[ ]] (see Figure 2 in the response to AI-27 Part i, [Ref. 4]). The applicant compared the result of the study in terms of the cladding temperature for the PCT.

The applicant noted that the [[ ]]. The applicant concluded, however, that the noding sensitivity shows that [[ ]].

In addition, the applicant provided a limited sensitivity study to justify the nodalization of the APR1400 in the vicinity of the break. The sensitivity study effectively subdivided one node on the RCP discharge pipe near the break, thereby changing the L/D ratio for that subdivided node and demonstrated only a small effect on the PCT.

Following the guidance of RG 1.157, which in Section 3.5 states: "*Sufficient sensitivity studies should be performed on the noding and other important parameters to ensure that the calculations provide realistic results*"; the applicant performed sensitivity studies on the nodalization near the break and determined the appropriate EM representation by assessing the simulation noding in comparison to test results. Other important parameters, such as the L/D, were assessed. Finally, the applicant performed sensitivity studies with the APR1400 model to validate the nodalization near the break. The NRC staff finds that the applicant's test comparisons and sensitivity studies are sufficient to justify the noding near the break, and on the basis of that information concludes that this issue is resolved.

ECCS Injection Point Nodalization

RG 1.157 Section 3.5 discusses nodalization near the ECCS injection point and states that "…sufficient sensitivity studies should be performed on the noding and other important parameters to ensure that the calculations provided realistic results." TR Appendix E discusses tests performed in the ATLAS facility to study direct vessel injection (DVI) performance relative to emergency core cooling water bypass and downcomer boiling. In AI-27, Part j [Ref. 3], the NRC staff requested the applicant to describe or reference the sensitivity studies performed to assess the selected nodalization near the ECCS injection point using the ATLAS test results.

The applicant described and justified the existing nodalization based on the downcomer geometry and DVI configuration [Ref. 4]. Subsequently, the applicant described the results of sensitivity analyses performed to assess the nodalization near the ECCS injection point. For ATLAS Test 15, the nodalization was varied [[ ]] by up to an order of magnitude. The sensitivity results indicated small variations in the integrated break mass flow rate. The calculated ECC bypass ratio showed only a small variation. The PCT variation was about 15 K (27°F), which is not significant as defined by 10 CFR 50.46. Based upon the results of the sensitivity studies, the NRC staff determined that variations in the noding and other parameters, such as [[ ]] a small effect on the calculations for the key parameters of integrated break flow, bypass ratio, and PCT.

The NRC staff finds that the applicant's information regarding nodalization near the ECCS injection point follows the guidance in RG1.157 and is acceptable, and on the basis of that information concludes that this issue is resolved.

Pressurizer Surge Line

TR Section 4.2.1 does not provide the description for the pressurizer surge line nodalization. According to the RELAP5/MOD3.3/K base model input deck, the surge line is modeled using a single inclined hydraulic component. Additionally, the code computes the loss coefficient though the surge line pipe, which neglects the turns and orientation changes in the line. In AI-27 Part k [Ref. 3], the NRC staff requested the applicant to justify the selected nodalization and loss coefficients, including references to nodalization sensitivity studies comparing the pressurizer flow into the upper plenum against available SET and/or IET data.

In response [Ref. 5], the applicant provided information regarding the modeling of the pressurizer surge line in the RELAP5/MOD3.3/K model. The model was used to simulate the pressurizer level response for LOFT test and Figure 2 in response to AI-27, Part k [Ref. 5], showed the comparison of the pressurizer level for the test simulation and the test data. Based upon the adequacy of this simulation, the applicant justified the use of this model for APR1400 plant calculations.

The applicant also stated that the turns and orientation changes in the surge line are represented by loss coefficients in the branch connections. The applicant performed a sensitivity study to determine the effect of the variation on the pressurizer depletion time. In addition, the applicant performed a sensitivity study in which the surge line was represented by multiple nodes instead of a single inclined node. The results of the sensitivity study showed a small effect on the rate of depressurization, collapsed liquid level, and mass flow rate through the surge line. However, there was an insignificant effect on the PCT during blowdown, while the PCT during the reflood phase was slightly lower for the more detailed surge line model. The applicant concluded that the effect of the variation of the loss coefficient and modeling was not significant.

Based upon the results of the sensitivity studies and the conservatism in the APR1400 surge line model, the NRC staff finds that the applicant's information regarding the justification for the nodalization and loss coefficients in the pressurizer surge line is acceptable, and on the basis of that information concludes that this issue is resolved.

Upper Guide Support Structure

The nodalization of the upper guide support structure region in the original TR showed a difference between the RELAP5/MOD3.3/K nodalizations shown in TR Figure 4-1 and that in Figure 17 in TR Appendix B. Therefore, in AI-27, Part i, [Ref 55], the NRC staff requested the applicant to justify the nodalization that will be used for LBLOCA licensing calculations.

In response [Ref. 5], the applicant stated that they will utilize the nodalization of TR Figure 4-1 for all licensing calculations and that it is reasonable to use the same nodalization for all sensitivity studies. The Revision 1 of TR Appendix B utilizes the LBLOCA licensing nodalization in Figure 17 and new sensitivity results are presented.

The NRC staff finds that the applicant's changes in Revision 1 of TR Appendix B that utilizes the LBLOCA licensing nodalization in Figure 17 are acceptable, and therefore concludes that this issue is resolved.

Radial Nodalization of Reactor Core

The guidance in NUREG/CR-5249, Section 2.0, discusses issues related to model nodalization. TR Figure 17 of Appendix B shows the nodalization of the primary system where the entire core is divided into [[ ]]. The NRC staff concern is that this nodalization may overestimate the cross-bundle core flow [[ ]]. In AI-76 [Ref. 3] the NRC staff requested the applicant to justify core radial nodalization from this perspective. If a different nodalization is to be used, the NRC staff requested the applicant to justify why it will provide conservative or realistic predictions of the core heat transfer behavior.

In response to the AI-76 [Ref. 5], the applicant stated, [[ ]] Furthermore, it is stated "in CAREM, the conservative F_r (radial peaking factor) is applied to hot bundle." In the June 2016 Audit Meeting [Ref. 12], the NRC staff and the applicant discussed this in conjunction with the related topic in AI-27, Part e. The applicant provided a presentation [Ref. 27] showing the planar average power distribution a burnup of [[ ]].

The applicant's presentation indicated that [[ ]]. So, the same question of thermal channel cooling arises as in AI-27, Part e. In response to AI-27, Part e, the applicant performed a sensitivity study using [[ ]]. In the presentation, the applicant also addressed questions regarding the radial peaking factors raised in AI-54 [Ref. 3] which the NRC staff finds justifies the peaking factor choices. The applicant's cross-flow model is discussed further in Section 4.9 below where the cross-flow loss coefficient calculations and uncertainty treatment are also acceptable.

The NRC staff finds that the applicant's information justifying core radial nodalization is acceptable because a conservative radial peaking factor is used, the cross-flow representation is appropriate, and the applicant demonstrated by sensitivity study that the model is conservative. On the basis of that information, the staff concludes that this issue is resolved.

4.6 Breaks

Spectrum

The TR describes the methodology for a LBLOCA and the applicant uses a representative scenario (termed the limiting large break) to demonstrate the methodology. Although the representative scenario is a double-ended guillotine break of a cold leg, approval of the

methodology is sought for any break that is a LBLOCA. However, the TR does not define the type of breaks that the applicant regards as LBLOCA, and therefore fall within the scope of the EM.

RG 1.157 specifies that breaks should be evaluated to include the range from “full double-ended” to small breaks. Furthermore, the guidance specifies that longitudinal breaks should also be considered. RG 1.157 also states that actual peaking factors and fuel conditions should be used. These will establish the limiting conditions applicable to subsequent design activities. The discussion in TR Section 3.1 did not include these and other details regarding the selection of the limiting break. As a result, in AI-3 and AI-6 [Ref. 3], the NRC staff requested the applicant to provide additional information. In the response to AI-3 [Ref. 4], the applicant stated that the requested information may be found in the response for AI-6, Part a.

In the response to AI-6 Parts a, c, d, and e [Ref. 4], the applicant stated:

- CAREM is applicable to the following spectrum of breaks:
    - Break location: primary loop piping, such as the RCP discharge leg (cold leg), RCP suction leg, hot leg.
    - Break type: guillotine, double-ended and longitudinal split.
    - Break size: From 0.5 ft^2 to 100 percent double ended[4].

This addresses the information requested in AI-3 and resolves the concern. In addition, the issues associated with AI-6 Parts a, c, d, and e are addressed as follows:

- TR Appendix K, which the applicant added to the TR in Revision 1, determines the limiting break and burnup conditions as follows:
    - Break location sensitivity: To determine the limiting break location
    - Break type sensitivity: To determine the limiting break type for the limiting break location
    - Pressurizer Location sensitivity: To determine the limiting location of the pressurizer (intact loop or broken loop)
    - Break size and burnup sensitivity: To determine the limiting break size for the limiting break location and type and limiting burnup condition since the burnup may affect the limiting break size

The applicant found that the limiting break location was the cold leg, the limiting break type was a double ended guillotine break, the limiting size was 100 percent, and the limiting time in life occurred for a burnup of 27,000 MWd/MTU. Based upon the studies in the TR and TR Revision 1, the NRC staff is able to conclude that the limiting location and type of break is a double ended guillotine break in the cold leg at a burnup of 27,000 MWd/MTU. The break size

[4] 100 percent break means 100 percent of pipe cross sectional area. Therefore, 100 percent double-ended guillotine break has a total of 200 percent break area of a pipe.

spectrum of 100 percent, 80 percent, and 60 percent in the TR and TR Revision 1 shows the 100 percent size to be limiting.

The NRC finds that the responses to AI-6 address the break spectrum and are acceptable because they clearly establish the break location, type, and size range for CAREM, they followed the guidance of RG 1.157, and are based upon the commitment made by the applicant in those responses.

The NRC staff imposes CAREM methodology application Limitation (1):

> *The CAREM methodology is limited to LOCA applications where the minimum break size is greater than or equal to 0.5 ft^2.*

Guillotine vs. Slot

In TR Revision 1, Appendix K, Figure 4 depicts the slot break model used in the break type sensitivity study. In this model of the split break in the cold leg, a single break valve is connected to pipe component 496-01 with an area 200 percent of the pipe cross sectional area. In contrast, the double ended guillotine break has one break valve connected to pipe component 491 and another break valve component connected to pipe component 496-01, both with areas of 100 percent of the pipe cross sectional area. The split break model also uses the Ransom-Trapp critical flow model with discharge coefficients of [[ ]] for single and two-phase flow respectively, which is the same critical flow model used for the double-ended guillotine breaks. When compared to the guillotine break, the slot break was less limiting as stated in Figure 5 in TR Revision 1, Appendix K.

In response to AI-61, Parts b-f [Ref. 4] discussed in further detail below, the applicant provided more detailed information regarding the slot break, including different models for representing the slot break and various sizes of the slot break ranging from 20 percent through 200 percent. The slot break model B shown in Figure 3 of the response to AI-61, Parts b through f is used in the TR and TR Revision 1 studies. Model A shown in Figure 3 in the response to Parts b through f of AI-61, may be a more representative slot break model. However, Figure 4 in the response to Parts b through f of AI-61 shows that results of both slot break models A and B are well below the PCT response resulting from the DECLG case.

The NRC staff finds that the applicant's information demonstrating that the guillotine break is more limiting in terms of PCT is acceptable, and on the basis of that information concludes that this issue is resolved.

Limiting Location

In the response to AI-6 [Ref. 4], the applicant clarified that they did not perform any sensitivity calculations to arrive at the location of the break in the cold leg and stated:

> *"Break location sensitivity in the cold leg is not performed because there is no additional pressure loss coefficient in the cold leg. Therefore, it can be thought that there is no significant difference in break location in the cold leg (pump discharge line). Moreover, it is widely accepted that the break location near the vessel is more limiting because it has more possibility to have ECC bypass or sweep-out of coolant than the other break locations in the cold leg. [[ ]]"*

The NRC staff notes that there is a slight increase in loss coefficient as the break is moved away from the cold leg nozzle. This slight increase would tend to mitigate the effects of ECCS bypass and downcomer water sweep-out, which would be beneficial to the calculation of PCT. The NRC staff also determines that a break further away from the cold leg nozzle would result in a slight increase in the downcomer water level and the associated driving head which would also tend to enhance core reflooding and mitigate increases in the cladding temperature. The slightly higher water level results from the "river" effect as the water must travel to the break location to spill into containment. The further the break is from the cold leg nozzle, the higher the water level would be.

The applicant performed a sensitivity study to determine the limiting location for the pressurizer. In TR Revision 1, Appendix K, the applicant summarized the results of two sensitivity calculations. In one case, the cold leg break was in the loop that did not contain the pressurizer, and in the second case, the cold leg break was in the loop containing the pressurizer. The applicant determined that the PCTs during the blowdown were similar in both cases, but that the reflood PCT is higher and quenching times are longer when the cold leg break is in the loop that does not contain the pressurizer.

The NRC staff finds that the applicant's information that a cold leg break further from the nozzle is less limiting and that based upon the sensitivity study the reflood PCT is higher and the quenching time is longer when the cold-leg break is not in the same loop as the pressurizer is acceptable, and on the basis of that information concludes that this issue is resolved.
Limiting Break Size and Elevation

TR Section 5.2.1.1 discusses the process followed by the applicant to determine the limiting break size. TR Section 5.2.1.1, citing Figure 5-9, states that the applicant selected the 100 percent guillotine break of the cold leg as the limiting case because its blowdown PCT is the maximum as compared to 80 and 60 percent breaks. However, Figure 5-9 shows that the reflood PCT for the 80 percent break is nearly the same as the blowdown PCT for the 100 percent break.

In AI-61 [Ref. 3], the NRC staff requested the applicant to address several concerns as briefly restated in the following Parts:

a. Provide the calculated PCT and corresponding elevation during the blowdown period and the reflood period for the 100 percent and 80 percent guillotine break cases.

b. Provide the basis for not examining the 90 percent and 70 percent guillotine break cases.

c. Describe the phenomena resulting in the 80 percent guillotine break case quenching before the 100 percent guillotine break case.

d. Provide information to justify that the 180 percent, 160 percent, 140 percent, etc., cold leg split (CLS) break PCT will be bounded by the guillotine break.

e. Provide evidence that the reflood PCT for the 80 percent guillotine break resulting from the application of distribution free statistics is not limiting.

f. Justify that the calculations performed are representative of the spectrum of break sizes and locations applicable and that split breaks are bounded by the limiting break.

In the response to AI-61, Part a [Ref. 4], the applicant provided the calculated PCT, and its location, for the 100 percent and 80 percent guillotine break cases for blowdown and reflood. Based on this information, the NRC staff is now able to better understand the blowdown PCT and reflood PCT transient response.

In the response to AI-61 Part c [Ref. 4], the applicant stated that fuel rod quenching occurred at about [[ ]] seconds for the 60 percent break case, while for the other break cases, fuel rod quenching occurred at about [[ ]] seconds. Furthermore, the different quenching response was mainly as a result of the elevation at which the PCT occurred. For the 60 percent case, the PCT occurred in heat structure node [[ ]], while the other cases had the PCT in heat structure node [[ ]]. The applicant stated that the fuel rod quenching in the 60 percent break case occurs at a later time, [[ ]] s, because the [[ ]] for the 80 percent and 100 percent cases. In the response to AI-61, Part c [Ref. 4], the applicant presented Figure 7 showing the collapsed core level for the 80 percent and 100 percent cases and stated that the collapsed water level for the 80 percent case is lower than the 100 percent case due to the delay in the time of SIT injection.

[[ ]].

The NRC staff finds that the explanation for the quenching time is reasonable and therefore concludes that this part of the issue is resolved.

In the response to AI-61, Part d [Ref. 4], the applicant provided a break size sensitivity study for the split break cases. The applicant examined different models for representing the split break in RELAP5 and examined split break sizes ranging from [[ ]] percent. Based upon the split break sensitivity studies, the NRC staff concludes that the sensitivity study illustrates that the double ended guillotine break cases bound the split break cases of various sizes, and that the issues associated with AI-61, Parts d and f are resolved.

In the response to AI-61, Parts b, [Ref. 4], the applicant performed a SRS calculation utilizing the break area as one of the uncertainty parameters for a double ended cold leg guillotine break. The SRS calculations incorporated the break size as one of the uncertainty parameters assuming a uniform distribution of sizes from [[ ]] percent and Figure 6 of the response provided the calculation results. The third highest PCT of these SRS calculations resulted in a PCT of [[ ]] percent double ended cold line guillotine (DECLG) break. The third highest PCT from the SRS calculations in the TR was 1264 K. The third

highest PCT case in the TR (1264 K) occurred during blowdown, while the third highest of the SRS calculations which included break area as an uncertainty variable ([[ ]] K) occurred during reflood. Rod quenching for the TR case occurred later than for the SRS case with break area uncertainty. There were no other details of the study provided. The applicant concluded that the 100 percent DECLG break presented in the TR is in fact limiting as they had stated in the TR.

Appendix K in TR Revision 1 provides the results of sensitivity studies for the break location, break type, break size, and limiting time in life. The applicant found that the limiting break location was the cold leg, the limiting break type was a double-ended guillotine break, the limiting size was 100 percent, and the limiting time in life occurred for a burnup of 27,000 MWd/MTU. RG 1.157 states that "*the range of break sizes considered should be sufficiently broad that the system response as a function of break size is well enough defined so that interpolations between calculations, without considering unexpected behavior between break sizes, may be made confidently.*" Based upon the studies in the TR and TR Revision 1, the NRC staff is able to conclude that the limiting location and type of break is a double ended guillotine break in the cold leg at a burnup of 27,000 MWd/MTU. Based upon the studies performed in response to the audit issues, the NRC staff is able to conclude that the break size spectrum is sufficiently broad and that smaller break sizes are bounded by the spectrum analyzed. The break size spectrum of 100 percent, 80 percent, and 60 percent in the TR and TR Revision 1 shows that the 100 percent break size is limiting. The NRC staff finds that the applicant's information regarding the spectrum of break sizes, types, and locations meets the guidance of RG 1.157, and on the basis of that information concludes that this issue is resolved; however, the NRC staff imposes CAREM methodology application Condition (3):

> *The limiting break size shall be determined such that interpolation between break sizes would not result in a calculated peak cladding temperature that is significantly larger than 50° F, as specified by 10 CFR 50.46(a)(3)(i) for model changes.*

The NRC staff finds that the applicant's information supplied above to clarify the several issues regarding the determination of limiting LBLOCA breaks is acceptable, and on the basis of that information concludes that this issue is resolved, subject to Condition (3).

Long Term Core Cooling

The base case calculation of the APR1400 using the best-estimate operating conditions and nominal values is shown in TR Figures 5-12 through 5-28. The TR states that the water level in the reactor vessel is mainly determined by the balance of inlet (ECCS) and outlet (break) mass flow rates. In TR Revision 1, Figure 5-14 shows that during the late reflood period, ECCS water is supplied by the SIPs with a constant flow rate while the break flow, shown in Figure 5-13, decreases to almost zero beyond 225 seconds. From these results, one would expect continuous water level recovery in the reactor core region after 225 seconds. However, the water levels in the core and downcomer region, shown in Figure 5-15, are basically constant beyond 325 seconds. Therefore, in AI-62 [Ref. 3] the NRC staff requested the applicant to provide an explanation for the observed behavior. In TR Revision 1 [Ref. 13], the figures are extended to 400 seconds instead of being truncated at 250 seconds. The extended time scale in Figure 5-15 in TR Revision 1 shows that the downcomer and core collapsed water levels are stable and show a slow increasing trend, demonstrating that a continuous water level recovery results, which leads into the period of long term core cooling. Consequently, the transient behavior is more apparent and the previous NRC staff concerns associated with AI-62 are no longer present.

The NRC staff finds that the applicant's information supplied showing event progression out to 400 seconds is acceptable, and on the basis of that information concludes that this issue is resolved.

4.7 Thermal Conductivity

Thermal Conductivity Degradation

To address the impact of fuel pellet Thermal Conductivity Degradation (TCD), the applicant developed a procedure in which a penalty is applied to the fuel centerline temperature using results from the FATES-3B analysis. This method is described in TR Section 2.4.2 of Appendix K.

In applying this method, the applicant adjusts the heat structure inputs for the fuel in RELAP5/MOD3.3/K to match the interface data from the FATES-3B tests. This interface data includes the steady-state calculated fuel centerline temperatures (FCT) with an additional TCD penalty and the gap width. The applicant performs this calculation at the limiting burnup for several values of fuel rod linear heat generation rate (LHGR). Next, the applicant uses these calculations to develop functions that give fuel rod roughness as a function of LHGR, presented in Figure 27 and Figure 28 in TR Appendix K, for use in the 181 SRS calculations. Therefore, [[

]].

The NRC staff finds that the applicant's information on TCD describes an acceptable approach because it captures the best-estimate nature of the interface data while allowing for the sampling of thermal-conductivity during the SRS calculations, and on the basis of that information concludes that this issue is resolved.

Gap Conductance Model

TR Appendix B states that the applicant modified [[
]]. However, it was unclear what prompted this modification. In AI-24 [Ref. 3], the NRC staff requested the applicant to provide the basis for performing this modification.

In the response to AI-24 [Ref. 4], the applicant stated that [[

]].

The applicant demonstrated the effect of this modification to RELAP5/MOD3.3 using APR1400 plant simulations in TR Appendix B. The APR1400 simulations show the comparison of results before and after the applicant made this change. However, the applicant did not supply a comparison of results for either the APR1400 plant or an integral test using RELAP5/MOD3.3 and RELAP5/MOD3.3/K (i.e., with all changes present in the frozen version that is submitted for licensing). In AI-25 [Ref. 3], the NRC staff requested the applicant to provide the results of such a comparison.

In the response to AI-25 [Ref. 4], the applicant compared the cladding temperature for the high-power region using RELAP5/MOD3.3 and RELAP5/MOD3.3/K for LOFT Test L2-2. The applicant noted that the code inputs were the same except for the user input for [[ ]]. The applicant provided the calculation results, showing essentially no difference in the reflood PCTs, although the quenching time is delayed for both cases as compared to the test data. The applicant's comparison illustrated that the integral effect of this code modification is acceptable to the NRC staff.

The NRC staff finds that the applicant's information justifying the modification to return [[ ]] is acceptable, based on the results of the calculations provided by the applicant, and therefore concludes that this issue is resolved.

Transient Gap Conductance

TR Section 4.2.2.1.1 states that the uncertainty in the transient gap conductance is lumped with that of the [[ ]]" According to Volume 4, Section 9.3 of [Ref. 20], the gap conductance model in the code, derived from FRAP-T6, considers "...elastic deformation of cladding under the differential pressure..." A few lines later, the manual states that "...clad ballooning is not included in the [gap conductance] model." In AI-29, Parts a-f [Ref. 3], the NRC staff requested the applicant to address several concerns regarding the approach used to treat the uncertainty in transient gap conductance.

In the response to AI-29 [Ref. 4], the applicant described the RELAP5/MOD3.3/K dynamic gap conductance model which is a simplified deformation model generated from FRAP-T6. The clad ballooning model is the standard RELAP5 model. The response described how RELAP5/MOD3.3 captures the effect of clad ballooning on the transient gap conductance, but notes that the code cannot represent the effect of fuel pellet fragment relocation, so this effect is not calculated by RELAP5/MOD3.3/K, and thus is not considered in CAREM.

Accordingly, the NRC staff imposes CAREM methodology application Limitation (2):

> *The CAREM methodology may be used to predict the onset and potential for cladding deformation resulting from swelling, but may not be used in licensing calculations to predict flow blockage effects due to pellet relocation or hydraulic flow redistribution.*

The applicant stated that the geometry changes caused by ballooned cladding do not alter any other parameters which have any influence on the RELAP5/MOD3.3/K hydraulic solution. The applicant further stated that the effect of cladding swelling on gap conductance is reasonable and based upon the FRAP-T6 code model. The NRC staff considers the FRAP-T6 cladding deformation model reasonable and the model's effect on the geometric changes to the gap conductance are appropriate. Therefore the NRC staff finds the applicant's explanation of the effect of geometry changes on the hydraulic solution resulting from cladding swelling are acceptable within the constraints of Limitation (2).

The applicant stated that in CAREM, the steady state gap conductance data as calculated by the FATES-3B fuel performance code is used to set the initial condition of the fuel. The treatment of the initial gap conductance is discussed in Appendix K of TR Revision 1. The applicant stated that the gap width during the transient can change significantly compared with that of steady-state, and it can have a dominant effect on the transient gap conductance. The applicant noted that as discussed above, the gap width during transient is considered by [[

]] which is based upon FRAP-T6. The NRC staff considers the FRAP-T6 cladding deformation model reasonable and the explanation of the transient gap conductance acceptable because the deformation calculated by the cladding swelling model will directly govern the transient gap width. Therefore, the NRC staff finds this explanation acceptable within the constraints of Limitation (2).

The applicant described how cladding deformation can be divided into two categories: elastic and plastic deformation. The applicant stated that [[

]]. The applicant stated that the uncertainty of transient gap conductance is caused by the uncertainty in the steady-state gap width which is a dominant parameter in the transient gap conductance calculation. Consequently, the applicant concluded that the [[
]].

The NRC staff finds the explanation acceptable because [[

]], and therefore the NRC staff finds the explanation acceptable, and the issue is resolved.

The applicant stated that conservative gap conductance data was generated by [[
]], etc. Consequently, [[

]]. In the methodology to address the thermal conductivity degradation (TCD) issue, the [[
]]. The applicant does not directly account for [[

]].

The NRC staff accepts the applicant's assertions that the balloon/burst model in RELAP5/MOD3.3/K is acceptable for indicating the potential for ballooning or bursting. The NRC staff also agrees that the RELAP5 clad ballooning and burst models are not sufficient to actually determine the effect of such occurrences on the PCT. Therefore, CAREM may not be used in licensing calculations to predict flow blockage effects due to pellet relocation. Accordingly, the NRC staff imposes Limitation (2), which prohibits the use of CAREM for calculating the flow distribution and associated PCT following the predicted onset of clad ballooning.

4.8 Hydrodynamics

Multidimensional Effects

RG 1.157 states that "…one-dimensional approximations to three-dimensional phenomena will be considered if those approximations are properly justified." The core nodalization is [[ ]]. It is unclear how the methodology in the TR selects this nodalization over other options and how it can model multidimensional phenomena occurring within the core (e.g., "upper plenum to core CCW" in TR Table 3-2). Therefore, in AI-27, Part d [Ref. 3] the NRC staff requested the applicant to discuss any known multidimensional phenomena of importance to LBLOCA and justify the selected core radial nodalization.

In the response to AI-27, Part d [Ref. 5], the applicant stated that core nodalization of the APR1400 model and the SETs and IETs models also use [[ ]]. The response continues, stating that multidimensional flows can be broadly classified into two phenomena. First, higher vapor velocities and liquid entrainment occur in the higher power region of the core. The entrained liquid from the core is carried into the upper plenum where it is de-entrained forming a two-phase pool. The liquid from the pool can reenter the lower power region of the core due to the lower vapor velocities in those regions. [[ ]]. In CAREM, the applicant does not specifically model the multidimensional phenomena of the core, but it does consider the adverse conditions that may occur in the area around the hottest fuel rod. The applicant then stated that in the case of the APR1400 plant, multidimensional phenomena are not significant for the following reasons:

- [[ ]].
- [[ ]].
- [[ ]].

The applicant also considered the radial power profiles depicted in Figures 4.3-4 through 4.3-18 of the APR1400 Design Control Document (DCD) Tier 2 Chapter 4, Revision 1. The applicant compared the power in the lower power outer assemblies with the power in the central assemblies, neglecting one hot assembly. The comparison for the unrodded, full power equilibrium Xenon conditions indicated that the average power in the peripheral assemblies ranged from [[ ]] of the average power in the central assemblies.

In response to AI-47 [Ref. 5], the applicant discussed the CCFL condition based upon the Kutateladze number and stated that the Kutateladze number during the reflood period (70 seconds to 200 seconds) for AP1400 calculations was between 4.0 m/sec and 6.7 m/sec. The applicant also stated that the steam velocity between the core and upper plenum during the reflood period (70 seconds to 200 seconds) for the APR1400 RELAP5/MOD3.3/K calculation was approximately 40 m/sec. Consequently, the average core exit steam velocity would preclude any liquid flow from the upper plenum to the core.

In response to AI-60 [Ref. 4], the applicant provided liquid and vapor flow rates from the core to the upper plenum for the RELAP5/MOD3.3/K average assemblies and the hot assembly:

- Figure 17 shows the liquid mass flow rate [[ ]],
- Figure 18 shows the vapor mass flow rate [[ ]],
- Figure 19 shows the liquid mass flow rate [[ ]], and
- Figure 20 shows the vapor mass flow rate [[ ]].

The vapor mass flow rate during the reflood period (70 seconds – 200 seconds) shows upward flow into the upper plenum. However, [[ ]], the NRC staff was concerned that the calculated behavior does not seem to be realistic. Furthermore, the liquid back flow into [[ ]] which would be non-conservative behavior.

Figure 1 of the response to AI-27, Part d [Ref. 5], shows the mass flow rate at [[ ]].
To assess the effect of limiting flow back into [[ ]], the applicant performed a sensitivity calculation in which it assumed the reverse K-factor of the junction connecting [[ ]] exit and upper plenum to be a very large value. Figure 2 compares the original mass flow rate at [[ ]] exit with the case with the very large reverse K-factor. The very large reverse K-factor prevented flow back into [[ ]], similar to the effect of including a CCFL model, except that it precluded all downward flow into [[ ]], while a CCFL model may have calculated negative liquid flow if the conditions were appropriate. Figure 3 shows that the heat transfer mode at the PCT location was nearly identical and Figure 4 shows that the PCT response is very similar. This tends to imply that the unrealistic effect of downward flow at [[ ]] exit has little effect on the result, and therefore, a better model (i.e., use of a CCFL) is not required.

In response to RAI 399-8510, Question 9, Part 10 [Ref. 1.g], the applicant performed a multi-dimensional sensitivity calculation to confirm that the liquid fallback phenomenon from upper plenum to the core does not occur when lower power outer assemblies are represented in a separate channel in the RELAP5/MOD3.3/K model. [[ ]].

Figure 10-6 in the response to RAI 399-8510, Question 9, Part 10, [[ ]].

Based upon the figure for the cumulative liquid mass rate at the core to upper plenum, the applicant concluded [[

]].

In the responses cited above, the applicant concludes that [[

]]:

- the SET and IET assessments,
- liquid fallback from the upper plenum to the low-power region is not expected for the APR1400,
- the conservative power peaking factors for the hot fuel rod and [[ ]] (See SER Section 4.14).

The NRC staff finds that the applicant's information regarding how nodalization is selected over other options and how it can model multidimensional phenomena occurring within the core is properly justified and therefore acceptable, and on the basis of that information concludes that this issue is resolved.

Cross-Flow Junctions

Cross-flow junctions are used in the model between [[ ]]. The flow through these junctions in the core depends on the associated flow areas and loss coefficients. In AI-27, Part f [Ref. 3], the NRC staff requested the applicant to explain the approach used to determine these parameters and how the uncertainties are affected.

In the response to AI-27, Part f [Ref. 4], the applicant stated that in the areas of the core, cross-flow junctions are determined by the reactor core geometry and that they did not statistically treat the uncertainty associated with the flow area. They also did not statistically treat the uncertainty of the cross-flow junction loss coefficient because the effect of cross-flow is insignificant based upon information from [Ref. 51] which states:

> *"The power of the peripheral bundles is lower than the power of central bundles. This radial core power profile causes so called chimney effect (refer to Figure 1) As a result of this chimney effect, heat transfer to flow is enhanced in the high-power region and degraded in the low-power region. However, since PCT occurs in the high-power region, the net effect is a decrease in the PCT."*

The applicant also provided additional information related to the loss coefficients for cross-flow junctions. The NRC staff reviewed the applicant's methodology for the calculation of cross-flow in the response to AI-27, part f and found that the methodology conforms to a reasonable and common practice in RELAP5 calculations. The NRC staff is also aware of the relative insignificance of the cross-flow demonstrated in the 2D/3D Program Work Summary Report [Ref. 51]. The relative insignificance reduces the importance for specifying the uncertainty

related to cross-flow. The applicant has followed the guidance in RG 1.157, which recognizes that in the best-estimate analysis approach, not all models are required to be best-estimate with the associated uncertainty evaluation. RG 1.157 states:

> *In practice, best estimate codes may contain certain models that are simplified or contain conservatism to some degree. This conservatism may be introduced for the following reasons: (1) The model simplification or conservatism has little effect on the result, and therefore the development of a better model is not justified*.

The NRC staff finds that he applicant's methodology for calculating the cross-flow loss coefficients follows the guidelines of RG 1.157 Section 3.1.1, and since the effect of cross-flow is insignificant, it is not necessary to consider this as part of the uncertainty quantification, which is consistent with the guidance in RG 1.157 Part C, Section 1, and on this basis, it is concluded that the issue is resolved.

The NRC staff finds that the applicant's information explains how cross-flow junction parameters are determined and why an uncertainty assessment is unnecessary, and on the basis of that information concludes that this issue is resolved.

Counter Current Flow Limitation

CCFL on the tie plate is a complicated hydraulic phenomenon that consists of three flow streams:

- the steam flow up from the core,
- water up flow from the core, and
- the water fallback into the core.

RELAP5/MOD3.3/K treats CCFL with a general model that allows the user to select the Wallis form, the Kutateladze form, or a form in between the Wallis and Kutateladze forms. Experiments on CCFL are well correlated with the Kutateladze number, as shown in Figure 1 of the applicant's response to AI-47 [Ref. 5]. The applicant showed calculations of the Kutateladze number at pressures of 1 bar and 3 bar for fluid conditions at 100°C and obtained Kutateladze flooding limits of 4.0 m/sec at 3 bar and 6.7 m/sec at 1 bar.

The applicant showed that the steam velocity at the tie plate for APR1400 in Figure 2 of the response to AI-47 [Ref. 5] is always greater than 6.7 m/sec during the reflood period between 70 seconds and 200 seconds. The steam and liquid flow at the tie plate are shown in Figure 3 of the response to AI-47. The liquid flow is generally upward in the reflood period except the time period between 170 ~ 250 s, which is consistent with the limiting gas velocity. Therefore, the applicant concludes that the code prediction of thermal-hydraulic behavior on the tie plate is reasonable even without using the CCFL model.

The applicant also noted that RELAP5/MOD3.3/K implements CCFL by a change in the momentum equations (refer to Volume 1, Section 3.4.7 [Ref. 20]). The NRC staff observes that the applicant's CCFL models are validated mainly through steady state experiments, but the flow during the reflood period is very oscillatory. Consequently, CAREM does not use the CCFL option on the tie plate junctions.

The applicant's response also noted that RELAP5/MOD3.3/K in CAREM applications tends to over-predict the water carry-over from the core to the upper plenum. This is shown in the

assessments of FLECHT-SEASET [Ref. 42], Appendix P Evaluation of Steam Binding Using UPTF and PKL Test, pp. P-11) and NEPTUN [Ref. 42], Appendix C, Section 4.3.4). The applicant observes that the flow regime in the upper plenum during the reflood period is mainly annular or annular/mist and there are no models for the de-entrainment by internal structures in the upper plenum. Therefore, liquid flow from the upper plenum to the hot leg depends on the liquid volume fraction of the upper plenum and it is usually over predicted. Consequently, liquid volume remaining in the upper plenum is typically under predicted. This is evident in the applicant's UPTF-10B simulation [Ref. 42]. As a result, liquid flow to steam generator is excessive and the ensuing steam binding is too high. [[ ]].

In Summary, the applicant concluded that:

(1) the steam velocity at the tie plate is greater than CCFL limit velocity and there was no need to include the CCFL option on the tie plate junction in calculations;

(2) the RELAP5/MOD3.3/K over predicts the liquid carry-over rate from the core to the upper plenum, and this is a conservative approach with respect to PCT;

(3) [[ ]]; and

(4) [[ ]].

The NRC staff review of the radial power variation in the APR1400 core finds that the power in the low power peripheral assemblies could be as low as [[ ]] of the average assembly power. Since the steam flow from the assemblies is proportional to the assembly power, the NRC staff determines that the steam velocity exiting the low power peripheral assemblies would still exceed the Kutateladze flooding limit by at least a factor of 4. Since the applicant demonstrated that counter-current flow should be precluded at the core tie plate and since the applicant utilizes a conservative model for droplet vaporization in the steam generators, the steam binding is conservatively calculated because it maximizes the amount of vapor entering the steam generators. The bias applied for steam binding is therefore bounding because of the process for maximizing the vapor entering the steam generators. The NRC staff finds that the applicant's information regarding CCFL on the tie plate and the observed hydraulic phenomena calculated are acceptable because the steam velocities even in the low power assemblies exceeds the Kutateladze flooding limit by at least a factor of 4, and the steam binding bias is conservatively calculated. Therefore, on the basis of that information the NRC staff concludes that this issue is resolved.

4.9 Cladding Rupture

Section 3.2.5 of RG 1.157 states that "[f]or rods calculated to rupture their cladding during the loss-of-coolant accident, the oxidation of the inside of the cladding should be calculated in a best-estimate manner." In AI-30, Part b [Ref. 3], the NRC staff requested the applicant to discuss how this requirement is met in CAREM. In the response to AI-30, Part b [Ref. 4], the applicant stated that the reaction of zirconium and steam in RELAP5/MOD3.3/K code uses the correlation developed by Cathcart and described in Volume 1, Section 4.13 [Ref. 20]. The response also stated that the metal water reaction model is coupled with the fuel rod

deformation model so that if a fuel rod ruptures, the inside of the cladding can react with steam and the oxidation of the inside of the cladding is calculated.

RG 1.157 states: *"The data of Reference 11 are considered acceptable for calculating the rates of energy release, hydrogen generation and cladding oxidation for cladding temperatures greater than 1900°F."* Since Reference 11 in RG 1.157 is the Cathcart model, the applicant has followed the guidance set forth in RG 1.157, Section 3.2.5 for both the best-estimate model and the calculation of oxidation on the cladding inner surface in the event of a rupture of a fuel rod. The NRC staff finds acceptable the applicant's information that justifies the calculations of the fuel rods to rupture, including the oxidation of the inside of the cladding in a best-estimate manner, because it meets the guidance provided in RG 1.157 Section 3.5.2.

4.10 Loop Seal

Based on drawing 1-190-H-184-001C ("primary piping interfaces") provided in [Ref. 26], it appears that the loop seal elevation for APR1400 is close to the midpoint of the active core, creating the possibility of a loop seal during LOCA. As a result, in AI-2 [Ref. 3], the NRC staff requested the applicant to demonstrate that the peak cladding temperature remains within acceptable limits at all times. Furthermore, the applicant was requested to demonstrate the impact of having three (3) or four (4) trains of safety injection flow (as compared to the two trains assumed for the LBLOCA analysis in the TR) available during the long term reflood and LTCC phases under the above-described conditions.

RAI 143-8092, Question 15.06.05-1 [Ref. 8], is related to AI-2. In response to RAI 143-8092, Question 15.06.05-1 [Ref. 1.c], the applicant stated that the APR1400 pump design has an exit volute that is almost to the top of the cold leg, preventing backflow during blowdown or reflood. During the June 2016 Audit meeting [Ref. 12], the applicant stated that there would be no loop seal plugging, even when all trains of safety injection were available during a large break LOCA.

The staff reviewed the applicant's information and concluded that there is no potential for a loop seal plugging to occur. This is because the cold leg piping would need to be nearly liquid filled for fluid to flow back into the loop seal due to the RCP exit volute design, and a large break in the cold leg precludes the development of a significant liquid level. Therefore, the NRC staff, on this basis, concludes that this issue is resolved.

4.11 Loss of Offsite Power

TR Section 3.1.2.1 states that the reactor coolant pumps (RCPs) are assumed to be tripped upon loss of offsite power (LOOP). Since the operational RCPs may impact the peak cladding temperature as well as the coolant inventory, in AI-6, Part g [Ref. 3], the NRC staff requested the applicant to clarify whether the limiting break analysis included consideration of the location of the available safety injection pump trains with respect to the broken cold leg. In the response to AI-6, Part g [Ref. 4], the applicant provided a figure depicting the locations of the DVI lines relative to the cold legs and stated that two SIPs were assumed to be operating. The SIPs assumed to be operating were connected to DVI-2 and DVI-4, where DVI-4 is near the broken loop and DVI-2 is 180° opposite.

In the response to AI-6, Part g [Ref. 4], the applicant stated that:

- When the loss of offsite power (LOOP) is not assumed, [[

]].

- [[

]].

- The uncertainty ranges for [[

]].

- A figure provided as part of the response showed the calculated cladding temperatures with and without RCPs operating. The peak cladding temperatures (PCTs), with and without LOOP assumption, are [[ ]], respectively, and PCTs for both cases occur during the blowdown period. The reflood PCTs for both cases, with and without LOOP assumption, are [[ ]].

- The PCT with LOOP assumption occurred at heat structure node [[ ]], whereas PCT without LOOP assumption occurred at heat structure node [[ ]], which means the quenching time difference is caused by PCT elevation.

Therefore, the applicant concluded that the LOOP assumption is more conservative than the non-LOOP assumption. However, the applicant's finding indicated that [[

]]. Therefore, the NRC staff set forth Condition (4) on the use of CAREM requiring [[

]]. This condition is intended to apply when performing the scenario definition step (i.e., Step 1 of Element 1 of the CAREM iteration process).

The NRC staff finds that the applicant's information regarding the potential for the operational RCPs to impact the PCT as well as the coolant inventory for a LOOP coincident with the break is acceptable [[

]].

The NRC staff imposes CAREM methodology application Condition (4):

> *The limiting loss-of-offsite-power (LOOP) assumption shall be utilized and justified for the appropriate type of limiting PCT calculation, i.e., blowdown PCT or reflood PCT, while performing Step 1 of CAREM Element 1.*

4.12 Boiling in the Downcomer

The TR Revision 1 PIRT has [[
]]. During the June 2016 Audit Meeting, the applicant discussed this topic in a presentation to the NRC staff related to AI-17 [Ref. 1.a]. The applicant stated [[

]]:

"Downcomer level is sufficient to force coolant into lower plenum to begin refill."

The discussion stated that the appropriate calculations for heat transfer from structures to fluid and the potential for fluid vaporization were present in the calculation.

The NRC staff finds that the applicant's information regarding boiling in the downcomer during the refill period is acceptable because the heat transfer from the structures to the fluid is accounted for; and therefore concludes that this issue is resolved.

4.13 Flow Blockage

TR Appendix A, Revision 1, [[ ]]. Section 3.3.1 of RG 1.157 states that the calculation of the swelling and rupture of the cladding should be included in the analysis and should be performed in a best-estimate manner. The blockage may reduce the cladding temperature but that does not decrease the importance of the effect of blockage on the physical response. The calculated cladding temperature transient should reflect the cooling effects of clad swelling. In AI-72 [Ref. 3] the NRC staff requested the applicant to confirm that RELAP5/MOD3.3/K can calculate fuel cladding swelling and rupture. Furthermore, the NRC staff also requested the applicant to confirm that the calculated effect of swelling and rupture on fluid flow and cladding temperature calculations reflects the effect of fuel channel blockage.

In response [Ref. 5], the applicant stated that the RELAP5/MOD3.3 computer code has a model for cladding deformation described in Volume 1, Section 4.14 [Ref. 20]. The applicant incorporated an empirical cladding deformation model from FRAP-T6 [Ref. 41] into RELAP5/MOD3.3/K. This model estimates cladding strain by thermal, elastic, and plastic deformation and predicts rupture strain and blockage from NUREG-0630 [Ref. 33] data.

The applicant stated that RELAP5/MOD3.3 can calculate swelling and rupture effects based upon the FRAP-T6 [Ref. 41] code model. Further details are in the response to AI-29, Parts a through f [Ref. 4]. The calculation of gap conductance considers the effect of swelling, but does not directly calculate the effect of swelling on fluid flow blockage unless rupture occurs. When fuel rod burst is calculated to occur, the effect of flow channel blockage may be represented through additional loss coefficients in the neighboring junctions.

The NRC staff finds the applicant's clarifications on the impact of cladding deformation and rupture on flow blockage acceptable because it meets the guidelines set forth in RG 1.157

Section 3.11 and therefore concludes that this issue is resolved within the constraints of Limitation (2) cited above.

4.14 Power

Peaking Factors F_r/F_q

The guidance in RG 1.157, Section 4.3.1, establishes acceptable controls for the utilization of conservative parameters in best-estimate analysis. In AI-53, Part a [Ref. 3] the NRC staff requested the applicant to describe the determination of the [[ ]] in the LBLOCA analysis.

In the response to AI-53, Part a [Ref. 4], the applicant stated that [[ ]]. The applicant stated that the principle for CAREM is to use a value of the radial peaking factor which is bounding for the cycle-specific analysis.

The response also stated that [[ ]] using this methodology using sensitivity studies that consider the radial fall-off curve.

In AI-53, Part b [Ref. 3], the NRC staff requested the applicant to describe the process used to determine the power shape and peaking factor (F_q) for the base case plant calculation.

In the response to AI-53 Part b [Ref. 5], the applicant provided a [[ ]].

The applicant also stated that the LHGR limit is defined as the product of F_q and the core average LHGR. [[ ]].

The applicant stated [[ ]]. For the base case calculation, the power shape used is shown in Figure 7 of the response to AI-53, Part b.

Revision 1 of TR Section 5.1.1 reflects the clarification of the hot assembly power peaking factor as:

"[[

]]"

In response to AI-54 [Ref. 5], the applicant also provided the maximum peaking factors for the equilibrium cycles. The applicant stated that [[

]].

The NRC staff finds that the applicant's information justifying the use of peaking factors in a suitably conservative manner is acceptable, and on the basis of that information concludes that this issue is resolved.

Power Distributions

RG 1.157 states that individual best-estimate models should be compared to applicable experimental data to ensure that realistic behavior is predicted and that relevant experimental variables are included. It further states that uncertainty analyses should ensure that a major bias does not exist in the models and that the model uncertainty is small enough to provide a realistic estimate of the effect of important experimental variables. The analyses in the TR utilized a top-skewed power distribution. The applicant performed a significant number of comparison calculations to the FLECHT-SEASET test data that utilized a cosine power distribution. However, since the licensing calculation is based upon a top-skewed power distribution, the NRC staff noticed that the applicant did not consider the FLECHT-SESET top skewed power tests. In AI-80 [Ref. 3], the NRC staff requested the applicant to provide the basis for this observation.

In the June 2016 Audit Meeting, the applicant presented information regarding the FLECHT-SEASET skewed power distribution tests. The applicant stated that the skewed power shape in the FLECHT-SEASET skewed power tests is not representative of the top skewed power distribution used in the APR1400 CAREM analyses and that the FLECHT-SEASET cosine power distribution test provides a power distribution that is more representative of the expected APR1400 power distribution. The applicant discussed the integrated power from the bottom of the core to the PCT location in conjunction with the effect of power distribution on the flooding rate, fuel rod quenching and entrainment effects.

[[

]].

The NRC staff finds that the applicant's information regarding axial power distributions including comparisons to test data is acceptable, and on the basis of that information concludes that this issue is resolved.

Figure 4-1 Comparison of FLECHT cosine, skewed, with CAREM axial power shape

4.15 Cladding Oxidation

The Cathcart model is used in RELAP5/MOD3.3 and RELAP5/MOD3.3/K to calculate cladding oxidation. TR Section 4.2.2.1.1 provides the uncertainty for the Cathcart model and page 4 of 23 [Ref. 26] describes the input for the uncertainty in the cladding oxidation reaction. The description implies that [[ ]]. However, the acceptance criterion applies to core-wide cladding oxidation and, in addition, RG 1.157 does not distinguish between oxidation in different fuel rods for best-estimate analysis. Therefore, in AI-30 [Ref. 3], the NRC staff requested the applicant to justify the basis for consideration of the uncertainty in cladding oxidation uncertainty [[ ]].

In the response to AI-30 [Ref. 4], the applicant stated that the core-wide cladding oxidation is related to the maximum hydrogen generation of 10 CFR 50.46(b). In CAREM, [[

]]. In the discussion of the reporting of the third highest PCT from the SRS as the licensing basis in AI-51, Part c [Ref. 3], the applicant clarified that it would determine the core wide hydrogen generation from the highest of all the SRS

calculations, not the third highest as in the case of PCT, to provide additional conservatism. In a revised response to AI-30, Part a [Ref. 4], the applicant provided Figure 1 showing the results of calculated hydrogen generation associated with [[ ]] for the results of the SRS calculations. Since the SRS cases include the range of PCTs including those for analyses with reflood peaks, the CAREM methodology is conservative.

The NRC staff finds that the applicant's justifications that the basis for consideration of the uncertainty in cladding oxidation uncertainty [[ ]] is acceptable, and on the basis of that information concludes that this issue is resolved.

5. CONDITIONS AND LIMITATIONS

The NRC staff evaluation of the applicant's CAREM large break LOCA analysis methodology finds that TR Revision 1, the underlying methodology, and the provided EM are acceptable to support APR1400 applications subject to the following Conditions:

(1) Deleted.

(2) Changes to the downcomer modeling shall require additional justification and NRC approval.

(3) The limiting break size shall be determined such that interpolation between break sizes would not result in a calculated peak cladding temperature that is significantly larger than 50° F, as specified by 10 CFR 50.46(a)(3)(i) for model changes.

(4) The limiting loss-of-offsite-power (LOOP) assumption shall be utilized and justified for the appropriate type of limiting peak cladding temperature (PCT) calculation, i.e., a blowdown PCT or a reflood PCT, while performing Step 1 of CAREM Element 1.

The NRC staff evaluation of the applicant's CAREM large break LOCA analysis methodology finds that TR Revision 1, the underlying methodology, and the provided EM are acceptable to support APR1400 applications subject to the following Limitations:

(1) The CAREM methodology is limited to LOCA applications where the minimum break size is greater than or equal to 0.5 ft^2.

(2) The CAREM methodology may be used to predict the onset and potential for cladding deformation resulting from swelling, but may not be used in licensing calculations to predict flow blockage effects due to pellet relocation or hydraulic flow redistribution.

6. CONCLUSIONS

TR Revision 1 presents the applicant's realistic, or best-estimate, LBLOCA analysis methodology for APR1400. A summary of the Application is provided in Section 3 of this SER.

Many issues were discussed during the NRC Audit Meetings and are described in the NRC Audit Summary Report [Ref. 12]. The review of the TR resulted in a number of AIs. A limited number of the issues could not be completely resolved during the Audit Meetings, which resulted in the NRC staff generating RAIs. The applicant responded to all of the AIs and RAIs and all of these were evaluated and dispositioned by the NRC staff. Those that contribute to the statements of acceptability issued by the NRC staff are documented in Section 4 of this SER.

TR Revision 1 complies with the regulatory criteria in 10 CFR 50.46 and the guidance in the SRP, Sections 15.0.2, Revision 0 and 15.6.5, Revision 3, RG 1.203, RG 1.157 and the CSAU methodology documented in NUREG/CR-5249. The review of the TR and the evaluation of the responses to AIs and RAIs resulted in several important technical observations that are as follows:

- The treatment of TCD is addressed in TR Revision 1.

- Liquid flow response from the core to upper plenum during reflood may be calculated in an unrealistic manner in the APR1400 RELAP5/MOD3.3K model. The applicant has provided information and calculations justifying that liquid penetration back into the core is prohibited due to the magnitude of core exit steam velocities. However, the APR1400 RELAP5/MOD3.3K calculations indicate significant liquid penetration back into the core from the upper plenum. Thus, calculations using a more detailed representation of the core and upper plenum may result in lower calculated PCTs.

The NRC staff concludes that the KHNP LBLOCA methodology, CAREM, as documented in TR Revision 1 is acceptable for analysis of the APR1400 LBLOCA subject to the Conditions and Limitations stated in Section 5 of this SER.

7. REFERENCES

1. APR1400-F-A-TR-12004-P Revision 0, "Realistic Evaluation Methodology for Large-Break LOCA of the APR1400," December 2012, ADAMS Accession No. ML130230128.

2. APR1400-F-A-NR-14003-P, "Post-LOCA Long Term Cooling Evaluation Model," Revision 0, ADAMS Accession No. ML15012A019 (Proprietary)/ ML15009A201 (Non-Proprietary).

3. APR1400 Large Break Loss of Coolant Accident Audit Issues, ADAMS Accession No. ML18011A393.

4. Response to Audit Issues, Topical Report APR1400-F-A-TR-12004, "Realistic Evaluation Methodology for Large-Break LOCA of the APR1400," August 8, 2016 (ADAMS Accession No. ML16223A158).

5. Response to Audit Issues, Topical Report APR1400-F-A-TR-12004-P (Realistic Evaluation Methodology for Large-Break LOCA of the APR1400," December 20, 2016 (ADAMS Accession No. ML17005A007).

6. Response to Audit Issues, Topical Report APR1400-F-A-TR-12004-P (Realistic Evaluation Methodology for Large-Break LOCA of the APR1400," December 11, 2017, ADAMS Accession No. ML17345A958.

7. APR1400 Topical Report RAI No. 1-7425, Questions APR1400-1 through APR1400-3, ADAMS Accession No. ML14100A671, April 10, 2014.

8. APR1400 DCD RAI No. 143-8092, ADAMS Accession No. ML15221A006, August 7, 2015.

9. APR1400 DCD RAI 399-8510, ADAMS Accession No. ML16034A056, February 3, 2016.

10. APR1400 Topical Report RAI 7-8567, Questions APR1400-4 through APR1400-9, ADAMS Accession No. ML16098A098, April 7, 2016.

11. Korea Hydro & Nuclear Power Co., Ltd - Transmittal of Response to:

    a. RAI 1-7425, ADAMS Accession No. ML14113A325, April 10, 2014RAI 1-7425, ADAMS Accession No. ML14113A325, April 10, 2014

    b. RAI 143-8092, ADAMS Accession No. ML15246A531, September 3, 2015

    c. RAI 143-8092 Supplemental Response, ADAMS Accession No. ML16363A031, December 27, 2016

    d. RAI 399-8510, Question 6, ADAMS Accession No. ML16063A045, March 3, 2016

    e. RAI 399-8510, Question 7, ADAMS Accession No. ML17223A687, August 11, 2017

f. RAI 399-8510, Question 8, ADAMS Accession No. ML16063A045, March 3, 2016

g. RAI 399-8510, Question 9, ADAMS Accession No. ML16123A206, May 2, 2016

h. RAI 7-8567, 30-day response Questions APR1400 – 6, 7, 8, and 9, ADAMS Accession No. ML16116A446, April 25, 2016

i. RAI 7-8567, 45-day response Questions APR1400-4 and APR1400-5, ADAMS Accession No. ML16153A454, June 1, 2016

12. Audit Summary for APR1400 LBLOCA Topical Audit, ADAMS Accession No. ML17256B126, December 6, 2017.

13. Korea Hydro & Nuclear Power Co., Ltd - Transmittal of Topical Report APR1400-F-A-TR-12004-P, Revision 1, "Realistic Evaluation Methodology for Large-Break LOCA of the APR1400," Dated August 2017, ADAMS Accession No. ML17240A229 (Proprietary)/ ML17250A233 (Non-proprietary).

14. NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Compiled 2007.

15. 10 CFR Part 21, "Reporting of Defects and Noncompliances," 42 FR 28893, June 6, 1977.

16. 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," 68 Federal Register 54142, September 16, 2003.

17. Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989.

18. Regulatory Guide 1.203, "Transient and Accident Analysis Methods," December 2005.

19. NUREG/CR-5249, "Quantifying Reactor Safety Margins," December 1989.

20. RELAP5 MOD3.3 supporting documents, (ADAMS Accession No. ML110330203):

a. RELAP5/MOD3.3 Code Manual Volume I: Code Structure, System Models, and Solution Methods, NUREG/CR-5535/Rev P4-Vol I.

b. RELAP5/MOD3.3 Code Manual Volume II: User's Guide and Input Requirements, NUREG/CR-5535/Rev P4-Vol II.

c. RELAP5/MOD3.3 Code Manual Volume II: Appendix A Input Requirements, NUREG/CR-5535/Rev P4-Vol II App. A.

d. RELAP5/MOD3.3 Code Manual Volume III: Developmental Assessment Problems, NUREG/CR-5535/Rev P4-Vol III.

e. RELAP3.3 MOD3.3 Code Manual Volume IV: Models and Correlations, NUREG/CR-5535/Rev P4-Vol IV.

f. RELAP5/MOD3.3 Code Manual Volume V: User's Guidelines, NUREG/CR-5535/Rev P4-Vol V.

g. RELAP5/MOD3 Code Manual Volume 6: Validation of Numerical Techniques in RELAP5/MOD3.0, NUREG/CR-5535/Rev P3-Vol VI.

h. RELAP5/MOD3.3 Code Manual Volume VII: Summaries and Reviews of Independent Code Assessment Reports, NUREG/CR-5535/Rev P4-Vol VII.

i. RELAP5/MOD3.3 Code Manual Volume VIII: Programmers Manual,

j. NUREG/CR-5535/Rev P4-Vol VIII.

21. "CONTEMPT4/MOD5, A Multicompartment Containment System Analysis Program," NUREG/CR-4001, BNL-NUREG 51894, Brookhaven National Laboratory, 1984.

22. R.A. Dimmena, e. al., "RELAP5/MOD2 Models and Correlations," NUREG/CR-5194, EGG-2561-R4, 1988.

23. RELAP5/MOD3 Code Manual Vol.4, Models and Correlations (DRAFT)," NUREG/CR-5535, EGG-2596, 1990.06.

24. PNNL-19418, "FRAPCON-4.0: A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup," Vol.1, Rev.2, September 2015.

25. "KCE-1 Critical Heat Flux Correlation for Plus7 Thermal Design," APR1400-F-C-TR-12002-NP, Rev.0, KHNP/KNF, November 2012, ADAMS Accession No. ML14114A564.

26. Korea Hydro and Nuclear Power, Ltd., "KHNP Responses to Request for Additional Information No. 7425," APR1400-F-A-RA-14001-P, ADAMS Accession No. ML14112A432 (Proprietary)/ML14112A429 (Non-proprietary).

27. KHNP Presentation on CAREM, ADAMS Accession No.ML17228A819.

28. "Fluidic Device Design for the APR1400," KHNP APR1400-Z-M-TR-12003-P, Rev.0, December 31, 2012, ADAMS Accession No. ML130180120.

29. NUREG/CR-7024, "Material Property Correlations: Comparisons Between FRAPCON-3.4, FRAPTRAN 1.4, and MATPRO," 2011.

30. "Modification of LBLOCA Best-Estimate Evaluation Methodology of APR1400 Type 3 Nuclear Power Plants (2012)," KNF-TR-SGA-12005, KNF, September 2012.

31. EPRI-NP 2192 Project 14385-1, "Critical Flow Data Review and Analysis," S. Levy, Inc. January 1982.

32. NUREG/CR-0497, "MATPRO-Version 11(Rev. 1) A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior," 1980.

33. D.A. Powers, R.O. Meyer, "Cladding Swelling and Rupture Models for LOCA Analysis," NUREG-0630, U.S.NRC, March 1980.

34. Development of PCT Uncertainty Quantification Methodology Assessment of Separate Model and Construction of Thermal-Hydraulic Data Banks for Establishment of the Korean ECCS Evaluation Model," KINS/GR-011, 1990.12.

35. "Compendium of ECCS Research for Realistic LOCA Analysis," NUREG-1230, R4, December 1988.

36. "COBRA/TRAC - A Thermal-Hydraulics Code for Transient Analysis of Nuclear Reactor Vessels and Primary Coolant System, Volume 1, Revision 4," NUREG/CR-3046, PNL-4385, March 1983.

37. A. Guba, M. Makai, and L. Pál, "Statistical Aspects of Best Estimate Method-I," Reliability Engineering and System Safety, Vol. 80, Issue 3, pp.217-232, June 2003.

38. Korea Hydro and Nuclear Power, Ltd., "CAREM for LBLOCA Analysis of APR1400," APR1400-F-A-EC-13012- P, ADAMS Accession No. ML13239A464 (Proprietary)/ ML13239A343 (Non-Proprietary).

39. APR1400-F-A-NR-14002-P, Rev.0, "The Effect of Thermal Conductivity Degradation on APR1400 Design and Safety Analysis," September 2014.

40. H. Glaser, "BEMUSE Phase VI Report, Status report on the areas, classification of the methods, conclusions and recommendations," NEA/CSNI/R(2011)4, 28-Mar-2011; (https://www.oecd-nea.org/nsd/docs/2011/csni-r2011-4.pdf).

41. L.J. Siefken, C. M. Allison, M.P. Bohn, and S.O. Peck, FRAP-T6: A Computer Code for the Transient Analysis of Oxide Fuel Rods, EGG-CDAD-5410, Idaho National Engineering Laboratory, April 1981.

42. TR-33-002, 2002, "Topical Report for the Realistic Evaluation of Emergency Core Cooling System."

43. TR-KHNP-0018 (Proprietary), "Topical Report for LBLOCA Best Estimate Evaluation Methodology of APR1400 Type Nuclear Power Plants," 2010.8.

44. TR-KHNP-0008 (Proprietary), "Topical Report for Realistic Evaluation of Emergency Core Cooling System, Volume 1: Description of Best Estimate Methodology for Large Break LOCA," 2002.12.

45. G.D. Wyss, K.H. Jorgensen, "A User's Guide to LHS: Sandia's Latin Hypercube Sampling Software," SAND98-0210, February 1998.

46. KEPCO, KHNP, "APR1400 Design Control Document Tier 2," APR1400-K-FS-14002-NP, Rev.0, December 2014.

47. TR-KHNP-002, Topical Report "The best evaluation methodology for the Emergency Core Cooling System," KEPRI/KHNP, Dec. 2002.

48. M.J. Loftus, et al., "PWR FLECHT SEASET Unblocked Bundle Forced and Gravity Reflood Task Data Report," EPRI-NP-1459, NUREG/CR-1532, WCAP-9699, 1981.

49. APR1400-F-C-NR-12001-P, Rev.0, “Thermal Design Methodology,” July 2012.

50. KAERI/TR-437/94, “Uncertainty Quantification of RELAP5/MOD3/KAERI Critical Flow Model using Marviken Experimental Data,” S. Y. Lee, Oct. 1994.

51. NUREG/IA-0126, “2D/3D Program Work Summary Report,” June 1993.

52. “Phenomena Identification and Ranking Tabulation Korean Next Generation Reactor Large Break loss of Coolant Accident,” KINS/INEEL, 2001.

53. NRC Inspection of Korea Hydro & Nuclear Power Co., LTD, IR 05200046/2016201, April 8, 2016, ADAMS Accession No. ML16081A081

54. APR1400-K-Q-TR-11005-NP, Revision 5, “KHNP Quality Assurance Program Description (QAPD) for the APR1400 Design Certification,” April 2016, ADAMS Accession No. ML16123A406.

55. American Society of Mechanical Engineers (ASME) NQA-1-2008, “Quality Assurance Requirements for Nuclear Facility Applications,” New York, NY, March 2008.

56. Regulatory Guide 1.28, Revision 5, “Quality Assurance Program Criteria (Design and Construction), October 2017.

57. Submittal of Additional Information for the Condition (1) Regarding the APR1400 DCA – Safety Evaluation with No Open Items for Topical Report APR1400-F-A-TR-12004, April 25, 2018, ADAMS Accession No. ML18115A426.