U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

FINAL SAFETY EVALUATION FOR TOPICAL REPORT

WCAP-17203-P/WCAP-17203-NP, REVISION 0-2,

"FAST TRANSIENT AND ATWS METHODOLOGY"

WESTINGHOUSE ELECTRIC COMPANY

DOCKET NO. 99902038

1.0 INTRODUCTION AND BACKGOUND

1.1 INTRODUCTION

By letter dated June 30, 2010 (Reference 1), South Texas Project Nuclear Operating Company (STPNOC or STP) submitted Topical Report (TR) WCAP-17203-P, Revision 0, "Fast Transient and ATWS Methodology," for U. S. Nuclear Regulatory Commission (NRC) review and approval. The TR submittal provided a methodology for analyzing both limiting and non-limiting fast transients in initial and reload cores for currently operating boiling water reactors (BWRs) and advanced boiling-water reactors (ABWRs). As a result of the NRC staff review, several requests for additional information (RAIs) were issued. The applicant incorporated issues resolved through the RAI process into a revised version of WCAP-17203-P submitted by letter dated October 20, 2014, Nuclear Innovation North America LLC (NINA) submitted WCAP-17203-P, Revision 0-2, "Fast Transient and ATWS Methodology," for the NRC staff review and approval (Reference 2). By letter dated May 6, 2015, the NRC staff issued revised acceptance for review of WCAP-17203-P/WCAP-17203-NP, Revision 0-2, "Fast Transient and ATWS Methodology" (ADAMS Accession No. ML15111A272).

The purpose of the TR is to augment the existing methodology for fast transients and ATWS described in TR CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel" (Reference 3). While the methodology is applicable for the analysis of anticipated operational occurrences (AOOs) including ATWS for BWR product lines 2-6 (BWR/2 through BWR/6), it does so within the context of reload analysis and not first cores. The objective of this TR is to establish a methodology for analyzing both limiting and non-limiting fast transients in initial and reload cores for currently operating BWRs and the ABWR design. However, this safety evaluation (SE) concentrates on the application of this TR only on BWR/2 through BWR/6 plants.

This TR describes the methodology of the evaluation model (EM) as stipulated in the NRC guidance (Reference 4) for non-limiting and limiting AOOs including ATWS and Infrequent Events (IE) for BWR/2 through BWR/6 plants. An AOO is defined as a condition of normal operation that is expected to occur one or more times during the life of the nuclear power unit (Reference 5). Since ATWS events are considered AOOs that are followed by a failure of the scram protection system, they are addressed separately in the methodology. Licensing basis documents for individual plants will define categorization of each condition, AOO, IE, or postulated accident.

NUREG-0800, “Standard Review Plan” (SRP), Chapter 15 events are regrouped according to the acceptance criteria for AOOs including ATWS events as outlined in Chapter 4 and Section 15.8 of the SRP and a Phenomena Identification and Ranking Table (PIRT) which defines the phenomena that have to be addressed when evaluating the operating limits verified by Westinghouse methodology experts. Ranking of the phenomena is on a high/medium/low scale based on its influence on the figures-of-merit (FOM) defined in the acceptance criteria. FOM are those quantitative standards of acceptance that are used to define acceptable answers for a safety analysis. Once the operating limits and safety margins to acceptance criteria are determined for fast transients and ATWS, uncertainty analysis is conducted to evaluate the impact of uncertainties and biases on these limits in order to account for the uncertainty in the best-estimate result.

Westinghouse specifically requests the NRC review and approval of:

- The ranking designations in the PIRT
- The analysis methodology for evaluating fast transients
- The Monte Carlo-based uncertainty analysis methods

Even though the submitted TR is applicable to BWR/2 through BWR/6 and ABWR, the Office of Nuclear Reactor Regulation (NRR) review focused on the aspects of the TR that are pertinent to the consideration of future submittals related to fuel amendments regarding the BWR/2 through BWR/6 design. Therefore, this SE addresses the application of the methodology only to the BWR/2 through BWR/6 design.

This SE provides details of the review results of TR WCAP-17203-P/WCAP-17203-NP, Revision 0-2. The scope of the review is discussed in Section 1.0. The regulatory evaluation of the review is discussed in Section 2.0. Section 3.0 describes the NRC staff’s technical evaluation of the TR, including the discussion of the responses to the RAIs. The conditions, limitations, and the applicant’s commitments resulting from the review are included in Section 4.0. Section 5.0 lists the NRC staff’s conclusions and Section 6.0 lists all references cited in the SE.

1.2 BACKGROUND

Prior to this TR, Westinghouse used the NRC licensed methodology for fast transient analyses that is described in Reference 3. Though this methodology is applicable for the analysis of fast and slow transients for BWR/2 through BWR/6 reload analysis, it does not address all the transients required for first (initial) core applications. The objective of this TR is to establish a methodology for analyzing both limiting and non-limiting fast transients in initial and reload cores for currently operating BWRs as required by the initial safety analysis report (SAR).

According to Reference 3, ABB/CE/Westinghouse has differentiated between fast and slow transients as follows:

> These events are grouped into fast and slow transients based on the dynamic characteristics of the transient. "Fast transients" are those events of relatively short duration such that the impact of the spatial and temporal dynamics on the system nuclear and thermal-hydraulics is important to the overall plant response. "Slow transients" are defined as those transients for which the dynamic changes during the transient are sufficiently slow that the assumption that steady state conditions are achieved at each time step is either realistic or conservative.

WCAP-17203-P/WCAP-17203-NP, Revision 0-2, TR methodology is based on the NRC guidance for transients and accidents, Regulatory Guide (RG) 1.203, "Transients and Accident Analysis Methods," for AOOs, ATWS, and IE. Since ATWS events are considered AOOs that are followed by a failure of the reactor protection system (RPS) to scram, they are discussed separately in the TR. The IE or postulated accidents are defined in plant-specific documentation. Therefore, the specific categorization of IE and postulated accidents is not included in this methodology.

The PIRT provided in the TR defines the phenomena that have to be addressed when evaluating the operating limits and safety margins to acceptance criteria. Each identified phenomena is assigned an importance ranking corresponding to its influence on a FOM. The PIRT also provides the rationale for each ranking. Separate importance rankings are defined for different AOOs and ATWS events. Using the PIRT table as input the analysis methodology describes the evaluation process for determining the operating limits and the safety margins to the acceptance criteria.

The TR proposes a transient analysis statistical methodology that accounts for uncertainties and biases in the models, inputs, and parameters to ensure that operating limits and safety margins meet the required acceptance criteria. In order to do this, the TR proposes a statistical analysis method using 95 percent probability with a 95 percent confidence level (95/95) for calculating uncertainties associated with operating limits and safety margins.

This TR describes the methodology part of the EM as defined in the RG 1.203 and the Chapter 15 of SRP.

The SRP defines six areas of review for AOOs and ATWS analyses methods:

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

Each of these areas are briefly discussed below:

Documentation

The SRP guidance on documentation requires that the EM documentation must be scrutable, complete, unambiguous, accurate, and reasonably self-contained. The documentation must be consistent nomenclature and must be used throughout the entire model documentation. Also the code documentation must be sufficiently detailed that a qualified engineer can understand the documentation without recourse to the originator as required of any design calculation that meets the design control requirements of Appendix B to *Title 10 of the Code of Federal Regulations* (10 CFR) Part 50, and the documentation requirement in Appendix K to 10 CFR Part 50. The documentation must contain (1) an overview of the EM, (2) a complete description of the accident scenario, (3) a complete description of code assessment, (4) a determination of code uncertainty for a sample plant accident calculation, (5) a theory manual that contains field equations, closure relationships, numerical solution techniques and limits, and limits of applicability, (6) a user manual that provides details of how the code is used, and (7) a quality assurance plan that provides the procedures and controls under which the code was developed.

The TR provides sufficient documentation for the above criteria.

Evaluation Model

Per the SRP, an EM needs to include one or more computer programs 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 provides the description of the program(s) used or the programs that will be used for transient analyses.

Accident Scenario Identification Process

The purpose of the accident scenario identification process is to identify and rank the reactor components and physical phenomena modeling requirements based on their importance and their impact on FOM for the calculations. This process is highly dependent on the type of reactor and the accident scenario of interest. A separate accident scenario identification description is needed for each accident or transient class for which the code is to be used in order to describe the accident progression and dominant physical phenomena for that particular accident.

Code Assessment

The SRP guidance establishes that all code models used in the evaluation must be assessed with the frozen version of the EM. Also the guidance requires that separate effects testing must be performed to demonstrate the adequacy of the physical models to predict physical phenomena that were determined to be important by the accident scenario identification process.

The applicant states in TR Section 7.2.1 that the transient methodology presented is code-independent and therefore can be applied regardless of the computer code used. A complete code assessment is thus not presented in the TR; the assessment performed in the TR is for demonstration purposes only.

Since the TR introduces no models, this portion of the guidance is only applicable in that the RAIs and responses discuss the performance or acceptability of certain models to address issues related to scope of applicability changes by the TR (e.g., BWR/2 through BWR/6). The separate effects test (SET), integral effects test (IET), and scaling aspects of the guidance are also similarly applicable.

Uncertainty Analysis

The SRP guidance establishes that the uncertainty analysis must address all important sources of code uncertainty, including mathematical models in the code and user modeling such as nodalization. The major source of uncertainty should be assessed in a manner consistent with the results of the accident sequence identification process which this TR addresses. The uncertainty analysis must include those in theoretical models or closure relationships determined from comparison to separate effects tests, uncertainties due to scaling of the basic models, and closure relationships. Also, the sources of uncertainties in plant model input parameters for plant operating conditions, such as, accident initial conditions, set points, and boundary conditions.

The SRP guidance also states that when a code is used in a licensing calculation, the combined code and application uncertainty must be less than the design margin for the safety parameter of interest. Examples of safety parameters applicable to BWR/2 through BWR/6 analysis are reactor vessel pressure (RVP), linear heat generation rate (LHGR), and minimum critical power ratio (MCPR). The analysis should include a sample uncertainty evaluation for a typical plant application.

<u>Quality Assurance Plan</u>

The SRP states that the EM is maintained under quality assurance program that meets the requirements of Appendix B of 10 CFR Part 50. As a minimum, the program must address design control, document control, software configuration control and testing, and corrective actions.

2.0 <u>REGULATORY EVALUATION</u>

The NRC staff used certain sections of 10 CFR 50.34, "Contents of Applications: Technical Information," that require the licensee/applicant (or vendor) to provide safety analysis reports to the NRC detailing the performance of systems, structures, and components provided for the prevention or mitigation of potential accidents.

General Design Criterion (GDC) 10 (Reference 6) requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDL) are not exceeded during any condition of normal operation, including the effects of AOOs.

The SAFDLs that are addressed in GDC 10 are divided into two limits on MCPR and LHGR for clad strain and fuel centerline temperature (FCT), and peak fuel enthalpy. The MCPR safety limit is used as an acceptance limit to protect fuel cladding from overheating as described in Section 4.4 of SRP. The overpower LHGR limit protects the fuel cladding from exceeding 1 percent plastic strain and the fuel pellet from centerline melting per Section 4.2 of SRP.

The acceptance criteria for AOO events are defined in Section 15.0 of SRP and meet the GDC specified in Reference 6 that are applicable to AOOs (i.e., GDC 10, 13, 15, 17, 20, 26, 29, 60, and 64).

Criterion 13 requires that the instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation, for AOOs, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the

fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems.

Criterion 15 requires that reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs.

Criterion 17 requires onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety.

Criterion 20 requires that protection system be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that SAFDLs are not exceeded as a result of AOOs and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Criterion 26 requires two independent reactivity control systems of different design principles be provided by means of control rods and a reactivity control system with capability of reliably controlling the rate of reactivity changes.

Criterion 29 requires protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs.

Criterion 60 requires that the nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including AOOs.

Section 15.0 of SRP specifies the following acceptance criteria for AOO events:

- Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

- Fuel cladding integrity shall be maintained by ensuring that the critical power ratio (CPR) remains above the MCPR safety limit for BWRs.

- An AOO should not generate a postulated accident without other faults occurring independently or result in a consequential loss of function of the RCS or reactor containment barriers.

The design requirements for ATWS events for evolutionary plants are defined in SRP Section 15.8, Subsection 15.8.II in Paragraph 3.C of “Specific Acceptance Criteria” and restated as follows:

- Coolable geometry for the reactor core. If fuel and clad damage were to occur following a failure to scram, GDC 35 requires that this condition should not interfere with continued effective core cooling. 10 CFR 50.46 defines three specific core-coolability criteria: (1) Peak clad temperature shall not exceed 1221°C (2200°F), (2) Maximum cladding oxidation shall not to exceed 17% the total cladding thickness before oxidation,

and (3) Maximum hydrogen generation shall not exceed 1% of the maximum hypothetical amount if all the fuel clad had reacted to produce hydrogen.

- Maintain reactor coolant pressure boundary integrity. Appendix A to WASH-1270 states that in evaluating the reactor coolant system boundary for ATWS events, “the calculated reactor coolant system transient pressure should be limited such that the maximum primary stress anywhere in the system boundary is less than that of the ‘emergency conditions’ as defined in the ASME Nuclear Power Plant Components Code, Section III.” The acceptance criteria for reactor coolant pressure, based upon the ASME Service Level C limits, are approximately 10.3 MPa (1500 psig) for BWRs and approximately 22MPa (3200 psig) for PWRs (pressurized water reactors).
- Maintain containment integrity. Following a failure to scram, the containment pressure and temperature must be maintained at acceptably low levels based on GDC 16 and 38. The containment pressure and temperature limits are design dependent; but to satisfy GDC 50, those limits must ensure that containment design leakage rates are not exceeded when subjected to the calculated pressure and temperature conditions resulting from any ATWS event.

3.0 TECHNICAL EVALUATION

3.1 INTRODUCTION

TR WCAP-17203-P/WCAP-17203-NP, Revision 0-2, discusses methods for BWR fast transients including ATWS and IE that are occurring in ABWRs as well as BWR/2-6 designs, while extending their applicability to first core analysis and introduce a new Monte Carlo based uncertainty analysis. The TR includes the following sections:

- Transient Grouping and Plant Specification
- Acceptance Criteria
- Phenomenological Description
- Phenomena Identification and Ranking
- Analysis Methodology
- Uncertainty Analysis
- Demonstration Analysis

3.2 TRANSIENT GROUPING AND PLANT SPECIFICATIONS

The NRC staff in a RAI requested the applicant to provide a list of all fast transients as they pertain to the BWR/2 through BWR/6 contained in the TR in comparison to the similar ones provided in CENPD-300-P-A. The applicant states that CENPD-300-P-A is a comprehensive umbrella document that describes the application of the NRC approved fuel and core design and analysis codes in the licensing safety analysis process to evaluate any plant modification. In addition, it also describes methodologies applied to core and fuel design and for evaluating potentially limiting events for reload applications. WCAP-17203-P/WCAP-17203-NP, Revision 0-2, provides a generic, code-independent methodology for evaluating only fast transients including both potentially limiting as well as non-limiting events.

TR WCAP-17203-P/WCAP-17203-NP, Revision 0-2, provides a generic evaluation methodology for all fast transients considered in Table 2-1 of Reference 2 and the special event ATWS. In WCAP-17203-P/WCAP-17203-NP, Revision 0-2, the fast transients are divided into groups to facilitate the identification and ranking of the important phenomena in the following categories:

- Pressure Increase Events (PI)
- Pressure Decrease Events (PD)
- Reactor Coolant Flow Increase Events (RCFI)
- Reactor Coolant Flow Decrease Events (RCFD)
- Feedwater Flow Increase Events (FWFI)
- Feedwater Flow Decrease Events (FWFD)
- Reactor Coolant Temperature Increase Events (RCTI)
- Reactor Coolant Temperature Decrease Events (RCTD)
- Anticipated Transients without Scram (ATWS)

Specific transients and their frequency of occurrence are established in the plant licensing bases and documented in the SAR or design control document (DCD). AOOs and IE capture the phenomena classified in the PIRT which consists of identifying and ranking dominant phenomena during a transient with respect to their influence on the FOM.

The ATWS events that are evaluated are mitigated by the following manual and automatic shutdown process:

1. Reactor shutdown by alternate control-rod insertion (ARI), (10 CFR 50.62).
2. Reactor shutdown by fine-motion control rod drive (FMCRD) run-in (Only in ABWR design).
3. Reactor shutdown by activation of standby liquid control system (SLCS).

3.3 ACCEPTANCE CRITERIA

Section 3.1 of the TR provides the applicant's acceptance criteria for AOOs and ATWS events. Section 3.2 of the TR lists the FOM that are derived from the regulatory requirements corresponding to the acceptance criteria described in Section 3.1 of the TR. The acceptance criteria for AOOs are defined to meet the requirements related to the GDC for nuclear power plants specified in Appendix A to 10 CFR Part 50.

The acceptance criteria for AOOs are listed below:

- Limit for radioactive effluents must comply with regulations in 10 CFR Part 20 and 10 CFR Part 100.

- SAFDLs that are addressed in GDC 10 is divided into limits on MCPR, LHGR and clad strain and FCT, and peak fuel enthalpy. The MCPR safety limit protects the fuel cladding from overheating as per SRP, Section 4.4. The overpower LHGR limit protects the fuel cladding from exceeding 1 percent plastic strain and protects the fuel pellet from centerline melting as required by SRP Section 4.2.

- The upset limit for peak reactor vessel pressure is 110 percent of the reactor pressure vessel (RPV) design pressure, as per SRP, Section 5.2.2.

- Fuel enthalpy limit is applied to provide protection from rapid energy deposition events and therefore not used for the fast transients and the ATWS events listed in Table 2-1 of the TR.

For fast transients categorized as postulated accidents, the acceptance criteria considered are the radiological consequences within regulatory limit and the primary system pressure be maintained below the acceptable design. The evaluation of radiological consequences is beyond the scope of this TR, however, the number of failed fuel rods can be determined. For calculating the fuel rod failures, the MCPR or the cladding temperatures are evaluated based on the plant's licensing bases.

The acceptance criteria for ATWS event are listed below:

- Long term core cooling must be assured by meeting the cladding temperature and oxidation criteria according 10 CFR 50.46 and SRP, Section 4.2

- The RPV integrity should be maintained by limiting the maximum primary stress within the RCPB to the limits defined in ASME Section III, RPV integrity as required by GDC 14 of 10 CFR Part 50, Appendix A.

- The long-term containment capability must be assured by limiting the maximum containment pressure to the design pressure of the containment structure and the suppression pool temperature to the wetwell design temperature in order to ensure compliance with GDC 16.

- Long term shutdown cooling must be assured subsequent to an ATWS event and the reactor be maintained in a cold shutdown condition as required by GDC 35.

FOM (as defined in SRP Section 15.0.2) are quantitative standards of acceptance that are used to define acceptable answers to safety analysis, such as, departure from nucleate boiling ratio (DNBR) limits and fuel temperature limits. Table 3.1 of the submittal provides a list of FOMs for both AOOs and ATWS events.

The FOM identified in the TR for the AOOs are:

- Minimum critical power ratio (MCPR)
- Reactor vessel pressure (RVP)
- Linear heat generation rate (LHGR)

The FOM identified for the ATWS events are:

- Cladding temperature
- Reactor Vessel Pressure (RVP)
- Mass and energy release to containment

3.4 PHENOMINOLOGICAL DESCRIPTION

3.4.1 Pressure Increase (PI)

A general description of each of the transient groups, Pressure Increase/Pressure Decrease (PI/PD), Reactor Coolant Flow Increase/Decrease (RI/RD), Feedwater Flow Increase/Decrease

(FI/FD), and Reactor Coolant Temperature Increase/Decrease (TI/TD) is provided in the TR. The events pertaining to each category are discussed briefly.

The methodology described in the TR is general and the actual EM will be described sufficiently in plant licensing submittals. This is acceptable to the NRC staff.

In response to NRR RAI-17 and NRR RAI-18 (Reference 30), the applicant responded and modified the PI section of TR. Generally a PI transient is initiated by a disturbance in the valve position in the steam lines that causes a decrease in steam flow. For example, in the feedwater flow increase transient, the closure of the turbine stop valve initiates the transition of the event that was originally a feedwater flow increase event into the pressure increase event group listed in Table 4-1 of the TR. The pressure upstream of the closed valve will start to increase and will lead to a steam compression wave which will eventually increase the pressure inside the RPV. This increase in RPV pressure will result in an increase in the core inlet subcooling and cause the boiling boundary in the core to move upwards, resulting in a core average void decrease. The resulting negative void reactivity will increase the fission power and will result in larger power increase.

The core average void decrease will be interrupted by one of the following phenomena:

(1) In the absence of actions in the plant system, the resulting increase in heat flux generates more boiling and the resultant increase in voids reduces the fission power and causes a fuel surface heat flux decrease causing an increase in CPR.

(2) If the reactor scrams, the power will be rapidly decreasing due to insertion of control rods (CRs), the fission power decreases while the CPR starts to increase.

3.4.2. Pressure Decrease (PD)

Table 4-2 of TR lists the increase in steam flow (SRP Section 15.1.3) and inadvertent opening of a safety/relief valve (SRP Section 15.1.4/15.6.1) as pressure decrease transients. A PD transient results when steam flow is increased caused by malfunction of a valve in the steam line or when several safety/relief valves inadvertently open. The PD results in decreased core inlet subcooling and causes the core boiling boundary to move downwards in the core. This causes void increase and fission power decrease. The core average void increase will be interrupted in all cases when the power decreases and when the CPR increases.

3.4.3 Reactor Coolant Flow Increase/Decrease (RCFI/D)

Reactor Coolant Flow Increase (RCFI)

As Table 4-3 of TR indicates, the RCFI transient results from startup of an inactive recirculation pump or an operator/controller error, causes the amount of water flowing into the reactor core to increase, resulting in a gradual power increase due to a decrease in core void. The core average void decrease is interrupted by either (1) if no action is taken by safety systems, the core average void will continue to decrease, power increases, followed by steam flow increase leading to an increase in pressure in the steam dome resulting in a large steam pressure drop, or (2) if a reactor scram is initiated, power is reduced and due to negative reactivity inserted by the CRs, the resulting in the increase in CPR.

Reactor Coolant Flow Decrease (RCFD)

RCFD transients, caused by a trip of a pump or a flow controller malfunction, result in a smaller amount of coolant entering the reactor core and will cause an increase in average void resulting in a reduction of power. The effect of this transient on the core is a miss-match between the heat generated in fuel and transferred from the fuel, potentially causing fuel safety limits to be exceeded.

3.4.4 Feedwater Flow Increase/Decrease

Feedwater Flow Increase (FWFI)

A FWFI transient causes the amount of water flowing in to the core to increase, which causes excess heat removal causing the moderator void and temperature to decrease, thereby, reducing the CPR. The core average void decrease will be interrupted by one or more of the following:

- If there is no intervening action, the core average void will continue to increase causing increase in power and increase in steam flow leading to increase in steam dome pressure. The CPR is also reduced as a result of increase in power.
- If a turbine trip occurs, turbine stop valves close, resulting in decrease in steam flow. This results in pressure increase and decrease in core void followed by increase in power as the neutron flux increases. This is treated as a pressure increase transient.
- If the reactor scram is initiated, power will decrease from the core bottom due to inserted CRs resulting in an increase in CPR.

Feedwater Flow Decrease (FWFD)

During the loss of feedwater flow transient, the downcomer water level decreases while the temperature increases to saturation. This causes core inlet temperature to increase. The decreasing water level causes a reduction in core inlet subcooling, thereby, increasing the core average void while decreasing power. The core effects that are possible are: (1) decreased power will reduce the fuel temperature and limits the power reduction due to positive Doppler effect; and (2) One or more auxiliary feedwater system starts and a recirculation pump runback or trip and a reactor scram may be initiated.

3.4.5 Reactor Coolant Temperature Increase/Decrease

Reactor Coolant Temperature Increase (RCTI)

Loss of normal feedwater flow will result in RCTI. The impact of this group of transients on the system is similar to that under feedwater flow decrease.

Reactor Coolant Temperature Decrease (RCTD)

Transients that cause RCTD are decrease in feedwater temperature and inadvertent startup of an emergency core cooling system (ECCS). These transients will lead to a decrease in downcomer temperature, increase in core subcooling leading to a reduction in the core average void. Power increases causing steam flow to increase. This transient will result in the steam dome pressure increase, as well as a high average power range monitor may cause the reactor to scram.

3.5 PHENOMENA IDENTIFICATION AND RANKING TABLE (PIRT)

As stated in Section 1.1.4 of RG 1.203, “Transient and Accident Analysis Methods,” the EM development and assessment should be based on a credible and scrutable PIRT. The development of this PIRT is an important step in the EM development and assessment process. The PIRT is used to determine the requirements for the physical model development, scalability, validation, and sensitivity studies. The importance rankings used are “high,” “medium,” and “low” and all such rankings are quoted to indicate that they are rankings in the main body of this text.

The phenomena identified under the first seven categories defined in the TR (i.e., Categories A through G) are very similar to the phenomena identified in the PIRT developed in NUREG/CR-6744 (Reference 7), which was developed by the NRC for loss-of-coolant accidents (LOCAs) in operating BWRs. The last category in the TR PIRT (i.e., Category H) identifies components/systems important specifically for AOO and ATWS events.

Revision 0 of the TR (Reference 1) did not include the MSIV in the Category H of PIRT. Since the modeling of MSIV operation is essential for many AOOs considered in the TR, the NRC staff in NRO-RAI 8 requested the applicant to provide the basis for not including the MSIV components and the associated phenomena in the PIRT. Subsection 1.1.4 of RG 1.203, recommends PIRT development down to the component function or component performance parameter level. In such cases, MSIVs are different from steam lines, as are feedwater lines from feedwater pumps. In response to the RAI (Reference 8), the applicant clarified that the **[** **]** and committed to update the discussion of component H8 in Tables 5-2 and A-1 accordingly. The applicant further added as H17, **[** **]** H18, **[** **]** and H19, **[** **]** to the list of components in PIRT.

3.5.1 Phenomenon Identification and Ranking Table Importance Ranking

A PIRT for the AOOs and ATWS is presented in Section 5 of the TR and is stated to be applicable to ABWR and BWR/2 through BWR/6. The TR indicates that the phenomena identified in the PIRT are based on the following previously developed PIRTs, with some additions based on the opinions of an expert panel:

- “Phenomenon Identification and Ranking Tables (PIRTs) for Loss-of-Coolant Accidents in Pressurized and Boiling Water Reactors Containing High Burn up Fuel” (Reference 7)
- “Phenomenon Identification and Ranking Tables (PIRTs) for Rod Ejection Accidents in Pressurized Water Reactors Containing High Burnup Fuel” (Reference 31)
- “Phenomenon Identification and Ranking Tables (PIRTs) for Power Oscillations Without Scram in Boiling Water Reactors Containing High Burnup Fuel” (Reference 32)

The phenomena identified in the PIRT are defined in Appendix A of the TR and are grouped into eight categories:

1. Initial conditions
2. Transient power distribution
3. Steady-state and transient cladding to coolant heat transfer and core spray heat transfer
4. Transient coolant conditions as a function of elevation and time
5. Fuel rod response

6. Multiple rod mechanical effects
7. Multiple rod thermal effects
8. Plant component/system data

The PIRT presented in the TR provides the rationale for importance ranking of each phenomenon. However, the rationales for some phenomena were not adequately described in the TR, and RAIs were used to seek additional information as follows:

A1. **[ ]**

In the TR, **[ ]** The reason for these transient-specific ranking differences is not clear from the rationale provided in Table 5-2 of the TR, as was pointed out in NRR-RAI 8a (Reference 8). **[ ]**

A2. **[ ]**

In the BWR LOCA PIRT, **[ ]** The NRC staff in RAI-9b asked the applicant for a valid rationale for this ranking.

In the response to NRO-RAI 9b (Reference 8), the applicant clarified that **[ ]**

Through the RAI process, the NRC staff asked for the clarification on the ranking of **[ ]** The RAI responses (References 8, 9, and 33) provided detailed analyses of the effect of variation of the **[ ]** the response is acceptable, and NRO-RAI 9b and its supplements are resolved and closed.

A7. **[ ]** and A18. **[ ]**

The NRC staff asked clarifications for A7 and A18 in Revision 0 of the TR through an

NRO-RAI-9e (Reference 8). The rationale for A7 and A18 was simply that both have significant impact on determining the outcome of the transient. Table A-1 of the revised TR for A7 has indicated that the **[**

]

The rationale for A18 was revised in the latest version of the TR as **[**

]

A8. **[ ]**

This initial condition is ranked as being of **[**

] (A2) raised in NRO-RAI 9c. As a result of the RAI response (Reference 9), the rationale for this initial condition was appropriately updated.

A11. **[ ]**

[

] NRO-RAI 9d pointed out this inconsistency.

In response (Reference 8), the applicant agreed to revise the TR to make the ranking of **[**

] However, NRO-RAI 9d S01 (Reference 17) asked the applicant to justify this ranking. In response (Reference 24), the applicant reported results from sensitivity calculations that show phenomena A11 and A14 should actually be ranked **[**
] transients and committed to rank them as such in the TR. This is acceptable to the NRC staff, and Phenomenon A11 accurately reflects the latest response.

NRR-RAI 9b raised similar issues, and in response, the applicant performed additional sensitivity calculations for all transient classes (PI/PD, RCFI/D, FWFI/D, and RCTI/D) including ATWS to assess the rankings of Phenomenon A11 which demonstrated that **[**

]
Consequently, the applicant proposed a **[**
] for Phenomenon A11 and committed to update Table 5-2 of the TR, including the rationale (Reference 12).

B2. **[ ]**

[
] While it is important that this be correctly modeled, it was not clear why the TR ranks it **[ ]** This issue was raised in NRO-RAI 9f. The applicant explained that the **[**
]
(Reference 8).

C1. **[ ]**

Since the rationale given for this phenomenon was not meaningful, the NRC staff requested clarifications in NRR-RAI 8 (Reference 12). The applicant clarified in its response, that this phenomenon has insignificant impact on the system response or the outcome of the transient and committed to revise the rationale in the TR accordingly. The TR was revised and the response does not alter the conclusions regarding acceptability for BWR/2 through BWR/6.

C2. **[ ]**

These phenomena are assigned a **[ ]** in the TR. However, in BWRs, these phenomena typically **[**
] Because the basis for the **[ ]** is not described in the TR, NRO-RAI 9g sought clarification. In response (Reference 8), the applicant explained that the **[**
] of Phenomenon C2 is reasonable. The issue is resolved and closed.

D4. **[ ]**

In response to NRR-RAI 10 S1 (Reference 10) that requested the applicant to update the definition of this item, the applicant revised the definition in TR Table A-1 to clarify that it indicates **[ ]** This does not alter the conclusions regarding acceptability for BWR/2 through BWR/6.

D5. **[ ]**

As discussed for Phenomenon A7 **[ ]** the applicant revised the definitions of Phenomena A7 and D5 and updated TR Table A-1 accordingly in response to

NRO-RAI 10g S01 (Reference 17). The new rationale specifies that the **[**
] This is acceptable, and the issue is resolved and closed.

D6. **[ ]**

In NRO-RAI 9h, the applicant was asked to provide the basis for the **[ ]** of this phenomenon. In response (Reference 8), the applicant stated that, due to the **[**

] The NRC staff agrees, and the issue is resolved and closed.

E1. **[ ]**

[

] The NRC staff agrees with this explanation, and the issue is resolved and closed.

E4. **[ ]**

This phenomenon is assigned a **[**

] In response (Reference 17), the applicant provided a scaling analysis to estimate the **[**

] This response is acceptable to the NRC staff, and the issue is resolved and closed.

E6. **[ ]**

This phenomenon is assigned a **[**
] and NRO-RAI 9k S01 requested the applicant to explain the rationale for this ranking. In the response (Reference 17), the applicant clarified that this phenomenon refers to the **[**
] The applicant agreed to update the description of Phenomenon E6 in TR Tables 5-2 and A-1 to reflect this clarification. This response and resulting change to the TR are acceptable, so this issue is resolved and closed.

G1. **[ ]**

These phenomena are assigned a **[**

] Therefore, NRO-RAI 9l sought a basis for the ranking. In response (Reference 8), the applicant stated that in **[**

] The NRC staff agrees with this explanation. Therefore, the issue is resolved and closed.

H1. **[ ]**

As discussed previously in this section, in response to concerns in NRR-RAI 11 (Reference 30) and NRR-RAI 11 S1 (Reference 10) about **[ ]** the applicant revised the definition of Phenomenon H1 in the TR and created the new Phenomenon H18, **[ ]** The RAI response does not alter the conclusions regarding acceptability for BWR/2 through BWR/6.

H5. **[ ]**

NRR-RAI 9c S1 questioned the applicant's statement that **[ ]** In response (Reference 22), the applicant committed to update the rationale for this phenomenon to clarify that its purpose is **[ ]** The change has been confirmed in the TR, and the response does not alter the conclusions regarding acceptability for BWR/2 through BWR/6.

In summary, the responses to the RAIs and their supplements in the above list are acceptable, and the stated concerns are resolved. The applicant revised the PIRT in Table 5-2 of the TR and the phenomena definitions in Table A-1 in accordance with the responses provided.

3.5.2 Additional PIRT Phenomena

The NRC staff found that several phenomena were not directly identified in the TR PIRT even though they may be expected to play an important role under the conditions of the AOO and ATWS events. RAIs requested the applicant to provide an explanation for the lack of inclusion of the following phenomena in the PIRT:

[ ]

Operation of safety/relief valves (SRVs) is typically important for the transients identified in the TR. As NRO-RAI 10a pointed out, the flow through SRVs is expected to be governed by the **[ ]** In response to NRO-RAI 10a (Reference 8), the applicant explained that the **[ ]** is covered in the PIRT under Phenomenon H9, **[ ]** The response is acceptable, so the issue is resolved and closed.

[ ]

[ ] (i.e., due to trip of the reactor internal pumps (RIPs) in ABWR or recirculation pumps in BWR/2 through BWR/6). In the responses to NRO-RAI 10b S01 (Reference 17) and NRR-RAI 10a (Reference 30), the applicant proposed to update the PIRT to include Phenomenon D7 **[ ]** as a separate phenomenon and to update TR Tables 5-2 and A-1 accordingly. The applicant provided a detailed description of this phenomenon and its rankings for AOOs and ATWS events. The applicant's rationale for **[**

] Table A-1, Item D7 of the TR defines **[ ]**

[ ]

This phenomenon is modeled as an output of the steam separators. The NRC staff noted this in NRO-RAI 10c. In response (Reference 8), the applicant clarified that the **[ ]** in the PIRT. This is also indicated by the description of Phenomenon H2 provided in Table A-1 of the TR. The response is acceptable.

[ ]

The NRC staff in NRO-RAI 10d, requested information regarding the **[ ]** in the PIRT phenomena. This RAI also noted the phenomenological description of fast transients and ATWS events in Section 4 of the TR that there is a substantial interaction between the thermal-hydraulic system behavior and the neutronics due to the reactivity feedback effects. In response (Reference 8), the applicant clarified that **[ ]** in the PIRT. The response is acceptable.

[ ]

Modeling of **[ ]** as NRO-RAI 10f S01 and NRR-RAI 10b noted. In the responses, the applicant proposed to update TR Tables 5-2 and A-1 to include D8, **[ ]**

[ ]

The applicant clarified that the **[ ]** in the PIRT. The response is acceptable.

[ ]

As requested in NRO-RAI 10i S01 and NRR-RAI 10b, interfacial transfer of mass, momentum, and energy are highly dependent on two-phase flow regime or phase topology. The prediction of flow regime and use of appropriate flow-regime-dependent interfacial mass, momentum, and energy transfer constitutive relations (closure relations) are essential for the prediction of two-phase flow and void fraction distribution in the reactor. In its response (Reference 30), the applicant proposed **[ ]** The applicant committed to update TR Tables 5-2 and A-1 accordingly. The applicant also provided descriptions of these phenomena and their rankings for the AOOs and ATWS. The response is acceptable, and the changes have been confirmed in the TR.

The responses to the above NRO-RAIs and their supplements are acceptable, and the changes listed above have been confirmed. Therefore, the stated concerns are resolved, and the corresponding RAIs are closed. The responses to the NRR-RAIs do not alter the conclusions regarding acceptability for BWR/2 through BWR/6.

3.6 ANALYSIS METHODOLOGY

TR Section 6 presents the analysis methodology for each transient category defined in Section 2. The evaluation process for each transient group consists of the following steps:

- Definition of the limiting condition(s) that most significantly affect(s) the analysis
- Evaluation of core and fuel operating limits
- Parameter selection process
- Specification of input parameters used in the uncertainty analysis

3.6.1 Limiting Plant States and Events

The NRC staff requested clarification in NRO-RAI 14 (Reference 34) on the statement in Section 6.1 of the revised TR:

> *Each potentially limiting transient event is evaluated for the limiting plant condition(s) throughout the plant operating domain.*

The applicant responded that the limiting events are determined on a plant-specific basis. This is consistent with other supplied RAI responses that emphasize the need to maintain a plant-specific licensing basis that will include plant-specific transient grouping, parameter input, and analysis code(s). The response is acceptable to the NRC staff.

The NRR RAI 23 requested the applicant to expand the above discussion to ensure conservatism in the methodology. The revised TR defines the limiting conditions by initial power level, recirculation flow, system pressure and feedwater temperature, and associated uncertainties. The nominal reactor power plus a 2 percent uncertainty is used in the analysis complying with the SRP requirement. A lower power level can be used in the analysis if it can be justified. Figure 6-1 of the TR provides a graphical relationship for nuclear safety-related setpoints, such as, safety limit and its analytical limit, trip setpoint, and nominal setpoint. **[**

] According to 10 CFR 50.36(c)(ii)(A) the LSSS for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. When a LSSS is specified for a variable, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If the automatic safety system does not function as required, the licensee shall take appropriate action including shutting down the reactor.

The applicant revised several sections in Chapter 6 of the TR which will be discussed below.

3.6.2 Fuel and Core Operating Limits

Fuel and core operating limits consist of different limiting parameters that cannot be violated during operation of the plant, such as the Operating Limit MCPR (OLMCPR) and LHGR limitations introduced by transient overpower.

Based on several RAIs (NRO-RAI 15, NRO-RAI 15 S01, NRO-RAI 16, and NRR-RAI 22) (References 8, 17, and 31), the applicant made several changes in Sections 6.2 of the TR. Section 6.2.1 of the TR describes how the OLMCPR would be calculated for full-power conditions. A formula is presented for calculating a generalized limiting MCPR based on the OLMCPR and factors representing deviations from operating power and flow rate. NRO-RAI 16 requested that the applicant provide details on how these factors in the formula are determined. The applicant responded (Reference 8) with a correction to the formula contained within TR Revision 0 and with details on how the two factors will be derived based on the reactor's power-flow map. This response is acceptable, the correction was made in the revised TR, and the RAI is resolved and closed.

[

]

The applicant specifies that some AOO events can be more restrictive at off-rated conditions depending on the plant-specific allowable operating domain. In a response to NRO-RAI 17, to clarify the statement that some AOOs can be more restrictive at off-rated conditions, the applicant responded in Reference 11 that **[** **]** However, the response did not describe the methodology used to determine when and how off-rated conditions would be limiting. NRO-RAI 17 S01 requested a description of the methodology.

In response to a supplemental RAI (References 8 and 9), the applicant further clarified that the **[** **]** The applicant's explanation is satisfactory, and the issue is closed and resolved.

The potentially limiting events are analyzed using a detailed thermal-hydraulic code that are selected during this licensing event selection process. Expert engineering judgement or analytical processes are used to eliminate certain events and this disposition of events is documented.

3.6.2.1 MCPR Operating Limit with Uncertainty

For OLMCPR with uncertainty, the following procedures are followed:
[

]

3.6.2.2 LHGR Operating Limit

The LHGR operating limit is specified for each fuel type in a given cycle. The plant LHGR operating limit is the most restrictive of the following:
[

]

[

]

3.6.2.3 Overpressurization Protection Methodology

[

] These events are used to confirm the adequacy of the plant's pressure relief system prior to each reload cycle. The evaluation procedure are listed in Section 6.2.3.2 of the revised TR and in the response to NRR RAI-41.S1 (Reference 10). These evaluation procedures are:
[

]

The overpressurization MSIV closure event could be treated as an emergency condition consistent with the current version of the ASME code with acceptable results compared to ASME emergency condition limits, i.e., the reactor pressure acceptance limit of 120% of design pressure. However, **[** **]**
[
]

NRO RAI 28 asked the applicant to provide the rationale for choosing **[**
] as the dominant overpressurization event. In response, the applicant explained that, based on the sequence of events during an overpressurization transient, the **[**

] The supplement to the RAI asked for information on the process and methodology used to determine other pressurization events that may be more severe than MSIV closure and the steps taken to ensure the most severe pressurization event is identified and analyzed. The response (Reference 24) indicated that **[**

] The issue is closed and resolved.

3.6.3 Analysis Codes

Section 6.3 of the TR states that the methods under discussion are applicable to the NRC-approved 1D and 3D dynamic codes. For fast transients, NRC approved 1-D and 3-D system dynamic computational code is used for analysis.

For 1-D analysis, the **[**
] Fuel performance and thermal hydraulic data is integrated into the 1-D transient analysis.

The 3-D analysis of fast transients uses a 3-D kinetics dynamic code, all nuclear and thermal-hydraulic data taken directly from a static core simulator.

In the RAI response (Reference 8) and in complying with the SRP Section 15.02 requirements of the EM (an overview, accident scenario identification process, code assessment, and uncertainty analysis), the applicant provided documentation for the same in the TR. The EM that captures the calculational framework for assessing the behavior of the reactor coolant system during postulated accident or transient uses BISON (RPA-90-90-P-A) and POLCA-T (WCAP-16747-P-A) and their appendices. BISON, POLCA-T, and GOTHIC (WCAP-16608-P-A) codes are used for analyzing special events such as ATWS.

3.6.4 Analysis Methodology

Section 6.4 of the TR discusses the analysis methodology for AOOs and separates the discussion into subsections according to the categorization of the AOOs listed in Section 4 of the TR: PI/PD, RCFI/D, FWFI/D, and RCTI/D. For each AOO category, there is a discussion of the analysis code requirements and the modeling techniques required to address the transient category. Also presented is a list of parameters deemed to provide conservative assumptions in the analysis of each AOO event in Tables 6-1 through 6-8. For example, according to Table 6-1, analysis of PI transients shall use a **[**
]

Section 6.4.1.1, “Analysis Code Requirements,” of the TR lists PI/PD analysis code requirements and the requirement that conservative input data is required for the transient analysis. This section specifies that **[**

] The NRC staff requested more information on how combinations of transient categories will be determined to be more limiting than the conditions analyzed in the PIRT and to discuss how the combinations of categories will be chosen for evaluation.

The applicant responded to the RAI in Reference 8. The applicant states that combinations of transient categories are not explicitly determined and analyzed. In the proposed methodology, **[**

]

NRR RAI-26 (Reference 11) sought clarification for limiting PI and PD transients, including transients from one category to the other. The response to the NRR-RAI 26 (Reference 11) presented a sensitivity analysis for a PD transient (opening of all turbine and bypass valves) evolving into a PI transient to demonstrate the application of the information summarized in the response to the NRO-RAI 29 S01 (Reference 9).

Section 6.4.1.2 and Section 6.4.1.3 of the TR, respectively, provides details of the PI and PD transients' methodology and modeling techniques. **[**

] Table 6-1 of the TR lists pressure increase transients' input parameters with conservative assumptions. Section 6.4.1.3 lists methodology and modeling techniques for PD transient methodology.

Section 6.4.2.1 of the TR lists the analysis code requirements for RCFI/D that are based on the phenomenological descriptions used as the basis for PIRT. The requirements for the physical EM are: **[**

] Section 6.4.2.2 lists analysis code requirements, transient modeling techniques, and conservative assumptions for the RCFI transient input parameters. Section 6.4.2.3 provided analysis code requirements, transient modeling techniques, and conservative assumptions for the RCFD transient input parameters.

Section 6.4.3 of the TR describes the general analysis code requirements for the FWFI/D, modeling techniques used for FWFI and the FWFD, and how to conservatively bias the input parameters for the FWFI/D transients' analyses. Similarly, Section 6.4.4 provides the analysis

code requirements for RCTI/D transients and how the input parameters for the RCTI/D transients must be biased for conservatism.

In RAI (Reference 35) the NRC staff asked for technical justification and to perform sensitivity study to confirm that for a **[** **]** as per Section 6.4.3.3.2 of the TR. The applicant responded in Reference 13 that for a FWFD, **[**

]

The NRC staff asked for additional information (Reference 35) for validation of the fuel time constant in Section 6.4 of TR to be conservative for both hot and average rods, or commit to performing code-specific confirmatory calculations to validate the conservative direction of the fuel time constant prior to making assumptions for various transients in plant licensing calculations. The NRC staff asked to clarify whether a hot rod/average rod fuel assembly model is capable of accurately predicting a quantity of failed fuel rods for a pump seizure or shaft break accident, as opposed to verifying that the limiting rod is undamaged (Reference 35).

The applicant in Reference 15 stated that they intend to **[**

] as shown in the response to NRR RAI-16.S 1.

In the response to NRR-RAI 43(d) the applicant provided further details on the use of the hot rod and average channel approach to determine the number of failed fuel rods (Reference 12). According to the applicant, the **[**

] (see Commitment 2 in Section 4.2 of the SE).

In summary, the responses to the RAIs related to TR Section 6.4 and their supplements are acceptable, and the RAIs are closed and resolved. The responses supplied from RAIs for TR Section 6.4 do not alter the evaluation for BWR/2 through BWR/6 aside from specifying Commitment 2, as discussed above. Furthermore, the NRC staff has reviewed Section 6.4 of

the revised TR for the analysis methodology of the transients and determined that the applicant's procedure for analyses is acceptable.

3.6.5 Anticipated Transients Without Scram (ATWS)

TR Section 6.5 addresses the analysis methodology for ATWS event. The initiators for an ATWS event are usually caused by a rapid reduction in steam flow (rapid pressurization events) or events that can evolve to a rapid pressurization event during the course of the transient. For rapid pressurization events, there is a rapid increase in the reactor coolant pressure boundary pressure, and core power. The pressure and power increase is limited by the reactor protection system, typically an automatic recirculation pump trip (ATWS-RPT) on high reactor pressure and operation of the SRVs. Reactor shutdown is accomplished by automatic or manual initiation depending on the plant design. The different plant systems that can shut down the reactor are: alternate rod insertion (ARI), fine motion control rod drive (FMCRD), and SLCS (shutdown with boron injection in to vessel coolant).

[**]** NRC staff requested clarification (Reference 8) for this and to obtain explanation for the statement in the TR that, according to the PIRT, code capability assessment (CCA), data uncertainty assessment (DUA), and **[** **]** The applicant responded that the ATWS uncertainties may be grouped into categories such as:
[**]**

[**]**

The applicant revised the TR to clarify the text in Subsections 6.5.1.1 and 6.5.1.2. The NRC staff finds this response and the changes in the TR acceptable.

As previously stated, the analysis methodology for ATWS events is developed to comply with the requirements of 10 CFR 50.62, which requires the availability of certain equipment to automatically shut down the reactor. However, the systems specified in Section 6.5 of the TR did not include the requirement for a trip of the reactor coolant recirculating pumps in BWRs. The NRC staff requested clarification regarding the requirement of coolant recirculation pump trip to mitigate the ATWS. The applicant responded to the RAI in Reference 8. The applicant stated that the mitigation of ATWS events is accomplished by a multitude of equipment and procedures. These include ARI, fine motion control rod drive (FMCRD) run-in, feedwater runback, recirculation pump trip (RPT), recirculation runback, automatic depressurization system (ADS) inhibit, and SLCS initiation. The methods and procedures applied for ATWS mitigation are determined by the plant design and can be manually or automatically initiated. The analysis for three separate cases investigated (ARI, FMCRD run-in, and SLCS activation)

showed sufficiency to demonstrate the adequacy of all ATWS mitigation actions including RPT, per SRP Section 15.8.

The NRC staff requested the applicant in RAI-12 to explain the statement that boron transport in the analysis for ATWS events will be "conservatively modeled." The applicant responded that boron transport will be modeled using a method that has been reviewed and approved by the NRC in POLCA-T, and BISON (References 13, 14, and 15). Boron transport model used by POLCA-T is described in Section 7.1 of the TR and justification for this is done in Reference 16 which is currently under review by NRC staff. The boron transport model used by BISON is documented in Section 4 of Reference 15.

The applicant has further clarified the response to RAI-12 regarding the statement for "conservatism" for the phenomena of boron transport in RAI-12 S01 (Reference 16). The applicant defined **[**

]

NRR RAI-20 and its supplement NRR RAI 20 S1 requested information on how control systems and balance of plant equipment are relevant to performing bounding analyses of transient and ATWS events in the TR. In the responses (References 18 and 19, respectively), the applicant explained that the control systems are treated consistently with Table 7-1 of the TR. The control system may be treated conservatively, nominally with uncertainty analysis or nominally without uncertainty analysis depending on PIRT ranking, CCA ranking, and the DUA. **[**

]

3.6.6 Conclusions on Analysis Methodology

The TR provides guidance for fast transient and ATWS analysis methodology, including code requirements, analysis requirements, and assumptions used to preserve the conservative nature of the results generated.

This review is limited to a general assessment of the applicability of the methodology to fast transient and ATWS analyses. Although the analysis methodology provides detailed guidance for code assessment, a code assessment that demonstrates the full use of this guidance is not used in TR Section 8.4 for the demonstration case.

The analysis methodology described in the TR is generally similar to the previous analysis methodology for BWR reload cores (Reference 3) for those parameters that have a medium or low effect on a FOM. **[**

]

The TR explains that the analyses will be performed using 1D and/or 3D NRC-approved dynamic analysis codes. When 1D analysis codes are used, the process for obtaining a 1D model from a 3D model is required. This process is not described in detail in this TR, but it is the subject of the updated BISON and POLCA-T TRs (References 20 and 15). This TR also describes some of the general code calculation capabilities needed for analyses and demonstrates the use of the PIRT and the CCA to justify the code selection. These descriptions are acceptable.

The TR explains that an ATWS analysis is performed to verify the adequacy of the mitigating equipment required by 10 CFR 50.62. **[**

] Additional specific analysis assumptions are provided for the SLCS ATWS analyses. The assumptions are acceptable.

As noted in the TR and in the responses to the NRC staff's RAIs discussed in Sections 3.6.1 through 3.6.5 of SE, the methodology that will be employed during actual licensing will be plant-specific. Consequently, the review and approval of this TR is necessarily limited to a determination regarding the soundness of the approach described. The analysis methodology is generally acceptable. In addition, it has been confirmed that the applicant has incorporated the updates into the TR as committed to in RAI responses. The responses supplied from NRO-RAIs do not alter the evaluation based upon NRR-RAIs.

3.7 UNCERTAINTY ANALYSIS

This section describes how the applicant performed the uncertainty analysis to predict a best estimate value accounting for uncertainties and biases of the input and modeling parameters to ensure operating limits and safety margins meet the acceptance criteria. The statistical analysis for uncertainty calculations is based on 95% probability with a 95% confidence level that results in conservative values.

NRO RAI 19, NRO-RAI 19 S01, and NRO-RAI 19 S02 requested details regarding what precedent exists for use of **[**

] In response, the applicant provided examples of when NRC has previously accepted best-estimate methods for fast transients and noted that Section 4.4 of the SRP directs the applicant to treat uncertainties in the values of process parameters, core design parameters, and modeling parameters with at least a 95/95 level when evaluating thermal margins during AOOs. In addition, the applicant stated that it intends to use the best-estimate method (95/95 calculations) with respect to the CPR remaining above the MCPR, consistent with the guidance of SRP Chapter 15 and SRP Section 4.4.

Regarding NRC's rejection of best-estimate methods during the review of CENPD-300-P (Reference 3), the applicant noted in the response that Method B in the original CENPD-300-P submittal, which most closely resembles the proposed uncertainty analysis in this TR, is based on the assumption of normally distributed input parameters. The applicant stated that it would not apply Method B to determine thermal margins and noted that the new method of treating uncertainties in this TR imposes **[**

] The responses to NRO-RAI 19 and its supplements are acceptable, and the issue is resolved and closed.

The uncertainty analysis methodology is described in detail in Section 7 and Appendix B of the TR. The analysis provides for a best-estimate value and a 95/95 value to be determined. Appendix B provides theoretical basis for the statistical method used in the uncertainty evaluation. The applicant has used two methods: the statistical properties of the normal distribution method and the order statistics method. The details of the statistical analysis will be discussed in detail in the next sub-sections.

3.7.1 Selection of Input Parameters for Uncertainty Analysis

Table 7-1 that lists the uncertainty evaluation matrix shows how the uncertainty methods are assigned in accordance with the PIRT and CCA ranking. Based on this guidance, **[**

] Section 7.1 of the TR states that when the **[**

] In response to the NRC staff's request for additional information regarding this issue, the applicant explained how biases and uncertainties for parameters with a "high" PIRT ranking are treated. A follow-up RAI on why the applicant **[**

]

The NRC staff has determined that this response has been adequately resolved.

Table 7-1 lists three different types of treatment for code input and modeling parameters: nominal with uncertainty analysis, conservative, and nominal without uncertainty analysis. **[**

] In conservative treatment, the relevant input and modeling parameters are set to bounding value which leads to the conservative influence on the FOMs. **[**

]

The NRC staff reviewed Section 7.1 of the TR and the related RAI responses described above and determined that the definition for input and modeling parameters uncertainty analysis and processes are acceptable.

3.7.2 Code Capability Assessment

CCA evaluates the computer code for performing analysis of fast transients and ATWS events. CCA requires that the PIRT listed in the TR and the importance must be considered. The applicant states in the TR that the transient methodology is code-independent and therefore it can be applied independent of the code and therefore it will be included in a code-specific TR.

The database to determine the code accuracy, code ranking and the establishment of uncertainties requires the following items:

- Separate effect tests (SETs) needed to develop and assess empirical correlations and other closure models.
- Integral effect tests (IETs) to assess system interaction and global code capability.
- Benchmark with other codes (optional).
- Plant transient data (if available).
- Simple test problems (or analytical solutions) to illustrate fundamental calculational device capability.

Code accuracy focuses on the capability and performance of the code. Model accuracy for each of the phenomena identified in the PIRT must be examined by an evaluation panel of subject matter experts. The data qualification is determined based on quality of test data, uncertainties, knowledge of the test facility and its operations. The model capability is determined according to a three-level scale: High/Medium/Low (related to how the phenomenon is calculated or used in determining the FOM). An example of CCA matrix is demonstrated in Table 7-2 of the TR.

The NRC staff requested in RAI-21 to elaborate the CCA process and criteria used to assess model accuracy and include examples. The applicant stated in Reference 34 that the model accuracy is judged by subject matter experts based on the validation against the SET and/or IET data. **[** **]** As an example, the applicant provided an advanced control rod hydraulic insertion model that has been verified against measurements of control rod hydraulic insertion **[** **]** at different reactor dome pressures and different pressures in the hydraulic rod insertion system gas tanks. Figure 1 of RAI-21 response shows that the code model is in good agreement with measurements.

The NRC staff asked the applicant to provide a description of the criteria against which the qualification data will be judged in determining the sufficiency and relevancy of data to the phenomenon of interest. In addition, the RAI asked the applicant to provide additional information describing how scaling considerations are incorporated in the bias and uncertainty evaluation. In the response (Reference 30), the applicant indicated that the relevancy of a particular test record is based on the level of applicability of the test data to a particular phenomenon or model. **[** **]** This response provides sufficient information to understand the “relevancy” and “sufficiency” terminology but does not provide specific acceptance criteria.

The CCA presented in this Section is limited in scope and does not serve as a full examination of a particular code. However, the CCA process presented is acceptable for the demonstration of TR.

3.7.3 Data Uncertainty Assessment

The SRP Section 15.0.2 establishes criteria which ensures that the method for calculating uncertainty contains all important sources of uncertainty. The SRP Section 15.0.2 acceptance criteria must provide the source of uncertainties in theoretical models or closure relationships be determined from comparison of separate effects tests, uncertainties due to scaling of the basic models and closure relationships, and uncertainties due to plant nodalization and solution techniques. Uncertainties in the experimental data such as measurement errors and experimental distortions must be addressed. For separate effects, tests and integral effects tests, the reviewers should confirm that the differences between calculated results and experimental data for important phenomena have been quantified for bias and deviation.

The applicant in the TR states that the DUA process links corresponding code-dependent input and model parameters (Candidate Parameters (CP)) for pertinent phenomena to allow for further uncertainty and sensitivity analyses. The CPs that have very small uncertainties are eliminated from further analyses while the remaining parameters (Relevant Parameters (RPs)) are subject to further uncertainty and sensitivity analyses. Probabilistic distribution functions (PDFs) and respective bounding values are then assigned for each of the RPs. These parameters are then input to further uncertainty analysis. The DUA consists of three steps: identification of CPs, specification of RPs and establishment of PDFs, and respective bounding values for further uncertainty analysis.

The CPs are code input and modeling parameters to simulate high ranked phenomena identified in the PIRT. CPs are developed based on code documentation. The RPs are parameters that have significant influence on FOM and serve as input to further analysis.

The RPs uncertainty intervals are developed based on the guidelines defined in Code Scaling, Applicability, and Uncertainty (CSAU) (Reference 22) which stipulates that there is no single method providing the uncertainty range or bias for all RPs. Instead RPs are grouped according to the following methods:

[

]

The NRC staff in RAI-25 asked the applicant to clarify how the DUA is addressed in the methodology and how the uncertainty analysis in the TR takes into account the uncertainties in the mathematical models, closure relationships in the underlying codes, and user modeling (e.g., nodalization and solution techniques) (Reference 21). The NRC staff also asked how uncertainties in the experimental database are factored into the uncertainty analysis. The applicant explained that the DUA described in Section 7.3 of the TR provides the process by which the mathematical models and closure relationships are performed. The response contained examples (such as void correlation) and discussed the closure relationships in codes, user modeling, and uncertainty analysis.

Uncertainty evaluation is a part of the DCA process. **[**

]

The NRC staff requested that the applicant describes how uncertainties in the experimental data such as those arising from measurement errors and experimental distortions are factored in to the uncertainty analysis. The response explained that the uncertainties in experimental databases are addressed in two ways: **[**

]

The NRC staff requested additional information on the uncertainty methodology for experimental databases (RAI-25 S01, Reference 21). In the response, the applicant explained that the number of leading parameters identified for each important phenomenon (PIRT) is based on the number of output parameters associated with this phenomenon that have a relevant effect on the FOM. According to the RAI-23 S01 response, the "relevant" parameter is numerically defined for each FOM. The response further explains how experimental and modeling uncertainties are mathematically combined, depending on whether the uncertainties or errors are relative or absolute in nature. The process discussed is mathematically correct. The applicant also explained that **[**

] The NRC staff accepts this interpretation of how the uncertainty methodology is applied to the experimental database.

In RAI-25 S02 (an extension of RAI-25 S01), the NRC staff requested clarification and mode explanation on the "Leading Parameter" which was inadequately explained in applicant's response to RAI-25 S01 (Reference 21). The applicant stated that the "Leading Parameter" is defined as a **[**

] In an example provided by the applicant, a model validation against SET was performed with a plot of measured (from an experiment) versus calculated values. **[**

] This multiplier is used as an input parameter to the uncertainty analysis. This is a "Relevant Parameter," accounting for model uncertainty. Since it does not account for experimental uncertainty, it is not a "Leading Parameter." To account for both the experimental and model uncertainty, a model validation against SET is performed and the calculated results are compared to experimental data with uncertainty. To consider both the experimental and model uncertainties a different method of creating the envelope of 2-dimensional uncertainties is used as described in Section B-7.2 of WCAP-16747-P-A (Appendix B). The leading parameter is one of the uncertainty input parameters (See Table 8-3 of TR). It is sampled according to a

defined distribution function, together with other parameters such as MCPR, PCT as demonstrated in Section 8 of the TR.

The applicant has summarized the definitions for various parameters. Candidate parameters are all code input and modeling parameters needed to simulate the high ranked phenomena identified in PIRT.

Relevant parameters are selected from the candidate parameters having a significant influence on FOM. This process is described in response to RAI-23 S01 and is demonstrated in response to RAI-25 S02 (Reference 24).

Leading parameters are relevant parameters for which the uncertainty from comparison to the experimental database (SET) is combined with model uncertainty.

The DUA presented in this section provides a general guidance by which uncertainty determination process for parameters required for analyses is performed. It does not provide a full examination of any particular code. However, the NRC staff has determined that the process described in the TR and in the RAI responses are acceptable for the purpose stated in the TR.

3.7.4 Uncertainty Analysis Methodology

This section describes the methodology for calculation of combined bias and uncertainty when evaluating operating limits or safety margins to the acceptance criteria. The stated purpose of the uncertainty analysis in the TR is to predict a best estimate value of a FOM by accounting for uncertainties and biases of the input and modeling parameters to ensure that operating limits and safety margins meet the acceptance criteria. The TR describes the process for defining applicable input and model parameters as well as the process of uncertainty evaluation. **[**

]

The parameters for the EMs are ranked "high," "medium," or "low" importance in the PIRT. Each phenomena is assigned an "importance" grade corresponding to their influence on the FOM. In the TR, **[**

]

There are several steps the applicant uses in the transient uncertainty evaluation process. Tolerance limits with 95/95 level is used in the uncertainty evaluation for all operating limits or safety margins to acceptance criteria. **[**

] The number of output parameters that depend on the type of analysis must be specified. The output parameters are discussed in Section 6 of the TR. Using the tolerance limits and the number of output parameters, the number of code runs is calculated using Equation 11 or 22 of Appendix B of the TR. Based on PIRT and CCA the input parameters are defined in terms of probabilistic distribution using the Table 7-1 as guidance. A run matrix is created for the Monte-Carlo simulation generated by sampling the probabilistic distributions of the input parameters n-times where the "n" is determined from the above two steps. Using the run matrix parameters the transient event is computed for each case and the event acceptance criteria are extracted from the output files. Finally, the results are tested for normality using the procedure in Appendix B of the TR. If the data passes the normality test, then the 95/95 value

is calculated using the Equation 28 of Appendix B of TR. If the normality test fails, then the order statistics method is used instead to determine 95/95 value. The results are tallied, and using order statistics with the 95/95 methodology, the upper and lower limits are determined from the results. **[**

]

Section 7.4.1 of the TR describes how the parameters are selected in the uncertainty analysis using the guide according to Table 7-1 of the TR. The parameters fall in to three categories:

- Nominal with uncertainty: **[**

  ]
- Nominal without uncertainty: **[**

  ]
- Conservative: **[**

  ]

3.7.5 Uncertainty Evaluation Methods

Appendix B.1 of the TR provides a summary of theoretical basis for the statistical method used in the uncertainty evaluation, namely, the statistical properties of the Normal Distribution and the Order Statistics method. Appendix B.1 of the TR describes the technical basis for single parameter uncertainty evaluation using the order statistics method. Appendix B.2 describes the technical basis for multiple parameter uncertainty evaluation by the order statistics method. Appendix B.3 of the TR provides the technical basis for the uncertainty evaluation by statistical properties of the normal distribution methods. The following sections will briefly discuss the above three methodologies.

The uncertainty analysis methodology discussed in Subsection 6.2.1.1, Section 7.4, and Appendix B.1 of the TR alludes to the choice and use of an estimator grade, which influences the number of calculations that must be performed. In RAI 27, the NRC staff requested the applicant to clarify the method for determining the estimator grade for a specific type of analysis and to provide a discussion of, or reference to, the decision-making process for the selection of the estimator grade as a function of the tradeoff between the number of calculations and the “risk of over-conservatism.” The applicant’s response provided information (References 25, 26, and 27) that describes the use of non-parametric and order statistics in estimating uncertainties in PCT, local maximum clad oxidation for large break LOCAs. The response includes a detailed discussion on the influence of the estimator grade on the analysis output. The estimator is the smallest number of a code runs in order to estimate the tolerance limit with a specified probability on a specified confidence interval. The response to RAI-27 shows different estimator grade for different combinations of probability and confidence levels. The values listed in the Table for RAI-27 response is calculated using Equation 11 of Appendix B of the TR and is repeated in the SE as Table 1. The Table from RAI-27 response lists the confidence estimator grades for probability interval ranging from 95 percent to 99 percent and for trials 59, 93, 124,

and 153 trials. For example, the confidence represents the probability of exceeding an upper X^{th} quantile for the output parameter.

Table-1. Confidence level for various probability

Probability Interval (quantile) (%)	Confidence -1st Estimator Grade (59 Trials) (%)	Confidence -2nd Estimate grade (93 Trials) (%)	Confidence -3rd Estimator grade (124 Trials) (%)	Confidence -4th Estimator grade (153 Trials) (%)
95	95	95	95	95
97	83	77	72	68
98	70	56	45	37
99	45	24	13	7

For example, Table-1 shows that using the 1st estimator grade may be seen to be bounding, as the probability (confidence) of exceeding the higher value quantiles (>95%) is largest for this case. First order estimator leads to higher estimates of 95/95 value with higher probability than higher order estimators. This estimator grade requires the lowest amount of code runs and is therefore primarily used to evaluate the FOMs. Higher estimator grade may be chosen based on the analysis type. The applicant recognizes the importance of the estimator grade in the analysis and the corresponding number of code runs will be determined on a case-by-case basis and documented with the results of the analysis. The applicant identifies two types of analysis that influence the estimator grade: (1) simultaneous analysis of several output parameters (FOM) for the ATWS event and (2) uncertainty analysis of one output parameter typical for AOO analysis where MCPR is the parameter under consideration.

For the first case, where several output parameters are evaluated, the 1st estimator grade is used in the analysis. This is due to the relatively large amount of code runs necessary to fulfill the 95/95 criterion even for the 1st estimator grade.

For the second case where the uncertainty of one output parameter is evaluated, all relevant input and modeling parameters are set to their bounding values. In this case, first estimator grade may be chosen, as this choice bounds all other estimators.

In case extra conservatism is needed, a higher estimator grade may be used in the analysis. The number of code runs will be documented together with the analysis result.

Appendix B.2 of the TR describes multiple parameter uncertainty evaluation where simultaneous evaluation of several output parameters such as OLMCPR and reactor vessel pressure. The treatment for the multiple parameter uncertainty calculations is described in an extension of Guba's treatment in Reference 28. In this extension of Guba's formulation to multiple variables, the dependency of output variables is represented by an unknown joint density distribution function. In order to solve the problem of setting tolerance limits using an unknown density function, except that it is continuous, the order statistics is found to be a satisfactory solution suggested by Wilks (Reference 27) in which the distribution free limits can be given only by means of order statistics. Order statistics is unable to exploit the total amount of information from the sample when the density function is unknown. With probability and confidence known, the analyst can anticipate either a wider tolerance interval as in the case of known density function (Reference 28). Appendix B.2 summarizes the detailed analysis in

Reference 28 which defines the number of code runs needed to obtain a specific probability (%) at a confidence interval (%) for one-sided probability interval and a specie number of parameters evaluated simultaneously. The solution of Equation 22 for probability of 0.95 and confidence of 0.95 for 1 through 3 parameters is listed in Table-2.

Table-2 Number of Code Runs for Simultaneous Evaluation of 1-3 Event Acceptance Criteria

Number of Parameters	Number of Code Runs
1	59
2	93
3	124

Appendix B.3 of the TR lists several properties of the normal distribution uncertainty evaluation methods based on the assumption that the output parameter is distributed normally. Ideally for normally distributed continuous parameters, Appendix B.3 defines mean value and standard deviation. For practical situations, the finite number of code runs, this appendix provides sample mean and sample standard deviation.

If mean (μ) and standard deviation (σ) are known the upper estimate of the 95th probability level for the parameter under investigation ($X_{95,95}$) could be calculated with 100 percent confidence by the formula:

$$X_{95,95} = \mu + 1.645\sigma$$

In practice, both μ and σ are unknown and are estimated by the sample mean and sample standard deviation from a vector of N random parameter X. For this case, a 95th probability level of the parameter under investigation is estimated with 95 percent confidence by the equation:

$$X_{95,95} = \overleftarrow{\mu} + z_{95,95}\ s_n\ ,$$

Where $X_{95,95}$ is the 95/95 estimate of the parameter under investigation and $z_{95,95} > 1.645$ is a factor for the one-sided normal tolerance limit and s_n is the standard deviation calculated using Equation (26) of Appendix B.3 of the TR.

For an example, if the OLMCPR (i) value has to be obtained from the sample of 59 runs on a 95 percent probability level, with 95 percent confidence, then the sample mean is calculated using Equation 25 of Appendix B.3, the sample standard deviation according to Equation 26, $z_{95,95} = 2.024$, and the OLMCPR(i) per Equation 28 of Appendix B.3.

3.7.6 Summary and Conclusion for Uncertainty Analysis

This section considers the statistical treatment of uncertainties as well as biases. As discussed in Subsection 3.7.3 of the SE, the applicant provided an acceptable response to RAI 25 S02 regarding how biases are reflected in its uncertainty analysis. The NRC staff considered several related guidance documents in the process of evaluating the uncertainty methodology in the TR, namely RG 1.203 (Reference 4), which provides guidance regarding the subject of uncertainties in a very broad sense and RG 1.157 which explains the acceptance of the 95 percent probability (Reference 29).

RG 1.157 states that:

> The basis for selecting the 95% probability level is primarily for consistency with standard engineering practice in regulatory matters involving thermal hydraulics. Many parameters, most notably the departure from nucleate boiling ratio (DNBR), have been found acceptable by the NRC staff in the past at the 95% probability level.

The applicant's uncertainty analysis is intended to predict a best estimate value to account for uncertainties and biases of the relevant input and modeling parameters to ensure operating limits and safety margins meet the acceptance criteria. The methods outlined above can correctly result in the determination of FOM with 95/95 interval provided the uncertainty methodology is correctly implemented. The NRC staff has noted that there are many areas where engineering and statistical methods are applied appropriately and successfully in evaluating nuclear reactor operating limits and margins to safety including areas that are deterministic in nature.

The uncertainty analysis as described in the TR has been verified and found mathematically correct and acceptable. It can provide statistically correct 95/95 values for FOM and can be used to establish operating limits and safety margins. The TR and responses to RAIs address sources of code uncertainty, including the mathematical models in the code and the user-selected inputs. However, appropriate care and review should be performed regarding the chosen distributions for input parameters that are subject to the Monte Carlo sampling.

The NRC staff has found the uncertainty analyses methodology acceptable in the determination of operating parameters, safety limits, and margins to safety for operating BWR/2 through BWR/6 plants.

3.8 DEMONSTRATION ANALYSIS

Chapter 8 of the TR describes the application of the Fast Transients and ATWS methodology for Load Rejection without Bypass transient (LRWBP) event. The description consists of transient group and power plant specification, definition of operating limits or safety margins to acceptance criteria, PIRT selection, computational tool selection, CCA, DUA, nominal case analysis, and uncertainty evaluation. The plant type used for the demonstration analysis is the ABWR design. Therefore the details of the demonstration analysis is not described in this Section.

4.0 LIMITATIONS, CONDITIONS, AND COMMITMENTS

The NRC staff's approval of WCAP-17203-P/WCAP-17203-NP, Revision 0-2, for use in the BWR/2 through BWR/6 designs is subject to the following conditions/limitations and in accordance with the commitments made by the applicant in the cited RAI responses.

4.1 Limitations and Conditions

1. Since the CCA and DUA are review acceptance criteria per SRP Section 15.0.2 for transients and accidents EM, in order to support the approval of changes to any EM, any licensing submittal that uses the EM methodology defined in the approved version of the TR must include a detailed description of the changes to the EM.
2. Even though the TR is applicable to BWR/2 through BWR/6 and ABWR, this SE addresses the application of the methodology only to BWR/2 through BWR/6.
3. The applicant provided a response to the RAI-1 (Reference 30) indicating that the EM capturing calculational framework for evaluating the behavior of the RCS during a postulated accident is contained in code specific TRs: RPA-90-90-P-A (BISON) and WCAP-16747-P (POLCA-T) and its appendices. For AOO fast transients and ATWS events (design bases and methodology for WCAP-17203-P) the code methods listed within the TR are WCAP-16747-P-A (POLCA-T) and qualification codes are WCAP-16747-P Appendices C and D, as well as BISON series of codes. Only NRC-approved codes shall be utilized for fast transients and ATWS analysis.

4.2 Commitments

1. In the response to RAI-5 regarding TR Section 3.1, "AOO Acceptance Criteria," Westinghouse states that the reactor coolant system design pressure for ATWS event should not exceed the ASME Service Level C limit consistent with SRP Section 15.8, "Anticipated Transients Without Scram." In the same response (to RAI-5) Westinghouse commits to using a value of 120% design pressure when evaluating reactor pressure vessel integrity during an ATWS event.

2. In the response to NRR-RAI 43, the applicant committed to validate the assumed biases for the hot and average rods in Section 6.4, "Analysis Methodology," of the TR by performing code-specific confirmatory calculations on first application.

3. In the response to NRR RAI-34 S1, the applicant committed to use the generically approved CCA and DUA processes to determine the final list of both code and input uncertainty parameters specified on a plant specific basis which will be used in licensing applications.

5.0 CONCLUSION

Westinghouse presented a methodology of the EM for analyzing fast transients and ATWS events for first and reload cores. The methodology presented in WCAP-17203-P/ WCAP-17203-NP, Revision 0-2, for fast transients and ATWS events can be analyzed using 1D or 3D dynamic transient analysis computer codes as described in the SE. Using transient grouping and acceptance criteria for AOOs and ATWS events as per Chapters 4 and 15 of the SRP, a PIRT is created. The PIRT defines the phenomena which have to be addressed when

evaluating operating limits and safety margins to acceptance criteria. Once the operating limits and safety margins to acceptance criteria are determined for AOOs, uncertainty analysis is performed to evaluate the impact of uncertainties and biases on these limits in order to account for the uncertainty in the best-estimate result.

The NRC staff has reviewed WCAP-17203-P/WCAP-17203-NP, Revision 0-2, TR with respect to the transient grouping, acceptance criteria, PIRT parameters, transient analysis methodology, and uncertainty analysis, and determined that the methodology is acceptable for analysis of fast transients and ATWS events for BWR/2 through BWR/6 subject to limitations/conditions contained within this SE and the commitments made by the applicant listed in Section 4 of this SE.

6.0 REFERENCES

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2. Submittal of WCAP-17203-P/WCAP-17203-NP, Revision 0-2, “Fast Transient and ATWS Methodology,” South Texas Project, Westinghouse Electric Company, dated October 20, 2014 (ADAMS Accession No. ML14301A278).

3. CENPD-300-NP-A, “Reference Safety Report for Boiling Water Reactor Reload Fuel,” ABB Combustion Engineering, Inc., dated July 1996 (ADAMS Accession No. ML110260388).

4. U.S. Nuclear Regulatory Commission Regulatory Guide 1.203, “Transient and Accident Analysis Methods,” dated December 2005 (ADAMS Accession No. ML053500170).

5. U.S. Nuclear Regulatory Commission NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” dated March 2007.

6. *U.S. Code of Federal Regulations,* “General Design Criteria for Nuclear Power Plants,” 10 CFR Part 50, Appendix A.

7. U.S. Nuclear Regulatory Commission NUREG/CR-6744, “Phenomenon Identification and Ranking Tables (PIRTs) for Loss of- Coolant Accidents in Pressurized and Boiling Water Reactors Containing High Burnup Fuel,” September 2001 (ADAMS Accession No. ML013540628).
8. Letter from Scott Head (Nuclear Innovation North America LLC (NINA)) to U.S. Nuclear Regulatory Commission, U7-STP-NRC-110015, South Texas Project Units 3 and 4, “Response to Request for Additional Information” (RAI-1, RAI-2, RAI-3, RAI-4, RAI-5, RAI-8, RAI-9, RAI-10, RAI-11, RAI-12, RAI-15, RAI-16), dated January 18, 2011 (ADAMS Accession No. ML110200121).

9. Letter from Scott Head (NINA) to U.S. Nuclear Regulatory Commission, South Texas Project Units 3 and 4, “Response to Supplemental Request for Additional Information, (RAI-9 S01, RAI-17 S01, RAI-19 S01, RAI-20 S01, RAI-23 S01 through RAI-29 S01), dated June 13, 2011 (ADAMS Accession No. ML11166A048).

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11. Letter from Scott Head (NINA) to U.S. Nuclear Regulatory Commission, South Texas Project Units 3 and 4, “Response to Request for Additional Information” (RAI-6, RAI-7, RAI-17, RAI-18, RAI-23, RAI-24, RAI-27, RAI-29, RAI-30, and RAI-33), dated March 21, 2011 (ADAMS Accession No. ML110840711).

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Attachment: Comment Resolution Table

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Date: August 9, 2018