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Response to Request for Supplemental Information – ANP-10339P

ANP-10339Q1NP
Revision 0

Topical Report

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Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

List of Tables

Nomenclature

Acronym	Definition
AC	Alternating Current
ADV	Atmospheric Dump Valve
AFW	Auxiliary Feedwater
AOO	Anticipated Operational Occurrence
ARC	Automatic Rod Control (system)
ARITA	ARTEMIS / RELAP Integrated Transient Analysis
BOC	Beginning of (operating) Cycle
BWR	Boiling Water Reactor
CHF	Critical Heat Flux
CIPS	Crud Induced Power Shift
CSAU	Code Scaling, Applicability, and Uncertainty
CVCS	Chemical and Volume Control System
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling System
ED	Event Dependent
EM	Evaluation Model
EMDAP	Evaluation Model Development and Assessment Process
EOC	End of (operating) Cycle
EOP	Emergency Operating Procedure
ESFAS	Engineered Safety Feature Actuation System
FCM	Fuel Centerline Melt
FOGG	Feed Only Good (Steam) Generator
FWLB	Feedwater Line Break
GDC	General Design Criteria (10 CFR 50)
LCO	Limiting Condition for Operation
LOCA	Loss of Coolant Accident
LONC	Loss of Natural Circulation
LSCM	Loss of Subcooled Margin
MFW	Main Feedwater
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MSLB	Main Steam Line Break
MSS	Main Steam System
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient (of reactivity)
NI	Nuclear Instrumentation
NRC	(US) Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OPDT	Over-Power Delta-T (reactor trip) (Westinghouse plants)
OTDT	Over-Temperature Delta-T (reactor trip) (Westinghouse plants)
PA	Postulated Accident
PCI	(Fuel) Pellet/Cladding Interaction

Acronym	Definition
PIRT	Phenomena Identification and Ranking Table
PORV	Power Operated Relief Valve
PWR	Pressurized Water Reactor
PZRO	Pressurizer Overfill
RCCA	Rod Cluster Control Assembly (Westinghouse plants)
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPS	Reactor Protection System
SAFDL	Specified Acceptable Fuel Design Limit
SG	Steam Generator
SGO	Steam Generator Overfill
SGTR	Steam Generator Tube Rupture
SOTS	Symmetric Offset (reactor) Trip System (Combustion Engineering plants)
SRP	Standard Review Plan (NUREG-0800)
TCV	Turbine Control Valve
TGV	Turbine Governor Valve
TM/LP	Thermal Margin/Low Pressure (reactor trip) (Combustion Engineering plants)
TSV	Turbine Stop Valve
VOPT	Variable Over-Power (reactor) Trip (Combustion Engineering plants)

1.0 REQUEST NO. 1 FOR SUPPLEMENTAL INFORMATION

Reference 1 contains the following request for supplemental information regarding Topical Report ANP-10339P, Revision 0 (Reference 2):

The U.S. Nuclear Regulatory Commission (NRC) staff has identified that there is insufficient technical detail provided within Topical Report ANP-10339P pertaining to the process by which the six event classification phenomena identification and ranking tables (PIRTs) were developed, including all of the phenomena considered for each of the PIRTs, their associated rankings, and the justification for the rankings. This information is necessary for the NRC staff to focus its detailed review efforts in assessing adequacy of the proposed code system and evaluation models to each of the identified non-loss-of-coolant accident event categories in Chapter 15 of the Standard Review Plan (NUREG-0800). Provide information, preferably in a tabulated form, that presents all the phenomena considered across each of the six event-classification PIRTs, the associated ranking of each phenomena (i.e., non-applicable, low, medium, or high), and the justification for each ranking.

2.0 RESPONSE TO REQUEST NO. 1 FOR SUPPLEMENTAL INFORMATION

2.1 *Introduction*

The phenomena identification and ranking tables (PIRTs) for use in the NUREG-0800 (Reference 3) Standard Review Plan (SRP) Chapter 15, non- loss of coolant accident (LOCA) event analyses are described below. PIRTs were developed in support of the topical report ANP-10339P (Reference 2) as presented in Table 2-1 through Table 2-8. The PIRTs summarize the ranking and rationale for the ranking of the processes and phenomena identified for each applicable event category in the SRP. The highly ranked phenomena received the highest attention in terms of fidelity of modeling in the codes considered for the evaluation model (EM). An overview of the PIRT process is provided in Section 2.2 and Section 2.4.

2.2 *Overview*

The PIRT process originated as part of the Code Scaling, Applicability, and Uncertainty (CSAU) (Reference 4), as a cost-effective method to identify and select the processes and phenomena which dominate a certain transient behavior. In subsequent applications it has been recognized as a generalized application to support the decision-making processes.

The evaluation model development and assessment process (EMDAP) is discussed in Reference 5 and provides guidance for development and review of safety analysis EMs. While the development of the ARTEMIS / RELAP Integrated Transient Analysis (ARITA) methodology as a whole did not follow the EMDAP in the order or detail prescribed in Reference 5, the guiding principles of EMDAP were maintained.

The process of constructing a PIRT is outlined in Reference 5 as part of the EMDAP under Element 1, "Establish Requirements for Evaluation Model Capability." Element 1 of the EMDAP provides the application envelope for the EM by identifying and establishing the importance of the constituent phenomena, processes and key parameters within that envelope. There are four steps in Element 1 of EMDAP

(Reference 5, Figure 2). An overview of each of these steps and the main principles are reiterated below:

Step 1. Specify analysis purpose, transient class and power plant class – The analysis purpose is important since any given transient can be analyzed for different reasons. For instance, a given event may be analyzed to assess system overpressure or compliance with fuel-related acceptance criteria; thus, the stated purpose influences the subsequent EM development process. The transient class considers the non-LOCA events of SRP Chapter 15 with the purpose of complying with the General Design Criteria (GDC). The definition of the scenarios is dependent on the plant-class as dominant phenomena can vary with reactor design and plant-specific configurations.

Step 2. Specify the figures of merit – For the second step, the specific acceptance criteria described in the SRP for non-LOCA events are identified. Surrogate figures of merit can be used during the model development phase, given proper justification.

Step 3. Identify systems, components, phases, geometries, fields, and processes that should be modeled – Identification of EM characteristics is based on system hierarchical decomposition methods such as those used in the methods for scaling complex systems. Each hierarchical level can be decomposed into constituents at the next level down. A system can be divided into interacting subsystems; each subsystem into interacting modules, each module into interacting constituents, each constituent into interacting phases, each phase characterized by one or multiple geometrical configurations, each phase described by field equations (mass, momentum and energy), each field can be affected by several transport processes.

Step 4. Identify and rank key phenomena and processes – Development of the PIRT begins with the process identification in the previous step. The influence of each process and phenomena that occur varies greatly over the duration of a transient. After the identification has been completed, the processes and phenomena are assigned ranks based on their effect on the applicable figures of merit. The phenomena and processes that are highly ranked require greater modeling fidelity. The level of knowledge, or uncertainty regarding each highly ranked process or phenomenon should be assessed as part of the evaluation of the overall uncertainty applicable to the proposed EM with respect to appropriate figures of merit for the event (i.e., the acceptance criteria), or the limited assessment of biases and uncertainties used to determine guidance given as part of the EM for the use of "suitably conservative" input parameters for applications of the EM, if a complete uncertainty analysis is not required.

The entire development process is iterative in nature and can be repeated multiple times, with feedback loops connecting to the other development and assessment activities. Sensitivity studies can be used to help determine the relative importance of phenomena identified and for final validation as the EM development process is being iterated.

The final versions of the PIRT and EM are frozen for review purposes. The documentation of the PIRT is essential to provide the guidance to the subsequent steps of the development process. The PIRT determines the requirements for the model development, scalability, validation and sensitivity studies, and guides the uncertainty analysis and the assessment of the overall EM adequacy.

2.3 *Preparers and Review Panel*

The authors/preparers of the PIRT (those who provided the initial rankings) and PIRT peer/expert review panel members (those who suggested adjustments to the initial rankings, based on a consensus of opinions) were selected based on their technical expertise and background in safety analysis, core design, fuel performance, and methods and code development.

2.4 *Methodology/Approach*

A description of the process used to develop the PIRTs for the SRP Chapter 15 non-LOCA events supported by the ARITA methodology is as follows:

- The issue driving the need, e.g., licensing, operational, programmatic, was defined.
- Specific objectives were defined.
- Hardware and equipment scenarios were defined. A specific range of hardware configurations and specific scenarios were specified.
- Primary evaluation criteria were identified and defined. The primary evaluation criteria are the key figures of merit used to judge the relative importance of each phenomenon. The primary evaluation criteria used were derived from regulatory requirements for SRP Chapter 15 non-LOCA safety analyses acceptance criteria.
- System, components, phases, geometries, fields, and processes that must be modeled were identified. These are identified for the expected range of defined hardware and equipment scenarios.
- All plausible/consequential processes/phenomena were identified. A primary objective of this step was completeness. In addition to preparing the list of phenomena, precise definitions of each phenomenon were developed and documented.

- Importance ranking and associated rationale for each process/phenomenon were defined using a High (H), Medium (M), Low (L) ranking scheme. Processes/phenomena that are not applicable to a particular event or acceptance criterion are indicated by "N/A". The processes/phenomena that are ranked "H" or "M" are considered highly ranked. Importance was ranked initially by the preparers from each discipline relative to the primary evaluation criteria. The final ranking represents any adjustments needed to achieve consensus among the panels of discipline expert/peer reviewers to arrive at a single importance rank for a given phenomenon.

2.5 *Assumptions*

The following assumptions were made for the development of the PIRTs in Table 2-1 through Table 2-8:

- Event assessments apply to pressurized water reactors (PWR).
- Event assessments are not limited to a bounding event.
- Effects on dose consequences are based on considerations such as the effect a phenomenon may have on the degree of fuel damage or failure and/or steam releases to the atmosphere.
- The entire range of upstream conditions for safety and relief valve critical flows is considered, not just those conditions which are used to size the valves for event simulations.
- Specific codes, code sets and code coupling are not defined by the process used to develop the PIRTs.
- Plant operation is in accordance with established Technical Specifications.
- Operator actions are in accordance with established plant operating procedures.

2.6 *Non-LOCA PIRTs*

PIRTs for the SRP Chapter 15 non-LOCA events supported by the ARITA methodology are given in Table 2-1 through Table 2-8. The PIRTs provide rankings and rationales for the rankings for the identified processes and phenomena that potentially affect each of the supported events. PIRTs that address the nuclear steam supply system (NSSS) system response for each of the event categories are provided in Table 2-1 through Table 2-5. PIRTs related to core/subchannel thermal hydraulics, global and local core neutronics, and fuel rod responses to all categories of events are provided in Table 2-6, Table 2-7 and Table 2-8, respectively.

- SRP 15.1 Increase In Heat Removal by the Secondary System – This category of events results in reactor coolant system (RCS) overcooling with significant core power excursions prior to a reactor trip and the greatest potential for return to power after reactor scram. The power excursions challenge clad departure from nucleate boiling (DNB) and fuel centerline melt (FCM) criteria. Power excursions resulting from an increase in steam flow or excess load (SRP 15.1.3) event can challenge clad strain criteria for some plants and are among the more limiting events with respect to clad strain. Steam system piping failure events inside and outside containment (SRP 15.1.5) challenge radiological releases.

  The PIRT for this category of events relative to the NSSS system response in provided in Table 2-1.

- SRP 15.2 Decrease in Heat Removal by the Secondary System – This category of events results in RCS overheating and primarily challenges RCS and secondary system overpressure criteria. Some events can also challenge clad DNB and FCM criteria; however, this challenge is typically not as limiting as other classes of events since the decreases in RCS heat removal cause the primary side coolant to heat up as well as pressurize which tends to offset some of the decrease in critical heat flux (CHF) margin due to increased RCS temperatures. Since feedwater system pipe breaks (SRP 15.2.8) result in release of SG secondary fluid, breaks outside containment have radiological consequences. Although SRP 15.2.4 "Closure of Main Steam Isolation Valve" is identified in the SRP as a boiling water reactor (BWR) event, closure of one or more main steam isolation valves (MSIV) is nevertheless included in many PWR plant Chapter 15 evaluations; therefore it was included in the PIRT.

The PIRT for this category of events relative to the NSSS system response in provided in Table 2-2.

- SRP 15.3 Decrease in Reactor Coolant Flow Rate – This category of events primarily challenges clad DNB damage criteria, and to some extent RCS overpressure limit, as reductions in RCS flow cause the RCS coolant to heat up and expand prior to and immediately after the reactor trip. Since significant coolant flow is maintained prior to and immediately after the reactor trip for these events, and there is only a small increase in core power prior to trip if the plant Technical Specifications (TS) allow a positive moderator temperature coefficient (MTC) at beginning of cycle (BOC), the fuel rods are well cooled and the challenge to the FCM and clad strain damage criteria is mild. A small fraction of the fuel rods in the core may experience clad DNB for the reactor coolant pump (RCP) seized rotor (SRP 15.3.3) or shaft break (SRP 15.3.4) events, resulting in clad failure and radiological consequences.

  The PIRT for this category of events relative to the NSSS system response in provided in Table 2-3.

- SRP 15.4 Reactivity and Power Distribution Anomalies – With the exceptions of the single rod cluster control assembly (RCCA) withdrawal (SRP 15.4.3) event considered for Westinghouse plants, which is a postulated accident (PA) and therefore allowed to show a limited amount of clad failure due to undergoing DNB, the remaining events are anticipated operational occurrences (AOO) which are not allowed to result in fuel or clad damage (note, RCCA ejection, SRP 15.4.8, is outside of the scope of the ARITA methodology). Several AOOs challenge clad DNB as well as FCM criteria. The uncontrolled RCCA bank withdrawal events from subcritical (SRP 15.4.1) and at power (SRP 15.4.2) are typically among the limiting AOOs with respect to clad strain limits. Whereas significant core power overshoot may occur for Westinghouse plants operating with automatic rod control mode during a RCCA drop (SRP 15.4.3) event which may also challenge clad strain limits. The potential increase in power during startup of an inactive loop at an incorrect temperature (SRP 15.4.4) is also significant and may challenge clad strain limits. An inactive loop startup (SRP 15.4.4) event is the only asymmetric transient in this category with respect to the RCS coolant conditions. Single RCCA withdrawal (SRP 15.4.3) and improperly loaded fuel assembly (SRP 15.4.7) events can be asymmetric with respect to core power distribution. The power increase resulting from some control bank withdrawal events may be large enough for the RCS and secondary system pressures to rise significantly, although these events typically are less limiting than other overpressure events.

The Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (SRP 15.4.7) event is typically not analyzed as a transient but rather is evaluated based on steady-state power distributions. The Chemical and Volume Control System Malfunction that Results in a Decrease of Boron Concentration (SRP 15.4.6) event may challenge DNB and FCM limits in Mode 1 for some plants with automatic rod control systems as the regulating banks may insert as a result of the increasing core power and temperature from the boron dilution, resulting in increased radial and/or axial power distributions.

The PIRT for this category of events relative to the NSSS system response in provided in Table 2-4.

- SRP 15.5 and 15.6 Increase & Decrease in Reactor Coolant Inventory – LOCA (SRP 15.6.5) events and Radiological Consequences of Main Steam Line Failure Outside Containment (BWR) (SRP 15.6.4) are excluded from the PIRT since they are not considered to be non-LOCA events for PWRs. The remaining AOOs in SRP 15.5 and SRP 15.6, i.e., Inadvertent Operation of the Emergency Core Cooling System (ECCS) that Increases Reactor Coolant Inventory (SRP 15.5.1), Chemical and Volume Control System (CVCS) Malfunction that Increases Reactor Coolant Inventory (SRP 15.5.2), and Inadvertent Opening of a PWR Pressurizer Relief Valve (SRP 15.6.1), have rather limited impact on acceptance criteria. RCS depressurization resulting from a SRP 15.6.1 event challenges clad DNB criteria in the short-term and could result in release of steam and/or steam/water mixture to the containment if the event is not terminated by operator action. SRP 15.5.2 events can challenge clad DNB and FCM criteria for some plants via the changes in local core power distribution resulting from automatic control bank motion in response to changes in RCS boron concentration. SRP 15.5.1 and SRP 15.5.2 events increase RCS inventory and pressurizer level which can result in a significant increase in RCS pressure.

  The remaining two events, Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment (SRP 15.6.2) and Radiological Consequences of Steam Generator (SG) Tube Failure (SRP 15.6.3), are PAs which primarily challenge radiological consequences. SRP 15.6.3 results in the largest primary to secondary coolant leak of any SRP event, and has significant radiological consequences which are a function of system and human operator response to the event.

  The PIRT for this category of events relative to the NSSS system response in provided in Table 2-5.

Table 2-1
PIRT for SRP Category 15.1 NSSS System Response

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Table 2-2
PIRT for SRP Category 15.2 NSSS System Response

Table 2-3
PIRT for SRP Category 15.3 NSSS System Response

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Table 2-4
PIRT for SRP Category 15.4 NSSS System Response

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Table 2-5
PIRT for SRP Category 15.5 & 15.6 NSSS System Response

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Table 2-6
PIRT for SRP Category 15.1 to 15.6 Core/Subchannel Thermal-Hydraulic Response

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Table 2-7
PIRT for SRP Category 15.1 to 15.6 Core Neutronic Response

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Table 2-8
PIRT for SRP Category 15.1 to 15.6 Fuel Rod Response

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3.0 REFERENCES

1. Letter, Jonathan G. Rowley (NRC) to Gary Peters (Framatome Inc.), "Acceptance of Review of Framatome Inc. Topical Report ANP-10339P, 'ARITA – ARTEMIS/RELAP Integrated Transient Analysis Methodology' (EPID L-2018-TOP-0034)," December 18, 2018.
2. ANP-10339P, "ARITA – ARTEMIS/RELAP Integrated Transient Analysis Methodology". August 2018.
3. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," LWR Edition, March 2007.
4. NUREG/CR-5249 "Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident,"1989.
5. U. S. Nuclear Regulatory Commission NRC Regulatory Guide 1.203 "Transient and Accident Analysis Methods," 2005.