framatome

Response to Request for Additional Information – ANP-10339P

ANP-10339Q2NP
Revision 0

Topical Report

March 2020

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
AFW	Auxiliary Feedwater
ANS	American Nuclear Society
ANSI	American National Standards Institute
AO	Axial Offset
AOO	Anticipated Operational Occurrence
ARC	Automatic Rod Control
AREA	ARCADIA Rod Ejection Analysis
ARITA	ARTEMIS / RELAP Integrated Transient Analysis
ARO	All Rods Out
ASD	Appropriate Sampling Distribution
ASME	American Society of Mechanical Engineers
BAW	Babcock and Wilcox
BOC	Beginning-of-Cycle
BPV	Boiler and Pressure Vessel
CE	Combustion Engineering
CHF	Critical Heat Flux
CMA	Conservative Methodology Assumption
CSD	[ ]
CST	Condensate Storage Tank
CVCS	Chemical and Volume Control System
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DTC	Doppler Temperature Coefficient
ECCS	Emergency Core Cooling System
EM	Evaluation Model
EOC	End-of-Cycle
EPRI	Electric Power Research Institute
ESFAS	Engineered Safety Features Actuation System
FCM	Fuel Centerline Melt
FMA	Foundation Methodology Assessment

Acronym	**Definition**
FOM	Figure of Merit
FP	Full Power
FRR	Fuel Rod Response
FW	Feedwater
GCN	Global Core Neutronics
GE	General Electric
HCF	Hot Channel Factor
HEM	Homogeneous Equilibrium Model
HFP	Hot Full Power
HUM	Human Control System
HZP	Hot Zero Power
ID	Identifier (alphanumeric)
IET	Integral Effects Test
IOPORV	Inadvertent Opening of a Pressurizer Power Operated Relief Valve
ISF	Increase in Steam Flow
ITC	Isothermal Temperature Coefficient
LAR	License Amendment Request
LCN	Local Core Neutronics
LCO	Limiting Condition for Operation
LHGR	Linear Heat Generation Rate
LND	Log-normal Distribution
LOCA	Loss-of-Coolant Accident
LOCV	Loss of Condenser Vacuum
LOFT	Loss of Fluid Test
LOFW	Loss-of-Feedwater
LOOP	Loss-of-Offsite Power
LR	Locked Rotor
MDNBR	Minimum Departure from Nucleate Boiling Ratio
MFIV	Main Feedwater Isolation Valve
MFW	Main Feedwater
MRT	[ ]
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line

Acronym	Definition
MSLB	Main Steam Line Break
MSS	Main Steam System
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
N/A	Not Applicable
NEACRP	Nuclear Energy Agency Committee on Reactor Physics
NI	Nuclear Instrumentation
NRC	Nuclear Regulatory Commission
PA	Postulated Accident
PCS	Primary Coolant System
PDF	Probability Density Function
PDIL	Power Dependent Insertion Limit
PIRT	Phenomena Identification and Ranking
PORV	Power Operated Relief Valve
PSV	Pressurizer Safety Valve
PTD	Plant Transient Data
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
ROSA	Rig-of-Safety Assessment
RPS	Reactor Protection System
RSD	[ ]
RTD	Resistance Temperature Detector
RWST	Refueling Water Storage Tank
SAFDL	Specified Acceptable Fuel Design Limit
SBLOCA	Small Break LOCA
SBO	Station Blackout
SDBS	Steam Dump and Bypass System
SDM	Shutdown Margin
SEC	Secondary Coolant System
SET	Separate Effects Test
SG	Steam Generator
SI	Safety Injection

Acronym	**Definition**
SPERT	Special Power Excursion Reactor Test
SRP	Standard Review Plan
TBD	To Be Determined
TCS	Transient Cladding Strain
TCV	Turbine Control Valve
TG	Turbine Generator
TGV	Turbine Governor Valve
TH	Thermal Hydraulic
THAP	Thermal Hydraulic Additional Parameters
TIC	Time-in-Cycle
TS	Technical Specification
TSV	Turbine Stop Valve
UCBW	Uncontrolled Bank Withdrawal
US	United States
UTL	Upper Tolerance Limit
W3L	Westinghouse 3-Loop Plant
W4L	Westinghouse 3-Loop Plant

INTRODUCTION

Requests for additional information (RAI) related to the ARTEMIS / RELAP Integrated Transient Analysis (ARITA) Topical Report ANP-10339P are documented in Reference 1. Responses to these RAIs are provided herein. Markups to the ANP-10339P topical report to reflect any changes to the topical report due to the RAI responses will be provided at a later date.

1.0 RAI-1

Question:

ANP-10339P identifies specified acceptable fuel design limit (SAFDL) figures of merit and non- SAFDL figures of merit, but it does not identify the acceptance criteria for each. Please explicitly state the acceptance criteria for each SAFDL and non-SAFDL figure of merit, with justification. In the case where the limit is plant specific, state how that limit is determined for a given plant.

Response:

The ARITA non-Loss of Coolant Accident (LOCA) topical report (Reference 2) supports evaluation of SAFDL as well as non-SAFDL figures of merit. Figures of merit related to the SAFDLs pertain to the performance of the fuel rods and pellets for Anticipated Operational Occurrences (AOOs) and Postulated Accidents (PAs). SAFDL figures of merit addressed by the ARITA methodology include Departure from Nucleate Boiling Ratio (DNBR), Fuel Centerline Melt and Transient Cladding Strain (TCS). Reference 3 (Chapter 4.2) establishes the acceptance criteria for the SAFDLs. SAFDL figures of merit are evaluated for each plant application with Framatome fuel.

Non-SAFDL figures of merit relate to plant systems and components for AOOs and PAs. Non-SAFDL figures of merit supported by the ARITA methodology include Reactor Coolant System (RCS) and secondary side overpressure, Steam Generator (SG) overfill, pressurizer overfill, loss of subcooled margin and loss of natural circulation. Evaluation of non-SAFDL figures of merit is performed, as needed for a given application, in accordance with a plant's licensing basis.

Table 1-1 and Table 1-2 identify the acceptance criteria for SAFDL and non-SAFDL figures of merit, respectively, as applied to the ARITA methodology. Deviations from these criteria will be justified in a plant-specific submittal.

Table 1-1 SAFDL Acceptance Criteria Supported by the ARITA Methodology

Figure of Merit	Acceptance Criterion
Departure from Nucleate Boiling Ratio	The acceptance criterion for AOOs is the lower bound 95/95 DNBR must be greater than the DNBR correlation limit to prevent overheating the fuel rod cladding. Fuel failure is assumed if the lower bound 95/95 DNBR is less than or equal to the DNBR correlation limit for PAs. The method to calculate the DNBR correlation limit is discussed in Reference 2 (Section 9.1.4.4).
Fuel Centerline Melt	Fuel centerline temperatures and the fuel melt limits, dependent on fuel composition and burnup, are determined for each fuel rod type. The acceptance criterion for AOOs is the fuel rod specific upper bound 95/95 fuel centerline temperature must be less than the respective fuel melt limit to prevent overheating the fuel pellets. For PAs, fuel failure is assumed if the fuel rod specific upper bound 95/95 fuel centerline temperature exceeds the respective fuel melt limit. The method to calculate fuel pellet melt temperatures is shown in Reference 2 (Section 4.2.4.7.1).
Transient Cladding Strain	The allowed AOO transient-induced cladding strain increment is defined by a NRC approved limit by fuel cladding type. Consistent to the guidance in Reference 3 (Section 4.2), the strain criterion is defined as a transient-induced, uniform tangential deformation, elastic plus inelastic; steady-state creepdown and irradiation growth are excluded. Currently, the transient-induced cladding strain increment shall not exceed one percent for $M5_{Framatome}$ cladding up to rod average burnup of 62 GWd/mtU (Section 6.1.2 of Reference 4). For Zircaloy cladding, the strain limit is 1 percent [ ] (Section 6.1.2 of Reference 4). If the NRC approves the use of another limit for an existing or new cladding type, then that approved TCS limit is used for that cladding type. The method to calculate the TCS value using the GALILEO code is shown in Reference 2 (Section 4.2.4.7.1).

Table 1-2 Non-SAFDL Acceptance Criteria Supported by the ARITA Methodology

Figure of Merit	Acceptance Criterion
RCS Pressure	Acceptance criteria for RCS overpressure are established in the revision of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) code that is applicable to a given plant's licensing basis. The acceptance criterion used to demonstrate that the overpressure limits are met is the upper bound 95/95 RCS pressure must be less than 110% of design pressure for AOOs and 120% design pressure for PAs. Plant specific documentation defines the RCS design pressure. Plant documentation specifies the acceptance criteria applicable to events in their licensing basis.
Secondary Side Pressure	Acceptance criteria for secondary side overpressure are established in the revision of the ASME BPV code that is applicable to a given plant's licensing basis. The acceptance criterion used to demonstrate that the secondary side overpressure limits are met is the upper bound 95/95 secondary side pressure must be less than 110% of design pressure for AOOs and 120% design pressure for PAs. Plant specific documentation defines the secondary side design pressure. Plant documentation specifies the acceptance criteria applicable to events in their licensing basis. Typically, most plants have a single design pressure for the secondary side. Some plants, however, may specify multiple design pressures: one value for the SGs and another value for the main steam lines. For plant applications with multiple secondary side design pressures, either the more limiting of the design pressures would be used to establish the overpressure limits or separate overpressure limits would be determined, i.e., one set of limits for the SGs and another for the main steam lines. In the case of multiple design pressures, upper bound 95/95 SG pressure and upper bound 95/95 main steam line pressure would be compared against 110% (AOOs) and 120% (PAs) of the respective design pressures.
Steam Generator Overfill	Reference 5 identifies the potential for overfilling of the steam generators and introducing water in the steam lines. Steam generator overfill may result in a significant accumulation of liquid in the main steam lines which may produce loads beyond the design basis with failure of associated components. Plant emergency operating procedures and/or automatic functions

Figure of Merit	Acceptance Criterion
	exist to prevent steam generator overfill. Steam generator overfill can be challenged for events that are characterized by failure to isolate feedwater flow upon demand. The potential for overfilling the steam generator is compounded for a Steam Generator Tube Rupture event if failure in the feedwater system causes prolonged feedwater flow to the affected steam generator. Margin to steam generator overfill is defined as the difference between the total steam generator secondary side internal free volume and steam generator secondary side collapsed liquid volume. The acceptance criterion used to demonstrate that the steam generator does not fill is the lower bound 95/95 steam generator overfill margin must be greater than zero cubic feet. Maintaining positive margin verifies that operator actions and/or automatic functions are sufficient to prevent overfill.
Pressurizer Overfill	Preventing overfill of the pressurizer is related to the Reference 3 acceptance criterion that states: "An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently." Reference 7 provides additional discussion of this criterion. Pressurizer overfill may result in release of single-phase liquid through the pressurizer safety and relief valves which could cause mechanical failure in the affected valves. Such failure may result in substantial loss of RCS inventory effectively elevating an AOO to a more serious PA. Plant emergency operating procedures and/or automatic functions exist to prevent pressurizer overfill. Pressurizer overfill can be challenged for events that result in a heatup of the RCS, initiate the emergency core cooling system or initiate charging flow without compensating letdown flow. Margin to pressurizer overfill is defined as the difference between the total pressurizer internal free volume to the height of the safety relief valve inlet piping penetrations and pressurizer collapsed liquid volume. The acceptance criterion used to demonstrate that the pressurizer does not fill is the lower bound 95/95 pressurizer overfill margin must be greater than zero cubic feet. Maintaining positive margin verifies that operator actions and/or automatic functions are sufficient to prevent overfill. Preventing pressurizer overfill may not be applicable for some plants with pressurizer safety relief valves that are qualified for

Figure of Merit	Acceptance Criterion
	liquid release based on the Electric Power Research Institute (EPRI) Safety and Relief Valve Test Program in response to Reference 6 (Item II.D.1). For plants with such valves, valve inlet fluid thermal-hydraulic conditions during the time of single-phase liquid relief can be determined from the event analyses performed using the ARITA methodology. The valve inlet flow conditions during the time of interest can then be compared to the conditions of the EPRI tests. This comparison determines whether the plant conditions are bounded by or closely represented by the EPRI test data.
Loss of Subcooled Margin	Subcooled margin is a conservative surrogate figure of merit related to long-term core cooling for AOOs and PAs. Preventing a loss of subcooled margin maintains natural circulation when the Reactor Coolant Pumps (RCP) are not operating by precluding steam accumulation in the high points of the active flow path (e.g., upper region of the steam generator u-tubes). Preventing a loss of subcooled margin also precludes cavitation in the RCP impellers when the RCPs are in operation. Plant emergency operating procedures and/or automatic functions exist to prevent loss of subcooled margin. Loss of subcooled margin is typically most limiting in the RCS hot legs which are the hottest locations in the active RCS flow path. Loss of subcooled margin can be challenged by events that cause a heatup of the RCS resulting from degraded primary to secondary side heat transfer or events that severely depressurize the RCS. Subcooled margin is defined as the difference between local saturation and fluid temperatures along the active RCS flow path. The acceptance criterion used to demonstrate that subcooled margin is maintained is the lower bound 95/95 subcooled margin at the limiting location along the active flow path must be greater than zero degrees. Maintaining positive subcooled margin verifies that operator actions and/or automatic functions are sufficient to remove core decay heat and prevent void formation in the RCS coolant loops.

Figure of Merit	Acceptance Criterion
Loss of Natural Circulation	Like loss of subcooled margin, loss of natural circulation is a conservative surrogate figure of merit related to long-term core cooling that has historically been applied to more severe events such as a Station Blackout (SBO). Since SBO is currently outside of the scope of the NRC review of the ARITA methodology, loss of natural circulation figure of merit will be deleted.

2.0 RAI-2

Question:

ANP-10339P indicates that the ARITA evaluation methodology is intended for the analysis of PWR non-LOCA events identified in Chapter 15 of NUREG-800, the Standard Review Plan (SRP). Additionally, ANP-10339P discusses how the ARITA evaluation methodology is a composite of three variant evaluation methodologies (coupled EM, 0D EM, and static EM). However, not all of the Chapter 15 events are applicable for modeling by the ARITA methodology, and ANP-10339P does not clearly identify which of the variant evaluation methodologies will be used to analyze each of the Chapter 15 events that are applicable.

Provide a table showing every event that will be analyzed using the ARITA evaluation methodology and which of the three variant evaluation methodologies will be used for the analysis of each event.

Response:

The ARITA methodology will be used to analyze non-Loss-of-Coolant-Accident (non-LOCA) SRP events. Every SRP event that will be analyzed using the ARITA methodology is listed in RAI-9 Response Table 9-3. The evaluation model (EM) employed for an event analysis is determined by the criterion being analyzed and whether the event response is static or time-dependent. Analyses for events that evaluate Specified Acceptable Fuel Design Limits (SAFDLs) are performed using the Coupled EM or the Static EM (if the event is static), while the analyses of events that do not evaluate SAFDL criteria are performed using the 0D EM. The applicable EM for each SRP event is identified in RAI-9 Response Table 9-3, based on the typical criteria analyzed for each SRP event. For any event labeled in RAI-9 Response Table 9-3 that

[

]

3.0 RAI-3

Question:

Section 4.2.4.7.2, "Selection Process" on Page 4-36 of ANP-10339P states that

[ ] The discussion in this section also makes reference to [ ] and suggests these designs can be [ ] However, the discussion does not explicitly state any criteria for the justification of similar behavior. Provide clarification on the intent of this passage and identify what criteria are used to justify the behavior of [ ]

Response:

Page 4-36 of ANP-10339P (Reference 2) states that [ ]

The fuel rod selection process implemented in the ARITA methodology has the flexibility to include any desired fuel assembly and fuel rod type into the evaluation of fuel rod behaviors, i.e., fuel centerline melt and transient cladding strain. Typically, three batches of fuel assembly are irradiated in a core and the rod design in one particular batch, not counting the UO_2 enrichment, is the same. In some special circumstances, e.g., lead test assembly insertion, a new fuel rod design is introduced into the core.

The criterion used to determine if the behavior of a new fuel rod design is similar to an existing design is: Does the design of the new fuel rod or fuel pellet impact the margin to fuel centerline temperature or transient cladding strain limits? If there is an impact then these rods will be grouped separately.

Examples of changes to fuel pellet or fuel rod characteristics that could impact the fuel centerline melt or cladding strain are:

- [ ]
- [ ]
- [ ]
- [ ]

Examples of changes to fuel pellet or fuel rod characteristics that would not impact the fuel centerline melt or cladding strain are:

- [ ]
- [ ]
- [ ]

The above examples are intended to illustrate the use of the criterion used to determine if the behavior of a new fuel rod design is similar to an existing design but are not all encompassing.

4.0 RAI-4

Question:

Section 4.2.4.7.2, "Selection Process" on Page 4-35 of ANP-10339P and Section 4.2.4.7.3, "Calculation of FCM and TCS" on Page 4-37 of ANP-10339P respectively provide discussions on the selection of [ ] and on how fuel centerline melt (FCM) and transient cladding strain (TCS) are calculated using [ ] It is not clear that the selection process for [ ] guarantees the [ ] will be identified and used in calculation of FCM and TCS. Provide a detailed justification that a [ ] will not be identified in a full-core analysis versus [ ] Additionally, indicate whether Framatome has performed a validation of its selection process [ ] and if not, what analyses have been performed to validate the approach described in Sections 4.2.4.7.2 and 4.2.4.7.3 of ANP-10339P.

Response:

The fuel rod selection method presented in Section 4.2.4.7.2 of Reference 2 has considered [

]

The Increase in Steam Flow (ISF) sample problem presented in Section A.4 of Reference 2 considered [

] This comparison previously confirmed that the proposed fuel rod selection process is appropriate.

To further support the proposed fuel rod selection process in Section 4.2.4.7.2 of Reference 2, [

]

Table 4-1 and Table 4-2 below summarize the comparison [

] Therefore, the fuel rod selection method presented in Section 4.2.4.7.2 of Reference 2 is appropriate for [ ] for FCM and TCS calculations.

Table 4-1 Limiting Case FCM Margin for UCBW

[ ]

[ ]

Table 4-2 Limiting Case TCS Margin for UCBW

[ ]

[ ]

5.0 RAI-5

Question:

Section 4.2.4.7.4, "Calculation of TCS at Partial Power Conditions" on Page 4-37 of ANP-10339P indicates the [ ] The use of this type of distribution will result in [ ] Provide justification that use of [ ] is acceptable.

Response:

The sampling range of [ ] is selected to cover the typical partial power time. [ ] While the steady state depletion power history and the transient power history for a fuel rod are readily available from Neutronics codes, the rod axial power shape and conditioning time are more complicated. Plants tend to run at full power and limit the duration of reduced power operation. [ ] Therefore, the [ ]

The sampled conditioning time is fed into the fuel performance code to construct a complete fuel rod power history and evaluate the fuel performance during a transient.

[

]

This bias is discussed further in the response to RAI 6.

6.0 RAI-6

Question:

Section 4.2.4.7.4, "Calculation of TCS at Partial Power Conditions" on Page 4-38 of ANP-10339P indicates that, during the calculation of TCS at partial power conditions, the final TCS value [ ] The stated intent of this is conservatism, but the basis for the magnitude is not discussed. Provide a discussion on the basis for and determination of the [ ]

Response:

The determination of [ ] conservatism is based on [ ]

(a) [ ]

(b) [ ]

(c) [ ] and

(d) [ ]

[ ] The maximum difference for transient cladding strain is less than [ ] absolute for this range of partial power times.

A bounding, conservative evaluation was performed. [

]

The final calculated TCS value [ ] and then compared to the approved limit to account for potential variations in the actual part power operation.

7.0 RAI-7

Question:

Section 4.5 of ANP-10339P indicates the ARITA evaluation methodology makes use of the Wilks method, a non-parametric statistical approach, to determine a 95 percent probability with 95 percent confidence upper tolerance limit (95/95 UTL). Specifically, Section 4.5.1.1 discusses how the input probability distributions for the ARITA methodology [ ] The accuracy of the UTLs determined by the Wilks method is dependent upon representatively sampling the input probability distributions; independent NRC staff calculations indicate that [ ] may tend to underestimate the 95/95 UTL. Provide justification that the [ ] will not result in an adverse tendency to underestimate the 95/95 UTLs.

Response:

Distributions with infinite tails (e.g. the normal distribution) are [

]

Accurate representation of the [ ] Section 4.5.1.1 of ANP-10339P (Reference 2) explains that [ ] For distributions that have infinite tails (such as the normal distribution), [ ] In response to the concern raised in RAI-7 regarding the [ ] which is included as part of the response to RAI-9.

In general, [ ]

[ ]

Examples are provided below.

8.0 RAI-8

Question:

The statistical approach presented in ANP-10339P uses multiple figures of merit (FOMs) to demonstrate that fuel failure will not occur. The approach proposes demonstrating failure does not occur by showing that each FOM will be individually satisfied 95 percent of the time with a 95 percent confidence. However, demonstrating that each individual FOM is satisfied on a 95/95 basis would mean that the consideration of fuel failure from all possible sources would be less than 95/95. In other words, 95/95 assurance that the event is successfully mitigated would not exist by setting individual 95/95 criterion for each FOM when multiple FOMs are relevant to that event. Justify the adequacy of the reduced probability and confidence level associated with the proposed approach of using multiple FOM UTL statements or propose an alternative means of assuring a sufficiently high probability and confidence level (e.g., 95/95) of successfully mitigating an event (e.g., a simultaneous statement considering all possible sources of failure).

Response:

The ARITA statistical method as described in Section 4.5 of Reference 2 orders the results of a given case set separately for each figure of merit (FOM) of interest and identifies the rank associated with the 95/95 tolerance limit. Individually the assessments demonstrate that positive margin to fuel failure at the 95/95 level exists for that criterion. [ ] statement from the request for additional information "*However, demonstrating that each individual FOM is satisfied on a 95/95 basis would mean that the consideration of fuel failure from all possible sources would be less than 95/95.*" [

]

[ ]

A more detailed description of the method for [ ] is provided below.

The derivation begins with a discussion of what is being computed when a 95/95 upper tolerance limit is determined. First consider the case of a single variable X. An upper limit U^* is sought such that:

$$P(X \leq U^*) = 0.95 \qquad \text{Equation (1)}$$

Since U^* will depend on the unknown distribution of X (or on a known distribution that involves unknown parameters), U^* is replaced with a quantity U computed using data. The requirement in Equation (1) is modified to be:

$$P_{data}\left[P(X \leq U|data) \geq 0.95\right] = 0.95 \qquad \text{Equation (2)}$$

The brief details given above based on Equations (1) and (2) provide one explanation for the idea of an upper tolerance limit.

- In words the statement inside the square brackets in Equation (2) says: given the data, the probability that a random response drawn from the outcome space of the variable X will be less than or equal to the upper tolerance limit, U, is greater than or equal to 0.95.
- The probability that the data will guarantee such a requirement is to be equal to 0.95. That is, the probability that the statement in the square brackets in Equation (2) is true is to be equal to 0.95. Thus when we talk about a 95/95 upper tolerance limit, the first 95 refers to the distribution of X, and the second 95 (i.e., 95% confidence level) is with respect to the data.
- When the distribution of X is unknown, the requirement in Equation (2) will be satisfied conservatively (i.e., the confidence level will be larger than 95%), when U is obtained based on the order statistics from the data, provided:
    - An appropriately sized sample of results is randomly drawn from an outcome space.
    - The results are ordered from most to least severe for a given parameter.
    - The rank, a function of the sample size, associated with the tolerance limit of interest (here 95/95) is identified.

As an application of such an upper tolerance limit, suppose a desire exists to verify if the values of a random quantity X are below a specified limit (e.g. the number 10) with a high probability of 0.95. That is, we want to verify if:

$$P(X \leq 10) = 0.95 \qquad \text{Equation (3)}$$

When the distribution of X is unknown, or when it depends on unknown parameters, it is impossible to verify the requirement in Equation (3), since $P(X \leq 10)$ is now an unknown quantity. However, we can come up with a strategy based on data, and having an upper tolerance limit U based on data (U|data) provides one such strategy: simply verify if $U \leq 10$. It should however be noted that $U \leq 10$ is not equivalent to Equation (3), since Equation (3) itself cannot be verified unless the distribution of X is completely known (free of any unknown parameters), a very rare occurrence in practice.

The criteria associated with protection against fuel failure are:

- Departure from Nucleate Boiling (DNB)
- Fuel Centerline Melting (FCM)
- Transient Cladding Strain (TCS) where TCS is a consideration for AOO's.

In the application of the ARITA methodology it is not always necessary to include all of the fuel failure criteria in the demonstration of positive margin to fuel failure for a given event. An example of this is the complete loss of flow where FCM and TCS are not credible failure mechanisms given the characteristics of the event. As a result positive margin to fuel failure is demonstrated for the complete loss of flow event via assessment of MDNBR only. On the other hand, consider the uncontrolled bank withdrawal at power where, due to a variety of different potential conditions such as withdrawal rate, initial rod position, xenon condition, and so on, the event can have a large number of different behaviors. The behavior can range from large (slow or fast) changes in power when initiated from part power conditions to slow power increases that bring the core conditions just to the edge of the point where the reactor protection system intercedes. As a result, for this event, [

] The process for identification of the fuel failure criteria is informed by an assessment of the physical processes involved in the event and the effects of these processes on the conditions that are relevant for the fuel failure criteria.

[

]

9.0 RAI-9

Question:

ANP-10339P indicates the ARITA evaluation methodology is comprised of [ ] and that a non-parametric statistical approach is used to make a 95/95 UTL statement. But ANP-10339P does not clearly identify all of the uncertainties that will be sampled in the statistical approach, nor which uncertainties will be sampled [ ] ANP-10339P further does not identify in all cases the relevant uncertainty distributions and prescribed sampling ranges. Therefore, NRC staff cannot assess the completeness of the statistical approach or its adequacy. Provide a tabulated list of all the uncertainties being sampled in the non-parametric statistical approach and identify which will be used in each SRP Chapter 15 event analysis [ ] This table should identify the type of probability distribution for each uncertainty and the mean and standard deviation (or other relevant parameters as appropriate) that define the selected probability distribution. If the probability distributions are subject to change due to a dependent factor (e.g., fuel type or [ ]), or if plant-specific values may be used, provide a firm method for how the uncertainty parameters will be determined (e.g., from manufacturing tolerances, from plant specific information, etc.). In either case, justification should be provided for the chosen probability distribution, defining parameters, and prescribed sampling ranges. The table should also identify whether each sampled uncertainty is generically applicable or plant-specific.

Response:

A Phenomena Identification and Ranking Table (PIRT) for non-LOCA events, documented in Reference 9, identified processes and phenomena that dominate a certain transient behavior. The importance of each process and phenomenon was assessed for the pertinent figures of merit for each non-LOCA SRP event. Key parameters that can influence the dominant processes and phenomena are identified, assessed, [

]

Section 9.1 provides a summary of the processes and phenomena identified during the PIRT, supporting assessment data, and associated uncertainty parameters. Section 9.2 provides a detailed summary of the uncertainty parameter treatment and event initiator treatment.

9.1 *Assessment Data and Associated Model Uncertainty Parameters*

The PIRT, Reference 9, identifies processes and phenomena that dominate a certain transient behavior. The processes and phenomena identified in Reference 9 are listed in Table 9-1, along with pertinent tests and associated uncertainty assessment similar to that presented in ANP-10339P (Reference 2), Tables 8-2, 8-3, 8-4 and 8-5. Descriptions of the following test categories are provided in ANP-10339P, Section 8.0.

- Separate Effects Tests (SET)
- Integral Effects Tests (IET)
- Plant Transient (or Steady State) Data (PTD)
- Foundation Methodology Assessments (FMA)
- Conservative Methodology Assumptions (CMA)

For each PIRT process and phenomenon identified in Table 9-1, a list of [ ] The general parameter treatment identified in Table 9-1 is provided for cases where parameter modeling is needed for the EM. The more detailed parameter description contained in Table 9-2 should be consulted to fully identify the parameter treatment for the three EMs discussed in ANP-10339P, Section 4.0 - the Coupled, Static, and 0D EM.

[ ]

9.2 *Uncertainty Parameter Treatment*

9.2.1 Introduction

The ARITA methodology [

]

The following tables identify [

]

Table 9-2 and Table 9-3 describe [

]

Table 9-2 and Table 9-3 [ ] from the sample problems, documented in Appendix A of the ARITA topical report (ANP-10339P, Table A-4, A-5, A-11, A-12, A-20, A-21, A-29, A-30, A-37, A-38, A-44 and A-45). The sample problems provide [

]

Table 9-2 parameters are [ ] Additional information is provided in the notes following Table 9-2.

Table 9-3 lists the SRP events and [ ]

9.2.2 Treatment of General Parameters

Unless otherwise noted, [

]

[

]

The licensee [

]

Parameter Treatment

When determining the parameter treatment, the focus is to [

]

The rationale for parameter treatment is [

]

[

]

[

]

Table 9-2 and Table 9-3 identifies [

]

[

]

When [ ] is accounted for by [

]

9.2.3 Treatment of SRP Event Initiators and EM Application

Table 9-3 provides a listing of SRP events and [ ] The discussion provided for [ ] described in Table 9-3.

The anticipated EM employed for each SRP event is also listed in Table 9-3, based on the event-specific acceptance criteria. ANP-10339P, Table 4-2 provides an example of the acceptance criteria for each SRP event. When an event is only analyzed for non-SAFDL criteria, the analysis is performed using the 0D EM. When an event is analyzed for SAFDL criteria, the analysis is performed using the Coupled EM or the Static EM (if the event is static). When an event is analyzed for both SAFDL and non-SAFDL criterion, the analysis is performed using the Coupled EM.

The typical application of the EMs is shown in Table 9-3. For any event that has a SAFDL listed as a criterion in the SRP and that [

]

9.3 *Determining Parameter Treatment for an SRP Event*

The PIRT for non-LOCA events (Reference 9) identifies processes and phenomena that dominate certain transient behavior and ranks their importance for each non-LOCA SRP event and each analysis criterion of interest. Table 9-1 [

]

The EM employed for the event analysis is determined by the criterion being analyzed and whether the event response is static or time-dependent. When an event is only analyzed for non-SAFDL criteria, the analysis is performed using the 0D EM. When an event is analyzed for SAFDL criteria, the analysis is performed using the Coupled EM or the Static EM (if the event is static). When an event is analyzed for both SAFDL and non-SAFDL criterion, the analysis is performed using the Coupled EM. The applicable EM for each SRP event is identified in Table 9-3, based on the typical criterion analyzed for each SRP.

[

]

9.3.1 Example – Parameter Treatment for SRP 15.4.2 Uncontrolled Bank Withdrawal at Power Event Analysis for SAFDL Criteria

Table 9-1 Assessment Data and Associated Model Uncertainty Parameters

[ ]

[ ]

[ ]

[ ]

Table 9-2 Model Uncertainty Parameter Treatment

[ ]

[

]

[ ]

[ ]

[

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Table 9-3 SRP Event Initiator Uncertainty Parameter Treatment

[

]

[ ]

Table 9-4 Example: Sampled or Biased Parameters for SAFDL Analysis of SRP Event 15.4.2 Uncontrolled Bank Withdrawal at Power

[ ]

[

]

[

]

[

]

10.0 RAI-10

Question:

The statistical statement the ARITA methodology intends to demonstrate is not clearly articulated in ANP-10339P. However, in the post-submittal meeting held on May 8-9, 2019, Framatome clarified that the intended statistical statement is as follows:

> *For* [ ] *if this event occurred from within the licensed operating space, the limiting MDNBR (FCM, TCS, peak pressure, etc.) margin to the established limit is W (X, Y, Z, ...N) with 95% probability at 95% confidence.*

Confirm the italicized quotation above represents the statistical statement the ARITA methodology intends to demonstrate or provide appropriate modifications. Justify the intended statistical statement for the ARITA methodology is sufficient to demonstrate that a reactor is operating safely at all points within its allowed operating domain.

Response:

The italicized quotation does represent the statistical statement the ARITA methodology intends to demonstrate with one correction shown in bold type below:

> *For* [ ] *if this event occurred from within the licensed operating space, the limiting MDNBR (FCM, TCS, peak pressure, etc.) margin to the established limit is W(X, Y, Z, ...N) with* ***at least*** *95% probability at 95% confidence.*

Following industry advancements over the last 30 years, ARITA is a statistical methodology that addresses the analyses required to demonstrate compliance with NRC requirements and regulations for anticipated operational occurrences (AOOs), and postulated accidents (PAs). Statistical methodologies like ARITA sample the event outcome space and determine the results for the figures of merit that bound the majority of the outcome space with a high probability and confidence. The general design criteria (GDC) that are considered for the safe operation of a PWR are specified in Reference 20. [

]

[

]

In response to Regulatory Guide 1.157, the nuclear industry has developed multiple examples of realistic LOCA methodologies. [

]

Framatome believes that the ARITA statistical statement is sufficient for demonstrating that there is appropriate margin to the acceptance criteria for AOOs and postulated accidents based on the discussion in Regulatory Guide 1.157, the acceptance criterion for meeting the requirements of GDC 10 and 12 listed in SRP 4.4, and the NRC approval of multiple statistical methodologies that also use a 95/95 statistical statement. The response to RAI 11 provides additional discussion on the adequacy of the ARITA statistical methodology for demonstrating appropriate margin to the acceptance criteria.

11.0 RAI-11

Question:

[ ] deterministic and statistical evaluation models have generally selected bounding initial conditions and event definition parameters to ensure that a plant will satisfy acceptance criteria for analyzed events initiating at all postulated conditions in the permissible operating domain. [

]

Thus, Framatome's proposal to sample [

]

For example, [

]

Considering the discussion above, please justify sampling [

]

Response:

Following industry advancements over the last 30 years, ARITA is a statistical evaluation model. The response to RAI 10 describes the highly conservative nature of deterministic methods and the movement by both the NRC and the industry to more realistic statistical methods. Given the NRC's previous acceptance of statistical methods, and the language included in Regulatory Guide 1.157 (Reference 22) and SRP Section 4.4 (Reference 3), the response to RAI 11 will focus on the following two aspects of the ARITA statistical methodology:

[ ]

The ARITA evaluation model is a statistical model that evaluates the uncertainty in a conservative outcome domain to define a 95/95 tolerance limit. [

]

In conclusion, the ARITA methodology results in conservative 95/95 tolerance limits because the methodology samples from a conservative outcome domain. [

]

Therefore, the ARITA methodology is conservative for determining the appropriate margins to assure that the acceptance criteria are met for AOOs and PAs.

Table 11-1 Demonstration of Conservatism in the Post-Scram MSLB Sample Problem

[ ]

12.0 RAI-12

Question:

Section 4.5.1.5 of ANP-10339P indicates that if a non-parametric statistical analysis is performed where the outcome shows a failure to meet an acceptance criterion, then a reanalysis is performed with a [ ] and a different random sampling seed. NRC staff's concern with this approach is that it leaves open the question of whether a reanalysis that satisfies the acceptance criterion does so based upon the substance of the [ ] or merely the selection of a new random seed. Passing the acceptance criterion based primarily upon the selection of a new random seed would be inappropriate because it would effectively degrade the statistical confidence level. In past reviews, NRC staff found that reanalysis under such circumstances should reuse the original random sampling seed. Provide justification that selecting a new random seed when a reanalysis is performed will not degrade the statistical confidence level. Conversely, provide justification that when a reanalysis is performed wherein the acceptance criterion is met, it is the result of a [ ] and not the result of a more favorable seed.

Response:

The topical report will be revised to address the NRC staff's concern. If an analysis shows failure to meet acceptance criterion, any cases with failure in the original analysis must be rerun [

] Once this justification has been performed, the calculation [

]

The original analysis [ ] will be documented in an auditable record.

13.0 RAI-13

Question:

ANP-10339P does not identify the conditions for which a reanalysis of an operating plant's safety analysis of record would be performed (e.g., due to plant operation deviating from the parameter distributions used in determining the 95/95 UTL in the safety analysis of record). Given that it is possible for plant configuration and operation to deviate sufficiently from the underlying bases and assumptions of the safety analysis of record (e.g., including assumed uncertainty distributions) such that the analysis is no longer applicable, it appears appropriate to establish a method to account for this. What is the threshold at which the statistical analysis of record is no longer applicable and must be reassessed, and how is it monitored to determine whether a plant has crossed this threshold?

Response:

The extent to which the safety analyses support a plant's licensing basis can vary from plant-to-plant. All utilities are required to support their plant's Technical Specifications (TS) with a bases document. This document provides the technical reasoning, method, or calculation supporting a particular licensing limit or technical specification. Since safety analysis provides the strongest support for the licensing basis, the primary goal a licensee has with regard to the performance of safety analysis is to achieve coverage for the relevant limits of operation.

The response to RAI 9 provides a list of parameters considered by the ARITA methodology and identifies which parameters are biased and which are sampled as part of the uncertainty treatment. The basis for the distribution is provided for each sampled parameter. For many of the sampled parameters, Framatome requires the licensee to define the standard deviation and/or range of the uncertainty to be considered in the safety analysis. The uncertainty information and operating ranges are provided to Framatome along with the rest of the key input to the safety analysis in a formal design input transmittal such as a Plant Parameters Document.

The applicability of the analysis to support a plant's operating limits is the responsibility of the licensee. When the licensee makes a change that affects the input provided in the Plant Parameters Document, such as a change to a parameter uncertainty resulting from new instrumentation, the change requires disposition. Framatome supports these dispositions by evaluating the input changes contained in an updated Plant Parameters Document. If the change cannot be dispositioned, a calculation of the expected impact or a complete recalculation of the safety analysis is performed.

Similarly, the licensee is responsible for ensuring that the plant is operated within the assumptions of the safety analysis that support the licensing bases. If a plant has the potential to deviate from the assumed operation for an extended period of time, the licensee is responsible for informing Framatome so that the impact on the safety analysis can be assessed. This is true of existing safety analyses and is not a new or unique requirement for the ARITA methodology.

14.0 REFERENCES

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