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SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors

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AREVA NP Inc.

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Nature of Changes

Item	Page	Description and Justification
1.	All	Initial Release

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Nomenclature

Acronym	Definition
AO	Axial Offset
AOO	Anticipated Operational Occurrence
ASI	Axial Shape Index
BOC	Beginning of Cycle
CE	Combustion Engineering
COLR	Core Operating Limits Report
COLSS	Core Operating Limit Supervisory System
CPC	Core Protection Calculator
CVCS	Chemical and Volume Control System
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
HFP	Hot Full Power
HZP	Hot Zero Power
LAR	License Amendment Request
LCO	Limiting Condition for Operation
LOCA	Loss-of-Coolant Accident
LOEL	Loss of External Load
LSSS	Limiting Safety System Setting
MDNBR	Minimum Departure from Nucleate Boiling Ratio
MFW	Main Feedwater
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSSV	Main Steam Safety Valve
NRC	Nuclear Regulatory Commission
PDIL	Power Dependent Insertion Limit
PORV	Power Operated Relief Valves
PWR	Pressurized Water Reactors
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPS	Reactor Protection System
RV	Reactor Vessel

Nomenclature *(Continued)*

Acronym	**Definition**
SAFDL	Specified Acceptable Fuel Design Limit
SBLOCA	Small Break Loss-of-Coolant Accident
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SRP	Standard Review Plan
TT	Turbine Trip
VHPT	Variable High Power Trip

1.0 Introduction

The AREVA Topical Report EMF-2310 Revision 1 (Reference 1) describing the methodology for Standard Review Plan Chapter 15 Non-Loss-of-Coolant Accident (LOCA) events was previously approved by the Nuclear Regulatory Commission (NRC) for analyses supporting Pressurized Water Reactor (PWR) plant types designed by Combustion Engineering (CE)[a] and Westinghouse.

The purpose of this supplement is to augment Reference 1 to improve the rigor and comprehensiveness of the methodology and to address specific regulatory guidance delineated in the Standard Review Plan (SRP) (Reference 2). This supplement will be used in conjunction with the main body of Reference 1 to support our clients' intentions to alter their plants' licensing bases via a voluntary license amendment request (LAR). The extent of implementation of this supplement will be subject to the discretion of the Licensees on a case-by-case basis.

Specifically, this supplement provides additional information to address the following areas:

- Asymmetric Events
- Intermediate Power Levels
- Process Variables Biasing
- Fuel Thermal Conductivity Degradation with Exposure
- Rod Ejection Acceptance Criteria – Energy Deposition

[a] The CE plants that EMF-2310 Revision 1 and this supplement pertain to are the non-Core Protection Calculator/Core Operating Limit Supervisory System (CPC/COLSS) plant designs.

2.0 Asymmetric Events

This section specifically addresses SRP Chapter 15 events that progress to an asymmetrical reactor coolant system (RCS) response to the initiating event. These asymmetric events are those resulting from:

- a malfunction associated with one or more steam generators (SG)s that results in asymmetric core inlet *temperature*; (See Section 2.2.1)
- a malfunction associated with one or more reactor coolant pumps (RCP)s that results in asymmetric core inlet *flow*. (See Section 2.2.2)

[

]

2.1 *Identification of Events with Potential Asymmetric Core Inlet Conditions*

SRP Chapter 15 provides a categorization of transients and accidents by frequency of occurrence and by initiating event. Each of the non-LOCA events was evaluated to determine if asymmetric core inlet conditions could occur during the event. [

] Table 2.1 provides a summary of those events.

2.2 *Description and Categorization of the Asymmetric Events*

A summary of the categorization of the asymmetric events is provided in Table 2.2. [

]

2.2.1 Asymmetric Temperature Events

In Table 2.2, [

]

[

]

[

]

2.2.2 Asymmetric Flow Events

In Table 2.2, the Category 3 events could result in *flow* asymmetry at the core inlet. The primary system response germane to the event mitigation is the timing of reactor trip. For these events, the primary reactor protection system (RPS) signal is a flow reduction signal generated from equipment in the RCS loops. [

]

2.2.3 XCOBRA-IIIC [ ] for Flow Asymmetry Events

Core inlet flow profiles are typically generated by evaluating data obtained from vessel model hydraulic flow testing of a specific plant configuration. Vessel model testing is typically conducted by the original plant vendor at full and partial pump operation conditions. [

]

Table 2.1 Identification of Asymmetric SRP Non-LOCA Events

[ ]

Table 2.2 Categorization of Asymmetric SRP Events

[a] A non-sectorized core model may be used for evaluations of non-SAFDL criteria when justified.

Figure 2.1 XCOBRA-IIIC Core Inlet Flow Distribution Factors

3.0 Intermediate Power Levels

[

.]

[

]

4.0 Biasing of Process Variables

The purpose of this section is to define the biasing of input process variables for non-LOCA transient calculations which show compliance with [

] is adequately addressed in Reference 1, and will not be addressed further in this supplement.

4.1 *Process Variables to be Biased*

A set of the process variables used in event analyses is biased to assure that the results are conservative. A process variable is a measured plant condition that is input to plant observation and control systems. Process variables are also input to the RPS and to LCOs. These process variables undergo electronic processing to determine the measured plant state, which adds uncertainty to the measured value. In addition, some process variables are used as a target for a control system, and may have a control deadband associated with it.

The biasing of process variables [

]

[

]

The biasing of parameters described below also applies to the Steam Generator Tube Rupture (SGTR) event, SRP 15.6.3. The guidance herein augments the parameter biasing described in Reference 1, Section 5.5.

The biasing of parameters will be justified for each plant-specific application. The parametric biasing will be implemented as directed by each Licensee as appropriate and consistent with the plant's configuration and licensing basis. The justification for whether an upper bound or lower bound is utilized may be based on plant-specific sensitivity studies, generic sensitivity studies and/or engineering evaluations.

Section 4.2 discusses which events are typically analyzed to show compliance with the applicable criterion. Table 4.1 lists the process variables that will be considered for biasing. The biasing of these variables will be event specific.

4.2 *Typical Events Analyzed*

[

] This approach is described in Section 5.2.1 of Reference 1.

This section lists the events which are typically considered for each of the acceptance criteria. This list is provided to identify the types of events that may be limiting for each criterion.

4.2.1 [ ]

Table 4.2 lists the events which are typically considered to show compliance with the primary and secondary system [ ].

4.2.2 [ ]

Table 4.3 lists the events which are typically considered to show compliance with the pressurizer and steam generator [ ].

Table 4.1 Process Variables to be Biased

[ ]

Table 4.2 Events Typically Considered for [Over-Pressure Criterion]

[ ]

Table 4.3 Events Typically Considered for [Over-Fill Criterion]

[ ]

5.0 Treatment of Fuel Thermal Conductivity Degradation with Exposure

5.1 *Thermal Conductivity*

As discussed in Section 3.2 of Reference 1, the S-RELAP5 non-LOCA model uses user specified input for fuel thermal properties (thermal conductivity vs. temperature, volumetric heat capacity vs. temperature) for the average core and fuel hot spot heat structures. The core average heat structures are used to determine fuel temperature for Doppler feedback and average core fuel surface heat flux. The hot spot is used for evaluations of margins to fuel centerline melt for fast events (and slow events if desired), and more recently to obtain volume weighted average hot spot fuel temperatures input to clad strain evaluations for fast events.

This supplement modifies Section 3.2 of Reference 1 to change the source of thermal conductivity data from RODEX2 to COPERNIC, in order to include the effect of thermal conductivity degradation with fuel burnup. The fuel thermal conductivity input for both the average core and hot spot heat structures will be based on the following relations, which account for degradation of this parameter with exposure. These equations are used in the approved COPERNIC fuel performance computer code (Reference 3, Eq. 4-39, 4-40, 4-45, 4-46).

The thermal conductivity relationship for fully dense fuel is:

[ ]

[

]

The correction for porosity is a function of temperature as follows:

$$\lambda_{POR}^{Bu} = \lambda_{100}^{Bu} \cdot (1 - \alpha \cdot POR)$$
$$\alpha = 2.58 - 5.8 \cdot 10^{-4} \cdot T_c$$

Where

T_c : temperature (°C), and

POR : porosity fraction

The UO_2 thermal conductivity relationship is adapted to gadolinia fuels using the following modifications. The ratio of the thermal conductivity of gadolinia bearing fuel to UO_2 fuel [

] is:

[ ]

Where

z : Gadolinium content (weight%),

T_k : temperature (K), and

D is given by the following expression:

[ ]

The fraction of the core fuel rods bearing gadolinia is small enough to be ignored in determining the average core heat structure fuel thermal properties. Average core heat structure fuel thermal conductivity will be based on porosity and exposure values for a core average power UO_2 fuel rod at rated thermal power to an exposure appropriate for the time in core life under consideration in the S-RELAP5 analysis.

Hot spot heat structure fuel conductivity will be based on the type of fuel rod (UO_2 or Gadolinia bearing), for a conservative combination of porosity and exposure for the time in core life under consideration [

] Figure 5.1 illustrates the degradation of thermal conductivity as a function of exposure for pure UO_2 fuel using example data.

5.2 *Gap Conductance*

As noted in Section 3.2 of Reference 1, the gap conductance varies significantly throughout the cycle due to the effects of fuel temperature and exposure. For all fuel types, the fresh fuel gap conductance is low near the beginning-of-cycle, where a typical value for the fresh fuel gap conductance at full power is about 1200 Btu/hr-ft^2-°F, and increases due to various exposure related effects until the end-of-cycle, where the fresh fuel gap conductance at full power may be greater than 5000 Btu/hr-ft^2-°F. At higher powers, the fuel temperature increases, and both the contact pressure between the pellet and cladding and the gap conductance also increase.

The degradation of thermal conductivity with exposure will affect the fuel gap conductance due to the resulting changes in fuel temperature and consequent increase in fuel pellet thermal expansion, as well as changes in other temperature dependent parameters. Gap conductances reflecting thermal conductivity degradation are expected to be higher than those calculated by RODEX2 without consideration of the degradation.

The gap conductance of the average core heat structure(s) in S-RELAP5 will be selected to result in a conservatively low initial value for core average fuel temperature for the time in cycle, to minimize Doppler feedback during the event. As noted in Section 3.2 of Reference 1, [

]

The gap conductance of the core hot spot heat structure(s) in S-RELAP5 will be selected to result in a conservatively high initial value for fuel centerline temperature for the time in cycle, to reduce the margin to the fuel centerline melt limit. Variation of the hot spot gap conductance during the transient (when justified) will also be conservatively implemented to maximize the peak fuel centerline temperature during the event.

Figure 5.1 Example Thermal Conductivity Data as a Function of Exposure for Pure UO_2 Fuel

6.0 Rod Ejection Acceptance Criteria

6.1 *Fuel Energy Deposition*

Predicted peak radial average fuel enthalpy when calculated in accordance with this methodology shall remain below 200 cal/g.

6.2 *Failures*

Estimated fuel failures for radiological consequences will continue to be based upon exceeding the DNBR limit and Fuel Centerline Melt Temperature.

If fuel temperatures are predicted above incipient centerline melt conditions, the applicant's license will be reviewed to confirm that a conservative radiological source term was applied to the portion of fuel beyond incipient melt conditions and combined with existing gap source term (cladding failure shall be presumed).

6.3 *Burnup Dependence*

The Rod Ejection Analysis will evaluate the limiting conditions relative to cycle burnup.

7.0 References

1. EMF-2310(P)(A) Revision 1, *SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors*, Framatome ANP, May 2004.

2. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition", March 2007.

3. BAW-10231P-A Revision 1, *COPERNIC Fuel Rod Design Computer Code*, September 2004 (AREVA document 43-10231PA-01).

4. XN-75-21(P)(A) Revision 2, *XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation*, Exxon Nuclear Company, January 1986.

Distribution

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