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Revision 0
Supplement 1
Revision 0

PWR Small Break LOCA Evaluation Model, S-RELAP5 Based

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AREVA NP Inc.

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Nature of Changes

Item	Page	Description and Justification
1.	All	This is a new document. Information provided herein complements and replaces some of the information contained in Revision 0 of the present document.

Contents

Figures

This document contains a total of 36 pages

Nomenclature

Acronym	Definition
BOC	Beginning of Cycle
CCFL	Counter Current Flow Limitation
CE	Combustion Engineering
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
ECCS	Emergency Core Cooling System
EM	Evaluation Model
EOC	End of Cycle
HHSI	High Head Safety Injection
HPSI	High Pressure Safety Injection
LHSI	Low Head Safety Injection
LOCA	Loss-of-Coolant Accident
LPSI	Low Pressure Safety Injection
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature (reactivity) Coefficient
NSSS	Nuclear Steam Supply System
NRC	United States Nuclear Regulatory Commission
PCT	Peak Cladding Temperature
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
SBLOCA	Small Break Loss-of-Coolant Accident
SG	Steam Generator
SI	Safety Injection
SIT	Safety Injection Tank

1.0 Introduction

Reference 1 was originally submitted for NRC review in January 2000. Relative to the earlier Small Break Loss-of-Coolant Accident (SBLOCA) methodology, EMF-2328(P) replaced the set of three computer codes in References 2 and 3 (ANF-RELAP, RODEX2, and TOODEE2) with just two (S-RELAP5 and RODEX2A). Because of common aspects, EMF-2328(P) was regarded as an extension of the earlier methodology presented in References 2 and 3. NRC approval was granted March 2001 with the basis as presented in Reference 4. Modifications have been incorporated in accordance with annual 10 CFR 50.46 reports by AREVA NP Inc. (AREVA).

A collection of Errata was released in January 2008 to incorporate corrections to the BETHSY assessment in Section 5.0 (see Reference 5).

This supplement to EMF-2328(P), Revision 0 (Reference 1) provides additional modeling information on how the SBLOCA evaluation model (EM) will treat eight areas:

- Spectrum of break sizes,
- Core bypass flow paths in the reactor vessel,
- Reactivity feedback,
- Delayed reactor coolant pump (RCP) trip,
- Maximum accumulator / Safety Injection Tank (SIT) temperature,
- Loop seal clearing,
- Break in attached piping,
- Core nodalization.

Each issue is explained as to its treatment within the EM and the basis for that treatment, followed by a direct reference to any specific alteration of the treatment described in the main body of the Revision 0 topical. These changes are intended to improve the rigor and completeness of the original methodology.

2.0 Spectrum of Break Sizes

2.1 Summary

The break spectrum will include a wide enough range of break sizes to establish a clear trend in the peak cladding temperature (PCT) and to identify the limiting break size, from the smallest break that exceeds the capacity of the makeup system up to, and including, 10% of the cold leg area break, at which point the break spectrum will be covered by the Large Break LOCA EM (Reference 11). The break spectrum will be refined to identify both the break size where the evolution of the mitigating systems (pumped or passive injection) would determine where the transient is being turned over and the break size corresponding to the most limiting PCT. Criteria for spectrum density are provided below.

2.2 Include LHSI/LPSI

Because the spectrum includes larger breaks, the EM will include Low Head Safety Injection (LHSI) / Low Pressure Safety Injection (LPSI) boundary conditions for all SBLOCA analyses performed after the publication of this supplement.

2.3 Hot Leg Bend Angle

2.3.1 Physical Phenomena

2.3.2 Implementation

Figure 2.1 presents an example noding diagram showing the revised noding.

2.4 CCFL

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Figure 2.1 Example of Revised Primary System Nodalization

3.0 Core Bypass Flow Paths in the Reactor Vessel

3.1 Summary

A drawing of a representative recirculating steam generator PWR reactor vessel is provided in Figure 3.1. [

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3.2 Physical Processes

[ ]

3.3 Implementation

Figure 3.1 Typical Core Bypass Flow Paths

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Figure 3.2 Example of Reactor Vessel Noding Diagram [

]

Figure 3.3 Example of Upper Head Spray Nozzle

4.0 Reactivity Feedback

4.1 Summary

The SBLOCA model will incorporate reactivity feedback including both moderator and Doppler. The modeling of moderator reactivity feedback will include the entire spectrum of core conditions, from no voiding (all liquid water) to complete voiding (all steam). Reactivity representing the control rods will be introduced with an appropriate time delay at the time of reactor trip. If necessary, a reactor trip on high power may be added to the model.

4.2 Change from Previous Treatment

In EMF-2328 Revision 0, the only reactivity modeled represents control rod insertion upon reactor trip. This is a conservative simplification whenever the moderator reactivity insertion is negative. The MTC is "most" negative at the End of Cycle (EOC) and the bounding conditions used as a basis for the analysis (i.e., the most adverse top skewed axial power shapes) are associated with EOC conditions. However, it is possible for the high boron concentrations associated with 24-month cycle lengths to produce a Beginning of Cycle (BOC) MTC that is slightly positive at full power. The BOC condition will be used as a basis for moderator reactivity feedback. This represents a conservatively bounding (although inconsistent) combination of boundary conditions.

4.3 Physical Processes

Because water is not perfectly incompressible, a substantial decrease in pressure results in some expansion of the primary coolant, decreasing moderator density. Because commercial light water reactor cores are under-moderated, a decrease in fluid density results in negative reactivity and a power decrease. However, at BOC conditions with high boron concentrations, the boron concentration change may be high enough to overwhelm this effect. If the boron concentration is high enough, the decrease in the concentration of soluble poison will be the dominant effect and possibly introduce positive reactivity.

Increases in fuel temperatures prior to reactor trip produce negative Doppler reactivity and reductions in reactor power.

4.4 Implementation

In addition to other neutronics parameters supporting the analysis of a SBLOCA, AREVA will use tables of

- K_{eff} as a function of moderator density, and
- K_{eff} as a function of fuel temperature.

While it is only applicable to negative reactivity, in accordance with Section I.A.2 of Appendix K, shutdown reactivities resulting from temperatures and voids shall be given their minimum plausible values, including allowance for uncertainty.

When Technical Specifications (or the COLR) allow a positive MTC at full power, the maximum plausible value will be incorporated in order to allow an increase in core power prior to reactor scram.

5.0 Delayed RCP Trip

5.1 Summary

The Revision 0 EM assumes that RCP trip occurs only at the time of reactor scram, coincident with the loss of offsite power. (See the third paragraph in Section 3.2 on page 3-3 of Reference 1, as well as the fourth and fifth paragraphs in Section 6.1 on page 6-1 and the second row in Table 6.2 on page 6-9 of Reference 1.) As an additional set of cases in which offsite power continues to be available, delayed RCP trip will also be analyzed. This includes:

- review of current plant licensing basis, including the details of Emergency Operating Procedures , and
- separate analysis of Hot and Cold Leg break locations, considering different break sizes for each.

5.2 History/Background

Section II.K.3.5 of NUREG-0737 calls for the analysis of a delayed RCP trip for SBLOCA analyses. This was followed with specific recommendations for manual RCP trip in References 7 and 8 as NRC Generic Letters applicable to Westinghouse and CE plants. The US fleet responded to these requirements with a variety of analyses assumptions regarding timely RCP trip. For the analyses performed under this evaluation model, the justification for the pump trip action will mirror the existing basis with the exception that the EMF-2328 EM, including this supplement, will be used for any calculations and that hot leg breaks will be included.

5.3 Physical Processes

Under continued operation of the RCPs, the heat transfer in the core and primary side of the SGs is by forced convection. While this provides increased core cooling and heat removal, it can also contribute to increased depletion of the amount of liquid in the RCS. With RCP trip on scram, the RCS transitions to relatively stagnant conditions with a steam bubble forming in the top of the reactor vessel. The size of the steam bubble increases and gradually extends into the upper plenum, the hot legs and the SGs. When the break uncovers, the break flow changes from mostly liquid to predominantly steam, trapping a significant amount of liquid in the lower regions of the RCS. In contrast, keeping the RCPs running forces a mixture of steam and liquid to flow throughout the RCS throughout the transient. There is no extreme separation of the liquid and steam phases and no break uncovery time. The fraction of the break flow that is liquid is lower at first, improving system liquid inventory, but higher after the time the break would have uncovered if the pumps were not running. If the RCPs continue to operate, the RCS liquid inventory is gradually depleted far below that resulting when the RCPs are tripped early. So long as the RCPs continue to operate, the core will be effectively cooled. However, if the RCPs loose power or are tripped following the extra depletion of liquid inventory, there may not be enough liquid within the RCS to provide effective core cooling while the Emergency Core Cooling System (ECCS) is refilling the system. The key aspect is that after continued operation, if the RCPs trip, the sudden drop in fluid velocity in the core will produce a rapid separation of the liquid and steam phases in the core region.

As explained in the first paragraph of Section 3.1.1 of Revision 0, a break located in the cold leg will be limiting for the assumption of RCP trip at scram. However, for delayed RCP trip, other aspects (such as the distribution of liquid throughout the RCS) may become important enough that a break located in the hot leg piping becomes limiting.

With loss of offsite power at the time of reactor scram, the worst single-failure in Revision 0 is the loss of an emergency diesel generator (Reference 1, page 3-2 and page 6-1.) Since the continued availability of offsite power is necessary to supply power to the RCPs, for delayed RCP trip this changes slightly to (just) failure of an emergency electrical bus. The net effect is the same in terms of mitigating equipment assumed to be operable.

5.4 Implementation

Based on indications/conditions consistent with the plant licensing basis and Emergency Operating Procedures, if applicable, a spectrum of hot and cold leg breaks will be analyzed to support the RCP trip procedure. This spectrum may include a sensitivity on RCP trip time if such is required to support the trip procedure.

6.0 Maximum Accumulator/SIT and Refueling Water Storage Tank Temperature

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7.0 Loop Seal Biasing

7.1 Summary

In order to bound the possibilities discussed below, and to ensure a conservative evaluation, the S-RELAP5 based SBLOCA EM [

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7.2 Physical Processes

During a cold leg SBLOCA accident in PWRs with U-tube SGs, liquid can accumulate and form liquid plugs in the low elevation points of the loops such as the pump suction. An illustration of the typical loop seal configuration is provided in Figure 7.1. The water trapped in the primary loops between the SG and the RCP prevents the steam that is generated by core decay heat from escaping to the break. As a result, the steam pushes against the liquid interface in the vessel and the cross-over leg piping and keeps the break in the cold leg covered with liquid, thus (1) maximizing break flow and (2) depressing core liquid level. As steam continues to be produced and liquid is being lost out of the break, the space occupied by the steam expands and depresses the liquid level in the core and the downflow section of the cross-over piping.

The differential pressure in the loop will increase and reach a maximum as the core liquid level is depressed to its temporary minimum value corresponding to the spill-under elevation of the horizontal section of the loop seal piping.

The core liquid level depression may be aggravated by the amount of liquid holdup that can occur in the SG U-tubes, SG inlet plena, or the riser section of the hot leg piping.

Several interdependent hydraulic paths can be identified in a PWR with U-tube SGs. One of the hydraulic paths starts at the core and continues through the control rod guide tubes (and other open structures connecting the upper plenum to the upper head), to the upper head and through the downcomer to the upper head bypass flow path, if present and open at operating conditions (i.e., if designed as a distinct bypass path), and then to the downcomer and back to the core. The other hydraulic paths start at the core and follow the normal flow path through the hot legs, SG tubes, cold leg cross-over piping, pumps and back to the downcomer and the core. The situation is further complicated by the inability of any simulation to account for all the plant conditions that contribute to loop seal clearing phenomena. In a system without asymmetries the broken loop would tend to clear before the intact loops, due to the proximity to the break. This effect is small and may be overshadowed by the as-built plant differences. Density effects due to loop-to-loop temperature differences as little as 3 - 4 °F, can induce the preferential clearing of a given loop. Similarly, the actuation of the SG Main Steam Safety Valves (MSSVs) may drive the loop seal clearing in one or two loops over the other.

Through the examination of experimental data and analytical expectations for loop seal clearing, a trend can be established such that fewer loops will be clearing with decreasing break size. An evaluation of dominant causes considered that the strongest influence was likely the SG pressure (and MSSV operation) and its relationship to the primary side pressures and coolant temperatures. The second most dominant phenomenon is the break effect on local RCS pressure.

After the loop seal clearing, the core mixture level recovers as pressure imbalances throughout the RCS are relieved. At this point the break flow transitions from mostly liquid, prior to loop seal clearing, to mostly steam and some of the energy accumulated in the system can be evacuated through the break.

The corresponding momentary core level depression that occurs during the loop seal clearing period may generate a cladding temperature excursion. Because the excursion is naturally self-limiting, it does not, of itself, pose a serious concern to the plant, with the notable exception of the larger small breaks, where the timing of the momentary level depression may overlap with the onset of the second temperature excursion due to reduced core inventory. Nevertheless, the number of loops that clear, can have significant consequences during the core boildown period due to flow resistance to the break and residual liquid left in the loops.

7.3 Implementation

7.3.1 Loop Seal Biasing

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7.3.2 Loop Seal Nodalization Changes

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Figure 7.1 Typical Loop Seal Configuration for Westinghouse Plants

Figure 7.2 Loop Seal Clearing vs. Break Size in Test Data

8.0 Break in Attached Piping

8.1 Summary

The ECCS must cope with ruptures of the main RCS piping and breaks in attached piping. To accomplish this, an evaluation is made of the ruptures in attached piping that also induce a compromise of the ability to inject emergency coolant into the RCS. Frequently, the limiting case for these breaks is a rupture of the SIT or accumulator line because one SIT or accumulator and one LHSI system and perhaps part of the High Pressure Safety Injection (HPSI) system will be lost directly to the containment. When combined with a single failure, the ECCS capability is significantly compromised. In order to assure acceptable ECCS performance, the scope of analysis will include accidents of this type.

8.2 Background

The RCS (Class 1) primary coolant pressure boundary does extend from the cold leg piping (itself) to the closest check valve in the piping. Attached piping, branching in from the Accumulators or the SITs and pumped SI (both HHSI/HPSI and LHSI/LPSI), may inject through common lines or through separate injection lines depending of the plant design. Regardless of length, the full RCS pressure is applicable to these runs of piping and breaks should be postulated within these pipes.

Section 6 "Conclusions" of the NRC Safety Evaluation of EMF-2328 (enclosed with Reference 4) calls for additional assessments for analysis of break sizes larger than 10 percent of the cold leg flow area. While a full double-ended break in Accumulator or SIT piping may slightly exceed this limit, this break size is a special exception for this particular application.

8.3 Physical Processes

Breaks located in piping attached to the cold leg are special cases in that (in addition to the loss of RCS inventory) delivery of ECCS is adversely affected.

For a full double-ended break in Accumulator or SIT piping, one less (or "N" – 1) of these tanks will deliver flow to the RCS. The affected tank will "spill" directly to Containment instead.

For a break in pumped SI piping, a disproportionate fraction of flow generated by the system will spill out of the break to Containment pressure, significantly reducing delivery to the intact cold legs. The extent of the flow asymmetry depends on system design and is very plant-specific. For example, orifices and globe valves may have been incorporated in the piping system for the purpose of flow balancing.

8.4 Implementation

At least one node or volume will be modeled between the cold leg and the break location (typically) at a check valve.

As a conservative simplification, containment pressure may be assumed constant at atmospheric pressure. The change in containment conditions is not expected to be significant.

Instead of a spectrum of break sizes, a double-ended break in attached piping is assumed as the worst case for the loss of RCS inventory and the ECCS flow split, i.e., the fraction spilling to Containment pressure versus delivery to the RCS.

For breaks in pumped SI piping, if calculations based on bounding assumptions do not show acceptable results, supporting detailed piping flow resistance network calculations may be required.

9.0 Core Nodalization

9.1 Summary

Analyses performed using the this supplement [

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9.2 Revised Core Nodalization

9.3 Treatment of Hot Assembly Exit

10.0 Conclusions

Supplement 1 presents a reconsideration of the treatment of the models and methodologies governing:

- Spectrum of break sizes,
- Core bypass flow paths in the reactor vessel,
- Reactivity feedback,
- Delayed Reactor Coolant Pump (RCP) trip,
- Maximum accumulator/SIT temperature,
- Loop seal clearing,
- Break in attached piping,
- Core nodalization.

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All results of the supplement are considered to be part of the SBLOCA methodology of EMF-2328. For clarity, such analyses will be referred to as being conducted with the evaluation model documented in the EMF-2328, Revision 0 topical plus Supplement 1 to EMF-2328.

11.0 References

1. EMF-2328(P)(A) Revision 0 "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2001.

2. XN-NF-82-49(P)(A), Revision 1 "Exxon Nuclear Company Evaluation Model – EXEM PWR Small Break Model," July 1988.

3. XN-NF-82-49(P)(A), Revision 1 "Exxon Nuclear Company Evaluation Model Revised EXEM-PWR Small Break Model, Supplement 1," December 1994.

4. NRC letter dated 3-15-2001 "Acceptance for Referencing of Licensing Topical Report EMF-2328(P), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based" submitted by Framatome ANP Richland, Inc. (FRA-ANP) on January 10, 2000, and supplement dated January 26, 2001." (with Safety Evaluation attached) (accession number ML010800365)

5. EMF-2328(P) Revision 0 Errata "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," January 2008.

6. NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," US NRC, December 1988.

7. NRC Generic Letter 85-12 "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps," (for Westinghouse NSSSs) June 1985.

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9. BAW-10168Q-11 "RSG LOCA BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," October 1994.

10. EMF-2100(P), Revision 16, "S-RELAP5 Models and Correlations Code Manual," December 2011.

11. EMF-2103(P)(A), Revision 0 "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," April 2003.

12. Tasaka, K., et.al, "Loop seal clearing and refilling during a PWR small-break LOCA" Proceedings of the U.S. Nuclear Regulatory Commission Sixteenth Water Reactor Safety Information Meeting, NUREG/CP-0097 Vol. 4

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14. Kim, Yeon-Sik, Cho, Seok "An experimental investigation of loop seal clearings in SBLOCA tests" Annals of Nuclear Energy 63 (2014) 721–730

Record of Changes

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11	Page 11-1	Replaced

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Figure 2.1 presents an example noding diagram showing the revised noding.

Figure 3.2 Example of Reactor Vessel Noding Diagram [

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7.3 Implementation

7.3.1 Loop Seal Biasing

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7.3.2 Loop Seal Nodalization Changes

[ ]

11.0 References

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2. XN-NF-82-49(P)(A), Revision 1 "Exxon Nuclear Company Evaluation Model – EXEM PWR Small Break Model," July 1988.

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7. NRC Generic Letter 85-12 "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps," (for Westinghouse NSSSs) June 1985.

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11. EMF-2103(P)(A), Revision 0 "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," April 2003.