Attachment 3

Non-Proprietary Comments on the Draft SE for PWROG-17034-NP

Attachment 3						
Non-Proprietary Comments on the Draft SE for PWROG-17034-NP						
Comment #	**DSE Page No.**	**DSE Line No.**	**Comment Type**	**PWROG Comment**	**Justification**	**NRC Response**
1	2		Editorial	The Table of Contents, Section 3.3, the word "Eenergy" should be "Energy".	Misspelling	
2	18	After 25	Reference Clarification	Please add the following reference after No. 15: 16. Berkow, H.N. Director (U.S. Nuclear Regulatory Commission) to Gresham, J.A. (Westinghouse), "Acceptance of Clarifications of Topical Report WCAP-10325-P-A, 'Westinghouse LOCA Mass and Energy Release Model for Containment Design – March 1979 Version' (TAC No. MC7980)," October 18, 2005, ADAMS Accession No. ML052660242. This reference appears as Number 14 in PWROG-17034-P and it has been added to the DSE markup.	WCAP-10325-P-A was never revised. Therefore, any plant that referenced WCAP-10325-P-A in a licensing report that was submitted to the NRC, or has referenced it in their UFSAR since approximately 2006, has also included this additional reference.	
3	3	9	Reference Clarification	Please add the reference identified in Comment #2 above, after Reference 3.	See Comment #2 above.	
4	6	40 & 41	Editorial	Please delete ", and the" and replace it with "with" and then delete "in containment is".	These editorial changes improve the readability of the sentence.	
5	6	44 & 45	Editorial	Please delete "One" and replace it with "Two" and change "is" to "are" and delete "rate at which the" and insert "releases which provide the rate at which the primary system mass and the stored energy" after "M&E".	This captures the importance of the mass releases and the energy releases or enthalpy of the mass being released to the containment response codes.	
6	7	3	Clarification	Please insert the word "time-dependent"	This clarifies that this is a transient	

Attachment 3						
Non-Proprietary Comments on the Draft SE for PWROG-17034-NP						
Comment #	**DSE Page No.**	**DSE Line No.**	**Comment Type**	**PWROG Comment**	**Justification**	**NRC Response**
				between "the" and "containment".	value, not a static value.	
7	7	12	Clarification	Please delete the word "correct" and replace it with "higher".	WCAP-17721-P-A contains the ASME values that were current as of September 2015. The values are higher than the values that were available in the early 1970's when the base methodology documented in WCAP-10325-P-A was originally developed. These values could change in the future therefore the word "correct" is only relative to September 2015.	
8	7	19	Editorial	Please add "a" before the word "separate".	A word is missing.	
9	9	20	Clarification	Please add "the accumulators continue to inject and the pumped" before "safety injection".	This change adds the details of all of the sources of the reflooding fluid.	
10	9	34	Clarification	Please add "primary side" before the word "coolant", and then replace the words "to the generator" with "inside the steam generator tubes".	This change adds the details of the location of the fluid on the primary side of the RCS.	
11	10	12 through 25 and 45 & 46	For Information Only	There is no text change. This information reinforces the timing and the end point for the post-reflood phase that were the purpose of the revisions that were made to PWROG-17034-P/NP pages 2-2 and 2-3. Actual clarification text changes that are suggested for page 10 of the DSE appear in Comments 12, 13, 14, 15, 16, 17, 18, & 19. The post-reflood phase ends at 3600 seconds for WCAP-10325-P-A, when the entire	The transfer of the energy stored in the primary and secondary inventory and metal is accomplished in several stages as discussed in Section 2.3 of WCAP-10325-P-A. The removal of the stored energy in the broken loop steam generator [ ]^{a,c} is only one stage.	

Attachment 3						
Non-Proprietary Comments on the Draft SE for PWROG-17034-NP						
Comment #	**DSE Page No.**	**DSE Line No.**	**Comment Type**	**PWROG Comment**	**Justification**	**NRC Response**
				primary and secondary side are depressurized to saturated conditions at 14.7 psia. The FROTH code stops running when [[.]] For the 4-loop reference plant discussed in PWROG-17034-P/NP, the FROTH code terminates [[ ]]. The EPITOME code performs the remainder of the post-reflood phase for the removal of the stored energy from the RCS primary metal, and the broken loop and intact loop steam generator secondary metal and secondary inventory by depressurizing the entire primary and secondary sides to saturated conditions at 14.7 psia at 3600 seconds. EPITOME then continues with the long-term decay heat/steaming phase from 3600 seconds to at least 1 million seconds.		
12	10	12	Clarification	Please add "and intact loop(s)" between "broken loop" and "steam generator's" and delete "has" and replace it with "and the entire primary side of the RCS have"	Please see the information provided in Comment #11. The post-reflood phase in the WCAP-10325-P-A methodology ends at 3600 seconds with saturated conditions at 14.7 psia in the entire RCS.	
13	10	13 & 14	Clarification	Please delete "the containment design pressure" and replace it with "14.7 psia" and delete "in the broken loop steam generator".	Please see the information provided in Comment #11.	

Attachment 3						
Non-Proprietary Comments on the Draft SE for PWROG-17034-NP						
Comment #	**DSE Page No.**	**DSE Line No.**	**Comment Type**	**PWROG Comment**	**Justification**	**NRC Response**
14	10	15	Clarification	Please delete "steam generator" and replace it with "primary and secondary metal and fluid inventory"	Please see the information provided in Comment #11.	
15	10	16	Clarification	Please delete "broken loop" and add an "s" to generator to make it plural.	Please see the information provided in Comment #11.	
16	10	22	Clarification	Please add 'and intact loop(s)" after "broken loop".	Please see the information provided in Comment #11.	
17	10	23	Clarification	Please add an "s" to "generator" to make it plural, delete the "s" from "depressurizes", and delete "the containment design pressure" and replace it would "saturation at 14.7 psia".	Please see the information provided in Comment #11.	
18	10	25	Clarification	Please delete "broken loop steam generator" and replace it with "steam generators" and replace "has" with "have".	Please see the information provided in Comment #11.	
19	10	45 & 46	Clarification	Please delete "broken loop steam generator" and replace it with "entire primary and secondary side of the RCS" and also delete "the containment design pressure" and replace it with "saturated conditions at 14.7 psia".	Please see the information provided in Comment #11.	
20	16	7	Clarification	Please delete "for pressure and temperature reduction during a LOCA" and replace it with "and does not utilize residual heat exchangers and fan coolers for accident mitigation". Then please insert "heat removal equipment at" between "the" and "large dry".	These text changes add specifics for the heat removal systems for a sub-atmospheric containment design and a dry containment design.	
21	16	8	Clarification	Please delete the "s" from containments and "has" and replace this with "plants	These text changes further clarify the differences between a sub-	

Attachment 3						
Non-Proprietary Comments on the Draft SE for PWROG-17034-NP						
Comment #	**DSE Page No.**	**DSE Line No.**	**Comment Type**	**PWROG Comment**	**Justification**	**NRC Response**
				includes a spray system," before "residual" and add an "s" to "exchanger" and then insert "may also include safety grade" before "fan coolers". After "fan coolers", please insert "for pressure and temperature reduction during a LOCA. For sub-atmospheric containments," and then change the "t" on "the" to lower case.	atmospheric containment and dry containments that do not all have safety grade fan coolers.	
22	16	10	Editorial	Please make "M&E release" plural by adding an "s" to "release".	The overall break flow for mass and energy/enthalpy is generally plural.	
23	16	12	Editorial	Please add "the" between "is" and "same"	An article was missing.	
24	17	22	Editorial	Please add the Westinghouse transmittal letter number, date, and Accession Number that are associated with Reference 4, which are: LTR-NRC-15-81, September 24, 2015, Accession No. ML15272A202.	The document number, date, and Accession Number are missing.	
25	17	35	Editorial	Please revise the Accession No. for NUREG-0800, Section 6.2.1.3, Rev. 3 to ML053560191.	The Accession Number is incorrect.	
26	18	7	Editorial	Please add the PWROG transmittal letter number that is associated with Reference 12, which is: OG-19-134.	The document number is missing.	
27	18	17	Editorial	Please add the PWROG transmittal letter number and date that are associated with Reference 14, which are: OG-19-35, dated March 6, 2019.	The document number and date are missing.	
28	19	10	Editorial	Please correct the date to: "2019"	This is a typographical error.	

Attachment 4

Non-Proprietary Markups of the Draft SE for PWROG-17034-NP

DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT PWROG-17034-P, REVISION 0

"EVALUATION OF THE WCAP-10325-P-A WESTINGHOUSE

LOCA MASS & ENERGY RELEASE METHODOLOGY"

EPID: L-2018-TOP-0013

Enclosure

TABLE OF CONTENTS

Energy

CONTENTS

and Ref. 16

1.0 INTRODUCTION

By letter dated March 29, 2018 (Ref. 1), the Pressurized Water (PWR) Reactor Owners' Group (PWROG) submitted Topical Report (TR) PWROG-17034-P, Revision 0, "Evaluation of the WCAP-10325-P-A Westinghouse LOCA [Loss-of-Coolant Accident] Mass & Energy [(M&E)] Release Methodology" (Ref. 2) to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. The purpose of this report was to justify the continued use of the previously approved LOCA mass and energy (M&E) analysis methodology contained in WCAP-10325-P-A (Ref. 3). This justification was needed as the properties of the reactor coolant system (RCS) material used in WCAP-10325 were found to have non-conservatively lower values compared to the current American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. In order to justify the continued use of WCAP-10325, the PWROG performed sensitivity analysis with and without changing material properties, and by comparing the resulting M&E and containment response analysis with the analysis of another NRC approved M&E methodology which did have the correct material properties, WCAP-17721-P-A (Ref. 4). The NRC submitted its acceptance of the TR for a complete review by letter dated May 9, 2018 (Ref. 9).

The PWROG letter dated March 29, 2018, was supplemented by letter dated July 9, 2018 (Ref. 10), which provided changes in pages 2-2 and 2-3 of the TR, and by letter dated July 17, 2019 (Ref. 12), which provided responses to the NRC staff request for additional information (RAIs).

This comparison and sensitivity analysis demonstrated that despite the non-conservative material properties used, WCAP-10325 remains a conservative LOCA M&E analysis methodology.

A complete list of the key correspondence between the NRC and the PWROG is provided in Table 1 below.

Table 1: List of Key Correspondence

Sender	Document	Document Date	Reference
PWROG	Submittal Letter	March 29, 2018	1
PWROG	Topical Report	March 29, 2018	2
NRC	Acceptance Letter	May 9, 2018	9
PWROG	Supplemental Changes to Pages 2-2 and 2-3 Letter	July 9, 2018	10
PWROG	Presentation Cover Letter	March 6, 2019	14
PWROG	Presentation Slides (Attachment 1)	January 30, 2019	14
PWROG	RAI Response Letter	July 17, 2019	12
PWROG	RAI Responses	July 17, 2019	13

2.0 REGULATORY EVALUATION

A licensee will use a variety of methods to evaluate the transients and accidents that could occur at its nuclear power plant. The NRC staff reviews these methods to ensure that they provide a realistic or conservative prediction such that it can be demonstrated that the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) can be satisfied.

Additionally, because the results of the transient and accident analysis methods are important to the safety of nuclear power plants, these methods must be maintained under a quality assurance program which meets the criteria set forth in 10 CFR Part 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

Other regulations which are applicable to transient and accident analysis methods are:

- 10 CFR 50.34, "Contents of Applications; Technical Information," which provides the requirements for the Final Safety Analysis Report required for each plant which includes the analysis of transients and accidents;

- 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," which provides the requirements for a LOCA analysis;

- 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," which provides the requirements to establish the temperature and pressure at the location of the electric equipment important to safety for the most severe design basis accident during or following which this equipment is required to remain functional;

- 10 CFR Part 50 Appendix A, "General Design Criteria [(GDC)]," which includes the principal design criteria for the facility;

- 10 CFR Part 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," which provides the definition of containment leakage test pressure and the leakage test requirements; and

- 10 CFR Part 50 Appendix K, "ECCS Evaluation Models," which provides further requirements for a LOCA analysis.

While these regulations are broadly applicable to containment analysis, there are three GDC in 10 CFR Part 50 Appendix A specifically for containment analysis: GDC 16, GDC 38, and GDC 50.

- GDC 16, "Containment design," requires the containment design to be essentially leak tight and that the containment design conditions important to safety are not exceeded for as long as the postulated accident conditions require.

- GDC 38, "Containment heat removal," requires that a system to remove heat from the containment must be provided. The safety function of the system shall rapidly reduce, consistent with the functioning of other systems, the containment pressure and temperature following any LOCA, and it should maintain the pressure and temperature at acceptable levels.

- GDC 50, "Containment design basis," as it relates to the containment and subcompartments being designed with sufficient margin, requires that the containment and its associated systems can accommodate, without exceeding the design leakage rate, and the containment and subcompartment design can withstand the calculated pressure and temperature conditions resulting from any LOCA.

Licensees perform simulations to demonstrate that these criteria have been met, and as part of their regulatory oversight the NRC staff will review these simulations. To assure the quality and uniformity of NRC staff reviews, the NRC created NUREG-0800, "Standard Review Plan [(SRP)] for the Review of Safety Analysis Reports for Nuclear Power Plants" (Ref. 5) to guide the staff in performing their reviews. Regulatory guidance for containment reviews is provided in Section 6.2.1 of the Standard Review Plan (SRP), "Containment Functional Design" (Ref. 6).

The focus of this safety evaluation (SE) is the simulation which predicts the M&E released into containment as a result of large break LOCA. The guidance for such M&E reviews is given in Section 6.2.1.3 of the SRP, "Mass and Energy Release Analysis for Postulated Loss-Of-Coolant Accidents (LOCAs)" (Ref. 7). Similar guidance was also published in the American Nuclear Society's Standard ANS 56.4, "pressure and temperature transient analysis for light water reactor containments" (Ref. 8).

WCAP-10325 has previously been approved by the U.S. NRC and much of that review will not be repeated here. This review will focus solely on the impact of the use of a non-conservative material property value, which has a direct impact on the stored energy of the metal in the RCS. Table 2 provides the criteria from 10 CFR, the SRP 6.2.1.3 and, ANS guidance, associated with this stored energy. While each criterion is worded slightly differently, all sources agree that the correct stored energy must be taken into account when performing the M&E analysis.

Table 2: Criteria for Stored Energy

Heat transfer from piping, vessel walls, and non-fuel internal hardware shall be taken into account. 10 CFR Part 50, Appendix K, Paragraph I.A.6
Sources of Energy. The sources of stored and generated energy that should be considered in analyses of LOCAs include: reactor power; decay heat; stored energy in the core; stored energy in the RCS metal, including the reactor vessel and reactor vessel internals; metal-water reaction energy; and stored energy in the secondary system (PWR plants only), including the steam generator tubing and secondary water. *Calculations of the energy available for release from the above sources should be done in general accordance with the requirements of 10 CFR Part 50, Appendix K, paragraph I.A. However, additional conservatism should be included to maximize the energy release to the containment during the blowdown and reflood phases of a LOCA. An example of this would be accomplished by maximizing the sensible heat stored in the reactor coolant system (RCS) and steam generator metal and increasing the RCS and steam generator secondary mass to account for uncertainties and thermal expansion.* SRP Chapter 6.2.1.3 Sub-Section II.1.
The core stored energy and the steady-state core temperature distribution, adjusted for uncertainties, shall be consistent with the initial conditions and consistent with the time of fuel cycle life required in 3.2.2.1. ANS 56.4, 3.2.1.3 Core Stored Energy

3.0 TECHNICAL EVALUATION

The TR PWROG-17034-P provides the justification for the continued use of WCAP-10325 to perform M&E release analysis following a LOCA for large dry and sub-atmospheric containment designs. Specifically, the TR provides a summary of two different approved M&E methodologies: WCAP-10325 and WCAP-17721. It discusses the non-conservative way WCAP-10325 treats some material properties and demonstrates that due to an overly conservative treatment of heat transfer, WCAP-10325 remains conservative and therefore acceptable for M&E analysis. To better organize the NRC staff's review, the technical evaluation has been separated into the following sections:

3.1 Background Information on M&E Release Analysis
3.2 Differences between WCAP-10325 and WCAP-17721
3.3 Evaluation of WCAP-10325 M&E Release and Containment Analysis
3.4 Evaluation of Containment Response Using WCAP-10325/COCO Code

The first section will discuss background information including containment analysis in general, what M&E analysis is and why it is important in general, and the different phases of the LOCA event from an M&E perspective.

The second section will discuss Westinghouse's approved M&E analysis methods contained in TRs WCAP-10325 and WCAP-17721, including the similarities and differences between the methods and their modeling of M&E release.

The third section will evaluate non-conservative use of material properties and demonstrate that even with this non-conservatism, WCAP-10325 remains a conservative analysis method due largely to the conservative heat transfer rate applied.

3.1 Background

Following a large-break LOCA, hot fluid from the RCS will flow into containment. This hot fluid will increase the pressure and temperatures in the containment and this increase could challenge the containment in the following two ways:

(1) high pressure in containment could cause excessive leakage and challenge the containment integrity if it exceeds its design pressure, and

(2) high pressure and temperature could create excessively harsh conditions such that any safety-related equipment in containment may fail to operate.

with

To ensure that LOCAs do not challenge the containment integrity or the equipment ~~in~~ containment, containments are designed to remove heat and reduce the pressure~~, and the~~ equipment ~~in containment is~~ designed to operate under harsh conditions. To demonstrate that the containment has been adequately designed, licensees must perform an analysis which simulates the conditions in containment following a LOCA. ~~Westinghouse performs~~ such containment analysis using specific containment codes. ~~One~~ of the most important inputs to these codes is the ~~rate at which the~~ M&E from the primary coolant, primary metal, fuel, secondary coolant, and secondary metal enters the containment from the RCS.

Two

are

releases which provide the rate at which the primary system mass and the stored energy

time dependent

It should be stressed that WCAP-10325's non-conservative volumetric heat capacity (density x specific heat) of the RCS material impacts the total energy stored in the primary and secondary side. However, while that total energy must be transferred to containment, the containment pressures and temperatures are not dependent on the total energy, but on the rate at which that energy is transferred. For example, the primary and secondary metal could contain an infinite amount of energy and if that energy was transferred to containment at or below the rate at which the energy is removed, the containment would not see an increase in temperature or pressure.

The focus of this SE is not on the difference between the initial stored energies in WCAP-10325 (which has the non-conservative volumetric heat capacity of the RCS metal) and WCAP-17721 (which has the ~~correct~~ volumetric heat capacity of the RCS metal), but on the difference between their M&E transfer ~~rates~~. higher

Calculating this M&E transfer following a large break LOCA includes simulating several phases following the initial break. Each phase (blowdown, refill[1], reflood, post-reflood, and long-term steaming) along with its approximate starting time, when it is defined to start, and when it is defined to end, and is given in Table 3 below. Note that WCAP-10325 or WCAP-17721 both simulate the first four phases of the LOCA and separate calculation is used for the long-term steaming analysis.

a

[1] The refill phase is often conservatively ignored as it represents a time of minimal M&E transfer into containment.

Table 3: Phases following a Large-break LOCA

Phase	Start Time (Seconds)	Starts When	Ends
Blowdown	0	When the break initiates.	When the RCS pressure and the primary reactor containment pressure are virtually equal.
Refill	20 - 30	When the RCS pressure and the primary reactor containment pressure are virtually equal.	When the emergency core cooling system (ECCS) refills the reactor vessel to the bottom of the active core.
Reflood	30 - 50	When the ECCS refills the reactor vessel to the bottom of the active core.	When the liquid level in the core reaches a height sufficient to essentially terminate liquid entrainment in the core.
Post-Reflood	150 - 200	When the liquid level in the core reaches a height sufficient to essentially terminate liquid entrainment in the core.	When the temperature of the RCS and the steam generators are essentially equal.
Long Term Steaming (or Decay Heat)	3600	When the temperature of the RCS and the steam generators are essentially equal.	When the RCS and steam generators are in thermal equilibrium with the surroundings.

This SE will focus on all phases of LOCA mentioned in Table 3.

3.2 Differences between WCAP-10325 and WCAP-17721

While both methodologies in WCAP-10325 and WCAP-17721 have been reviewed and approved by the NRC for simulating the M&E release following a large break LOCA, their modeling approaches are different. This difference is mostly the result of the over 30 years difference in when the methodologies were developed. WCAP-10325 was approved in 1983, and WCAP-17721 was approved in 2015 and therefore contains a significant number of advances due both to increased understanding of the phenomena and increased computational capabilities. It should also be noted that WCAP-17721 does contain the appropriate volumetric heat capacity values.

Blowdown

In modeling the first phase of the scenario, the WCAP-10325 method results in slightly higher predictions of the M&E release rate which impacts the blowdown pressure peak. This is

obvious from Figures RAI-2-B-1 and RAI-2-B-2 for time between 0 - 20 seconds in Reference 13.

Refill

Refill is modeled differently between the two methods. WCAP-10325 conservatively assumes that the lower plenum is instantaneously refilled with water from the accumulators while WCAP-17721 models this refill period. By assuming that the lower plenum is instantaneously refilled, WCAP-10325 increases the rate at which M&E enters the containment. As the pressures and temperatures in containment are dictated by the rate at which M&E enters containment versus the rate at which energy is discharged from containment, increasing the rate at which M&E enters containment will result in conservative (i.e., higher than expected) containment pressures and temperatures. Figure RAI-2-B-2 provided by the PWROG in response to RAI-2 (Ref. 13) clearly indicates conservatism in the WCAP-10325 analysis versus WCAP-17721 analysis in the refill phase between 20 - 30 seconds.

Reflood

the accumulators continue to inject and the pumped

Reflood is modeled somewhat differently between the two methods. During the reflood phase, safety injection fluid enters the reactor vessel. This fluid flows into the core where it creates a two-phase frothy mixture called a quench front. The quench front will travel from the bottom to the top of the fuel over a certain time (e.g., 100 seconds). During that time, the fuel below the quench front is covered by water, is relatively cool, and the fluid next to the fuel is single phase water. The fuel above the quench front is uncovered, relatively hot, and the fluid is steam with droplets. The fuel in the quench front is a mixture of covered and uncovered, and the flow is two-phase mixture.

primary side

inside the steam generator tubes

While the decay heat in the core is one contributor to the energy transferred to containment, another large contributor is the energy stored in the broken loop steam generator. While the RCS has been depressurized, the steam generators are assumed to have been "bottled-up" at their operating pressures and are therefore very significant sources of energy. While there may be 2, 3, or even 4 steam generators, it is primarily the steam generator of the broken loop which interacts with the RCS coolant, as there is minimal driving force to send the RCS coolant into any other generator. The only coolant available to ~~the generator~~ is the steam and droplets from the core. One of the conservatisms in the WCAP-10325 methodology is to assume that the fluid exiting the steam generator is saturated, meaning that any energy from the steam generator must first evaporate all the liquid droplets entering the steam generator. By not allowing superheat of the steam, and instead evaporating the droplets, WCAP-10325 conservatively predicts more steam entering containment than expected, which results in a higher containment pressure.

While WCAP-17721 used more mechanistic methods, the difference between the two methodologies in the energy transfer rate in containment, and hence resulting pressure and temperature was not very significant during the reflood phase, as demonstrated in Figure RAI-2-B-2 (Ref. 13) and Figures 4.2-1 and 4.2-2 (Ref. 2). The small difference is due to the relative short time of this phase and thus any differences in heat capacity would be expected to have a minimal impact.

s

and the entire primary side of the RCS have

primary and secondary metal and fluid inventory

Post-Reflood

The most significant difference between WCAP-10325 and WCAP-17721 is the modeling during the post-reflood phase. During this phase, the safety injection enters the cold leg, flows through the downcomer, through the core, and into the steam generators (primarily, the broken loop steam generator). While there is decay heat from the core, the most significant sources of energy during this phase are the energy in the secondary side liquid, secondary side metal, and primary side metal. Thus, due to these primary energy sources as well as the length of this phase (3400 seconds), any differences due to heat capacity would be expected to have its maximum impact.

14.7 psia

and intact loop(s)

For WCAP-10325, this phase ends when the broken loop steam generator's secondary side has depressurized to ~~the containment design pressure~~ (i.e., water and metal temperatures ~~in the broken loop steam generator~~ are at 212 °F). Once this occurs, all the heat transfer from the ~~steam generator~~ will have ceased. Because there is only a fixed amount of energy in the ~~broken loop~~ steam generator, this phase is dictated by the rate at which that energy is transferred to the primary coolant and into containment. A higher heat transfer rate would result in more energy transferred to containment and higher containment pressures and temperatures. Likewise, a slower heat transfer would result in less energy transferred to containment and lower containment pressures and temperatures.

s

saturation at 14.7 psia

and intact loop(s)

WCAP-~~10325~~ generates an artificially high heat transfer rate to ensure that the broken loop steam generator depressurizes to ~~the containment design pressure~~ by 3600 seconds following the break. This ensures that the post-reflood phase ends at exactly 3600 seconds, as the ~~broken loop steam generator~~ no longer ~~has~~ any remaining energy which could be transferred to containment. This artificially high heat transfer rate results in a very conservative M&E release for this phase.

steam generators

have

For WCAP-17721, this phase ends at the same time as WCAP-10325 (i.e., 3600 seconds), but with a significant amount of energy remaining in the broken loop steam generator. This is because WCAP-17721 does not use an artificially high heat transfer rate, []

[]. The difference between the energy transferred to containment by these two methodologies can be seen in Figures 5-2 and 5-3 of the TR. These figures demonstrate that by artificially increasing the heat transfer from the secondary side to the primary side coolant, WCAP-10325 removes the steam generator's energy at a rate much faster than would be expected as given by WCAP-17721 (WCT in the figures). The impact of this faster energy transfer rate results in conservatively higher predictions of containment pressures and temperatures, as seen in Figures 4.2-1 and 4.2-2 of the TR.

entire primary and secondary side of the RCS

Long-Term Steaming

saturated conditions at 14.7 psia

For both WCAP-17721 and WCAP-10325, the long-term steaming phase begins at 3600 seconds, however the plant conditions are different between the two methodologies. For WCAP-10325, the ~~broken loop steam generator~~ has depressurized to ~~the containment design pressure~~ and is therefore no longer acting as a heat source. Thus, the only remaining energy is the decay heat from the core which is boiled-off during this phase and hence any differences in heat capacities would have a minimal impact. []

]]. This phase ends when the primary and secondary side reach thermal equilibrium with their surroundings.

3.3 Evaluation of WCAP-10325 Mass & Energy Release

The WCAP-10325 analysis discussed in this SE includes the changes which were required due to the Westinghouse's NSAL-06-6, NSAL-11-5, and NSAL-14-2. As this review assumed that those issues were addressed, this approval is limited to use of WCAP-10325 with any compensating actions described in those NSALs.

WCAP-10325 was not updated to address the material property issue, which was captured in InfoGram (IG)-14-1. IG-14-1 discussed the discrepancy between the volumetric heat capacities used in WCAP-10325 and those of the most recent ASME Code. WCAP-10325 was found to use a lower volumetric heat capacity than is currently recommended by ASME. The consequences of using a lower heat capacity is that the analysis performed with WCAP-10325 will have a decreased initial stored energy in the RCS metal and secondary side steam generator metal than would be expected. Because this stored energy in the analysis is lower, the initial condition is non-conservative. Normally, this would require new analysis to be performed with the correct initial condition. However, in its submittal, PWROG argues that such re-analysis is not needed as WCAP-10325 remains a conservative method. This conservatism is due primarily to the artificial increase of heat transfer from the secondary side to the primary coolant during the post-reflood phase.

When WCAP-10325 was initially developed, there was no test data to determine the rate of heat transfer from the hot steam generator to the cooler two-phase RCS mixture. Further, even if some realistic heat transfer rate could be justified, computing power was not adequate to directly model the heat transfer from structures on the primary or secondary sides. Therefore, it was decided that this heat transfer rate should not be realistically modeled, but treated conservatively, which in this instance meant using a much larger heat transfer rate than would be realistically expected.

The rate of heat transfer was not chosen based on any physical mechanisms, but on a design requirement. The methodology described in WCAP-10325 is applied to two different containment designs, large dry containments and sub-atmospheric containments. At the time, one of the design requirements for sub-atmospheric containments was that they should depressurize to atmospheric pressure within one hour of the event initiation. Therefore, Westinghouse chose a heat transfer rate such that all the RCS metal and steam generator energy remaining after blowdown, refill, and reflood (approximately 200 seconds into the event) would be [[ ]]. This choice would both satisfy the design requirement for sub-atmospheric containments and result in an extremely conservative (i.e., higher than realistically expected) heat transfer rate for large dry containments.

Following the development of WCAP-10325, not only did computing power increase, but experimental data was taken to measure the rate of heat transfer from the hot steam generator to the cooler two-phase RCS mixture. Thus, when WCAP-17721 was developed, instead of using an artificial high heat transfer rate, Westinghouse chose to [[

]].

3.4 Evaluation of Containment Response Using WCAP-10325/COCO Code

The containment response analysis consists of determining the pressure and temperature transients due to the LOCA M&E release inside the containment. The purpose of this analysis is to: (a) determine the peak containment pressure 'Pa' during a design basis LOCA as a Technical Specification (TS) requirement for containment integrated leak rate test (ILRT) (Type A test), (b) confirm that the peak containment pressure and its wall temperature do not exceed their design limits, (c) confirm that the peak containment pressure and peak containment vapor temperature are enveloped by the equipment environment qualification profile, and (d) determine the sump water temperature for adequate net positive suction head for pumps that draw water for the sump during the LOCA recirculation phase.

The ILRT pressure 'Pa' is defined in 10 CFR Part 50, Appendix J, Section II.I as follows:

> "Pa (p.s.i.g.) means the calculated peak containment internal pressure related to the design basis accident and specified in the technical specification or associated bases."

Referring to Final Safety Analysis Reports (FSARs) for the Westinghouse PWRs with large dry and sub-atmospheric containments, it is noted that in some plants the limiting peak pressure occurs in the post-blowdown, while in others during the blowdown phase. During the blowdown phase, the peak pressure is due to a double-ended hot leg (DEHL) break. During the post-blowdown phases, the peak pressure is due to a double-ended pump suction (DEPS) break or a double-ended cold leg (DECL) break. The DECL and DEPS breaks are similar because in both cases, the core exit flow travels through the intact loop steam generators. However, the DEPS break is often found to be more limiting than the DECL break in terms of containment pressure as it produces a higher core reflood rate.

It is necessary to confirm that the WCAP-10325 M&E input (with NSALs -06-6, -11-5, and -14-2 accounted for and without considering the IG-14-1) remains conservative in determining the parameters (a), (b), (c), and (d) mentioned above for both blowdown and post-blowdown break analyses. This means ensuring WCAP-10325's ability to simulate both DEPS and DEHL breaks.

Double-Ended Pump Suction Break Loss-of-Coolant Accident Containment Response

The TR provides evaluation of the containment response for the DEPS break LOCA in which the post-blowdown pressure peak is higher than the blowdown pressure peak. To confirm if the WCAP-10325 methodology is conservative, the results of containment response are compared using the NRC accepted Westinghouse code COCO [Containment Pressure Analysis Code] (Ref. 11) while using the M&E release input from the WCAP-10325 and WCAP-17721 methodologies. The comparison is based on a DEPS break for a four-loop standard PWR with a large dry containment assuming a loss of offsite power coincident with a limiting single failure of the loss of one train of safety injection pumps and one train of containment heat removal system (i.e., one containment spray pump and one train of containment fan coolers). The selection of minimum safety injection flow is appropriate and is consistent with the discussion in Section 3.3 of WCAP-10325 because it conservatively maximizes the post-blowdown pressure.

That DEPS containment response analysis shows minor differences between the containment pressure and temperature response during the blowdown, refill, and reflood phases. This is expected as the methodologies use similar models during those phases and the phases are

relatively short. The main difference between the methodologies occurs in the post-reflood phase, as WCAP-17721 mechanistically models the heat transfer rate from the broken loop steam generator. WCAP-10325 is highly conservative in the post-blowdown phase because it is based [[ ]] from the LOCA initiation.

The time traces of the stored metal energy, secondary fluid energy, and stored energy release from these two methodologies can be seen in Figures RAI-2-A-5, RAI-2-A-6, and RAI-2-A-7 respectively in Reference 13. The time traces of break mass release rate and break energy release rate are shown in Figures RAI-2-B-1 and RAI-2-B-2 respectively in Reference 13. WCAP-10325 increased energy transfer resulted in much higher containment pressures and temperatures during the post-reflood period than predicted by the WCAP-17721, as demonstrated in Figures RAI-2-A-3 and RAI-2-A-4 in Reference 13, and Figure 4.2-3 in Reference 2. Thus, compared to the WCAP-17721 mechanistic analysis, WCAP-10325 conservatively over-predicts the energy released to containment during the post-reflood phase of the DEPS break LOCA. Therefore, the NRC staff finds that any difference due to the reduced heat capacity used in WCAP-10325 is more than accounted for in that methodology's conservative treatment of the heat transfer from the broken loop steam generator. Further, while the NRC staff would expect this analysis to vary for a specific plant design, the large conservatism of WCAP-10325 could be reasonably expected to be maintained across plant designs for the DEPS break LOCA due to it being inherent to the methodology in general, and not dependent on any specific plant parameter.

<u>Double-Ended Hot Leg Break Loss-of-Coolant Accident Containment Response</u>

As stated above, the staff review of FSARs showed that in several PWRs, the DEHL break LOCA results in blowdown pressure peak higher than the post-blowdown peak. Therefore, the NRC staff determined that PWROG evaluation of the DEPS break LOCA does not bound the entire Westinghouse PWR fleet with large dry and sub-atmospheric containments. To address this issue, the NRC staff requested the PWROG to analytically justify that the WCAP-10325 methodology (with NSALs accounted for, and without accounting IG-14-1) results in an acceptable or conservative simulation of containment response for PWRs which have a blowdown pressure peak higher than the post-blowdown peak. This was requested through an RAI.

In its response to the NRC RAI (Refs. 12 and 13), the PWROG confirmed that the DEHL break LOCA results in highest peak pressure during the blowdown phase because the break flow, from the reactor and pressurizer to the containment atmosphere, has a least resistance path such that the M&E release rates are maximized for the first 15 to 30 seconds from the LOCA. Instead of using the same approach as for the DEPS break (i.e., comparing the containment response), the PWROG performed a sensitivity analysis and determined the effect of the ASME material properties on the containment pressure response for the DEHL break LOCA by using the WCAP-10325/COCO methodology.

For this analysis, the PWROG used the results of a plant-specific sensitivity analysis. The analysis was performed by Westinghouse in response to NRC staff RAI SCVB-RAI-15(a) on a Braidwood Unit 1 license amendment request (Reference 15). The results of this analysis showed that using the ASME material properties in WCAP-10325, resulted in the peak containment pressure increasing by +0.1 psi compared to the resulting peak containment pressure using the original material properties in the WCAP-10325 methodology. The change

in the containment vapor temperature and sump water temperature was found to be insignificant. Table RAI-3/4-1 in Reference 13 shows the Braidwood Unit 1 results taken from Reference 15 sensitivity analysis. The PWROG stated that the representative 4-loop PWR analyzed for DEPS break LOCA in the TR is similar to Braidwood Unit 1. To cover the entire 4-, 3-, and 2-loop Westinghouse PWR fleet with large dry and sub-atmospheric containments that have higher LOCA blowdown peak pressure, the PWROG followed the following steps to calculate the effect of the ASME properties on the peak containment pressure:

(a) For the bounding analysis for the DEHL break LOCA, the following PWRs were selected:

- Two 4-loop plants; one with a large dry containment, and second with the highest licensed core power and the largest reactor vessel.

- Two 3-loop plants; one with the smallest volume for a large dry containment design, and second with an overall peak containment pressure that occurs during the initial blowdown phase.

- One 2-loop plant; the plants are all similar in power level, RCS fluid and metal volume and they all have a similar dry containment design.

(b) From the Braidwood Unit 1 data for total metal energy release due to the higher ASME material property, the fractional increase, which is used as an adjustment multiplier for the plants selected in step (a), is calculated as shown below. This adjustment multiplier represents the percentage increase in the metal energy release during blowdown due to the ASME material property change.

Adjustment Multiplier =

$$\frac{\text{Total Metal Energy Released (ASME Properties)} - \text{Total Metal Energy Released (Standard Properties)}}{\text{Total Metal Energy Released (Standard Properties)}}$$

= 0.0886

Braidwood Unit 1 is a 4-loop PWR with a large dry containment. The adjustment multiplier is calculated while accounting for its primary and secondary metal of the 4 steam generators and 4 reactor coolant loops. Therefore, the use of the adjustment multiplier would be conservative when applied to 3-loop and 2-loop break energy data.

(c) Applying the adjustment multiplier to the existing plant data for the total metal energy release with standard (current) metal properties, the increase in metal energy due to ASME properties is calculated for the PWRs selected in step (a).

Per Table RAI-3/4-2 in Reference 13, the metal energy increases due to the use of ASME properties for Braidwood Unit 1 and the PWRs selected in step (a). This increase is between 0.7 and 1.58 MBtu depending on the plant design.

(d) The plant-specific break energy increase 'x' due to ASME properties for the PWRs selected in step (a) is calculated from the following ratio:

$$\frac{\text{Increase in Metal Energy (Braidwood)}}{\text{Increase in Metal Energy (Plant Specific)}} = \frac{\text{Break Energy Increase (Braidwood)}}{\text{Break Energy Increase (Plant Specific) or 'x'}}$$

Not all of the increase in the metal energy will be released during blowdown. Therefore, Table RAI-3/4-2 in Reference 13, Attachment 1, provides the break energy increase 'x' due to the use of ASME properties for Braidwood and the PWRs selected in step (a). This increase of break energy during blowdown is between 0.3 and 0.7 MBtu.

(e) The plant-specific break energy increase 'x' in step (d) is added to the current energy release data of the PWRs selected in step (a). Using the increase energy, containment response analysis is performed using the COCO methodology. Table RAI-3/4-3 in Reference 13, shows the peak pressures with and without adding the break energy increase due to the use of ASME properties for Braidwood and the PWRs selected in step (a). This increase is between 0.05 and 0.10 psi.

The acceptability of the sensitivity analysis approach is confirmed by benchmarking with the Braidwood Unit 1 energy release data. The peak pressure increase shown in Tables RAI-3/4-1 is 0.1 psi and based on the sensitivity analysis approach in 0.09 psi as shown in Table RAI-3/4-3 of Reference 13. Since the Braidwood Unit 1 difference in the peak pressure increase is insignificant, the NRC staff determined the approach used for determining the peak pressure increase for the selected 4-, 3-, and 2-loop plants is acceptable. For the selected PWR in step (a), the effect of using the ASME properties in WCAP-10325 methodology is an increase in the peak containment pressure of ≤ 0.1 psi as seen in Table RAI-3/4-3 of Reference 13.

The NRC staff finds the sensitivity analysis approach, for a DEHL break LOCA acceptable because it is appropriately benchmarked with the Braidwood Unit 1 LOCA containment response data.

The NRC staff finds the selection of the two 4-loop PWRs in step (a) above for sensitivity analysis appropriate because due to their higher core power and larger reactor vessel, they would have greater values of the primary and secondary metal energy released during blowdown phase than the remaining 4-loop plants. The 4-loop plants also have greater metal energy releases than the selected 3-loop and 2-loop plants as shown in Table RAI-3/4-2 of Reference 13. The two 3-loop plants are also appropriately selected because during the LOCA blowdown phase the smallest volume containment would result in a higher-pressure response. The selection of one 2-loop plant is also appropriate because as stated by the PWROG, the remaining 2-loop plants have similar thermal power level and containment design.

The NRC staff finds the WCAP-10325 methodology, while including the NSALs 06-6, 11-5, and 14-2 and without addressing IG-14-1 acceptable for all Westinghouse PWRs with large dry and sub-atmospheric containments, because the effect of the ASME material properties has an insignificantly small increase by ≤ 0.1 psi in the blowdown peak containment pressure.

and does not utilize residual heat exchangers and fan coolers for accident mitigation

plants includes a spray system,

may also include safety grade

s

for pressure and temperature reduction during a LOCA. For sub-atmospheric containments,

heat removal equipment at

Sub-Atmospheric Containments

The NRC staff finds that the WCAP-10325 methodology, while including the NSALs 06-6, 11-5, and 14-2 and without addressing IG-14-1, acceptable for all Westinghouse PWRs with sub-atmospheric containments because they are similar in design and function to the ~~large dry~~ containments. The main difference is the sub-atmospheric containment has a ~~quench~~ spray system ~~for pressure and temperature reduction during a LOCA~~, whereas the large dry containments ~~has~~ residual heat removal heat exchanger and fan coolers. ~~The~~ quench spray system sprays colder water from the refueling water storage tank to reduce the post-blowdown containment pressure at a faster rate such that the WCAP-10325 M&E release do not produce a second peak. Regarding the blowdown peak containment pressure, the effect of the ASME material properties on the WCAP-10325 M&E release and containment response is same as for the large dry containments.

t

s

the

4.0 CONCLUSION

As stated in 10 CFR Part 50, Appendix K, and reflected in SRP 6.2.1.3 and ANS 56.4, heat sources from piping and other materials (e.g., secondary side fluid and metal) need to be considered for the containment M&E analysis. The NRC staff finds that despite having reduced volumetric heat capacity and hence a lower initial stored energy, the WCAP-10325 methodology predicts conservative M&E release for the DEPS break LOCA. For pressure peaks which occur in the blowdown phase (i.e., the DEHL break), the WCAP-10325 M&E analysis, while using the reduced material properties rather than their ASME values, results in an insignificantly small under-prediction by ≤ 0.1 psi in the peak containment pressure and is acceptable to the NRC staff. For pressure peaks which occur in the post-reflood phase (i.e., the DEPS and DECL breaks), Figures 5-2, 5-3 and 5-4 in Reference 2 demonstrate that WCAP-10325 predicts a much higher energy release than would be realistically expected during the LOCA post-reflood phase. This results in a much higher total energy release to containment over a much shorter time, as demonstrated in Reference 2, Figure 3.2-3. Consequently, the containment pressure, temperature, and sump temperature are conservatively over predicted in the post-blowdown phase as demonstrated in Figures 4.2-1, 4.2-2, and 4.2-3 in Reference 2.

The NRC staff finds there is reasonable assurance that the use of WCAP-10325, with NSAL-06-6, -11-5, and -14-2 addressed, will result in a conservative analysis and satisfy the criteria provided in SRP 6.2.1.3 and the requirements of 10 CFR Part 50, Appendix K. Therefore, the NRC staff concludes that the continued use of WCAP-10325 is acceptable for performing M&E analysis for plants with large dry and sub-atmospheric containments. This approval is subject to the conditions and limitations listed below.

5.0 CONDITIONS AND LIMITATIONS

1. This approval is limited to WCAP-10325 which is used in such a manner that address NSAL-06-6, NSAL-11-5, and NSAL-14-2.

2. This approval is for large dry containments and sub-atmospheric containments. This approval does not apply to ice condenser containments.

6.0 REFERENCES

1. Schrader, K, PWR Owners Group letter to U.S. Nuclear Regulatory Commission, "Transmittal of PWROG-17034-P, Revision 0, "Evaluation of the WCAP-10325-P-A Westinghouse LOCA Mass & Energy Release Methodology," OG-18-75, March 29, 2018, ADAMS Accession No. ML18114A184.

2. PWROG, "Evaluation of the WCAP-10325-P-A Westinghouse LOCA Mass & Energy Release Methodology," PWROG-17034-P, Revision 0, March 2018, ADAMS Accession No. ML18114A186 (*Proprietary Version, Non-Publicly Available*) and PWROG-17034-NP, Revision 0, March 2018, ADAMS Accession No. ML18114A185 (Non-*Proprietary Version, Publicly Available*).

3. Westinghouse, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version," WCAP-10325-P-A, Revision 0, Pittsburgh, PA, May 1983, ADAMS Accession Nos. ML080640615 (*Proprietary Version, Non-Publicly Available*).

4. Westinghouse, "Westinghouse Containment Analysis Methodology – PWR LOCA Mass and Energy Release Calculation Methodology," WCAP-17721-P-A, Revision 0, Pittsburgh, PA, September 2015, ML15272A206 and ML15272A207 (*Proprietary Version, Non-Publicly Available*), ML15272A203 and ML15272A204 (Non-*Proprietary Version, Publicly Available*). , LTR-NRC-15-81, September 24, 2015, ADAMS Accession No. ML15272A202

5. U.S. NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," June 1987 (Certain updated sections are available from the NRC).

6. U.S. NRC, Section 6.2.1, "Containment Functional Design (LOCAs)," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Revision 3, March 2007, ADAMS Accession No. ML070220505.

7. U.S. NRC, Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant-Accidents (LOCAs)" of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Revision 3, March 2007, ADAMS Accession No. ~~ML053550265~~. ML053560191

8. American Nuclear Society, ANS 56.4, "Pressure and Temperature Transient Analysis for Light Water Reactor Containments," December 1983.

9. Letter from Morey, D., USNRC to Nowinowski, W.A., PWROG, "Acceptance for Review of the Pressurized Water Reactor Owners Group Topical Report PWROG-17034-P, Revision 0, 'Evaluation of the WCAP-10325-P-A Westinghouse LOCA Mass & Energy Release Methodology,'" May 9, 2018, ADAMS Accession No. ML18116A029.

10. Schrader, K, PWR Owners Group letter to U.S. Nuclear Regulatory Commission, "Transmittal of Supplemental Change Pages 2-2 and 2-3 for PWROG-17034-P/-NP, Revision 0, 'Evaluation of the WCAP-10325-P-A Westinghouse LOCA Mass & Energy Release Methodology,'" OG-18-157, July 9, 2018, ADAMS Accession No. ML18193A456 (*Proprietary Version, Non-Publicly Available*) and ML18193A455 (*Non-Proprietary Version, Publicly Available*)

11. WCAP-8327, "Containment Pressure Analysis Code (COCO)," July 1974, ADAMS Accession No. ML092460709, (*Proprietary Version, Non-Publicly Available*)

12. Schrader, K, PWR Owners Group letter to U.S. Nuclear Regulatory Commission, "Transmittal of the Response to Request for Additional Information, Associated with PWROG-17034-P, Revision 0, 'Evaluation of the WCAP-10325-P-A Westinghouse LOCA Mass & Energy Release Methodology,' PA-ASC-1420," July 17, 2019, ADAMS Accession No. ML19203A154.

OG-19-134

13. PWROG, "Attachment 1: Responses to RAIs Associated with Topical Report PWROG-17034-P (Proprietary)," ADAMS Accession No. ML19203A158, "Attachment 2: Responses to RAIs Associated with Topical Report PWROG-17034-P (Non-Proprietary)," ADAMS Accession No. ML19203A155.

14. Schrader, K, PWR Owners Group letter to U.S. Nuclear Regulatory Commission, "Transmittal of the PWROG Meeting Materials from the January 30, 2019, PWROG-NRC Meeting to Discuss RAI 3 and 4 Responses for PWROG-17034-P/NP (PA-ASC-1420)," ADAMS Accession No. ML19070A120, and Attachment 1 (presentation slides), ADAMS Accession No. ML19070A121 (*Proprietary Version, Non-Publicly Available*).

15. Gullot, D, Exelon Letter to U.S. U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding Request for a License Amendment to Braidwood Station, Units 1 and 2, Technical Specification 3.7.9, 'Ultimate Heat Sink,'" April 29, 2016, ADAMS Accession No. ML16123A014.

OG-19-35, March 6, 2019,

16. Berkow, H.N. Director (U.S. Nuclear Regulatory Commission) to Gresham, J.A. (Westinghouse), "Acceptance of Clarifications of Topical Report WCAP-10325-P-A, 'Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version' (TAC No. MC7980)," October 18, 2005, ADAMS Accession No. ML052660242

7.0 LIST OF ACRONYMS

10 CFR	Title 10 of the *Code of Federal Regulations*
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
BTU	British Thermal Unit
CFR	Code of Federal Regulations
COCO	Containment Pressure Analysis Code
DECL	double-ended cold leg
DEHL	double-ended hot leg
DEPS	double-ended pump suction
ECCS	emergency core cooling system
EEQ	equipment environment qualification
FSAR	Final Safety Analysis Report
GDC	general design criteria
ILRT	integrated leak rate test
LOCA	loss-of-coolant accident
M&E	mass and energy
MBTU	Million British Thermal Unit
NRC	U. S. Nuclear Regulatory Commission
NSAL	Nuclear Safety Advisory Letter
PWR	pressurized water reactor
PWROG	Pressurized Water Reactor Owners' Group
RAI	request for additional information
RCS	reactor coolant system

SE	safety evaluation
SRP	standard review plan
TS	Technical Specification
TR	topical report
WCT	WCOBRA/TRAC

Principal Contributors: J.S. Kaizer
A. Sallman

Date: September 26, ~~209~~ $\boxed{2019}$