Enclosure 2

M170071

ABWR DCD Markups

GEH Supplemental Information
on
Peak Cladding Temperature/10 CFR 50.46

Table 1.6-1 Referenced Reports (Continued)

Report No.	Title	Tier 2 Section No.
NEDC-30851P-A	W. P. Sullivan, “Technical Specification Improvement Analyses for BWR Reactor Protection System”, March 1988.	19D.6
NEDE-31096-A	“GE Licensing Topical Report ATWS Response to NRC ATWS Rule 10CFR 50.62”, February 1987.	19B.2
NEDE-31152-P	“GE Bundle Designs”, December 1988.	4.2
NEDO-31331	Gerry Burnette, “BWR Owner’s Group Emergency Procedure Guidelines”, March 1987.	18A
NEDC-31336	Julie Leong, “General Electric Instrument Setpoint Methodology”, October 1986.	7.3
NEDC-31393	“ABWR Containment Horizontal Vent Confirmatory Test, Part I”, March 1987.	3B
NEDO-31439	C. VonDamm, “The Nuclear Measurement Analysis & Control Wide	20.3
NEDC-32084P-A Rev. 2	"TASC-03A - A Computer Program for Transient Analysis of a Single Channel", July 2002.	6.3
NEDC-31858P	Limits and Elimination of Leakage Control System”, 1991	
NEDE-31906-P	A. Chung, “Laguna Verde Unit I Reactor Internals Vibration Measurement”, January 1991.	7.4
NEDO-31960	Glen Watford, “BWR Owners’ Group Long-Term Stability Solutions Licensing Methodology”, June 1991.	4.4
NEDC-32267P	“ABWR Project Application Engineering Organization and Procedures Manual”, December 1993.	17.1
NEDO-32686-A	“Utility Resolution Guide for ECCS Suction Strainer Blockage”, October 1998.	6C

NEDO-33173 Supplement 4-A, Revision 1	"Implementation of PRIME Models and Data in Downstream Methods", November 2012.	6.3

Because either the ADS initiating signal or the overpressure signal opens the safety-relief valve, no conflict exists.

The LPFL Subsystem is configured from the RHR pumps and some of the RHR valves and piping. When the reactor water level is low, the LPFL Subsystem (line up) has priority through the valve control logic over the other RHR Subsystems for containment cooling. Immediately following a LOCA, the RHR System is directed to the LPFL mode. When the RHR shutdown cooling mode is utilized, the transfer to the LPFL mode must be remote manually initiated.

6.3.3.6 Limits on ECCS Parameters

Limits on ECCS parameters are given in the sections and tables referenced in Subsections 6.3.3.1 and 6.3.3.7.1. Any number of components in any given system may be out of service, up to the entire system. The maximum allowable out-of-service time is a function of the level of

At the time of ABWR DCD renewal the ABWR LOCA analysis was reviewed using current US NRC approved models (References 6.3-3 and 6.3-4) and modern fuel to confirm conformance to the acceptance criteria for ECCS performance (described in 6.3.3.2). In all cases, the analysis results were less than what was predicted using the 10CFR50.46 delta PCT in Reference 6.3-2.

6.3.3.7 ECCS Analy

6.3.3.7.1 LOCA Ana

The methoc hange criterion in 10CFR50 ER/GESTR models app for the response analysis are listed in Table 6.3-1 and Figure 6.3-44.

6.3.3.7.2 Accident Description

The operation sequence of events for the limiting case is shown in Table 6.3-2.

6.3.3.7.3 Break Spectrum Calculations

A complete spectrum of postulated break sizes and locations were evaluated to demonstrate ECCS performance. For ease of reference, a summary of figures presented in Subsection 6.3.3.7 is shown in Table 6.3-5.

A summary of results of the break spectrum calculations is shown in tabular form in Table 6.3-4 and graphically in Figure 6.3-10. Conformance to the acceptance criteria (PCT= 1204°C, local oxidation = 17% and core-wide metal-water reaction = 1%) is demonstrated for the core loading in Figure 4.3-1. Results for the limiting break for each bundle design in a plant will be given for information to the USNRC by the COL applicant. See Subsection 6.3.6.3 for COL license information. Details of calculations for specific breaks are included in subsequent paragraphs.

6.3.3.7.4 Large Line Breaks Inside Containment

Since the ABWR design has no recirculation lines, the maximum steamline break (985 cm^2), maximum feedwater lines break (839 cm^2), and the maximum RHR shutdown suction line

All instrumentation required for automatic and manual initiation of the HPCF, RCIC, RHR and ADS Systems is discussed in Subsection 7.3.1, and is designed to meet the requirements of IEEE-279 and other applicable regulatory requirements. The HPCF, RCIC, RHR and ADS Systems can be manually initiated from the control room.

The RCIC, HPCF, and RHR Systems are automatically initiated on low reactor water level or high drywell pressure. The ADS is automatically actuated by sensed variables for reactor vessel low water level and drywell high pressure plus indication that at least one RHR or HPCF pump is operating. The HPCF, RCIC, and RHR Systems automatically return from system flow test modes to the emergency core cooling mode of operation following receipt of an automatic initiation signal. The RHR LPFL mode injection into the RPV begins when reactor pressure decreases to the RHR's pump discharge shutoff pressure.

HPCF injection begins as soon as the HPCF pump is up to speed and the injection valve is open, since the HPCF System is capable of injection water into the RPV over a pressure range from 8.12 to 0.69 MPaD or pressure difference between the vessel and drywell.

6.3.6 COL License Information

6.3.6.1 ECCS Performance Results

The exposure-dependent MAPLHGR, peak cladding temperature, and oxidation fraction for each fuel bundle design based on the limiting break size will be provided by the COL applicant to the USNRC for information (Subsection 6.3.3).

6.3.6.2 ECCS Testing

6.3-3 "Implementation of PRIME Models and Data in Downstream Methods", (NEDO-33173 Supplement 4-A, Revision 1), November 2012.
6.3-4 "TASC -03A - A computer program for Transient Analysis of a Single Channel", (NEDC -32084P -A, Revision 2), July 2002.

In accordance [illegible] refueling in which each ECCS subsystem is actuated through the emergency operating sequence (Subsection 6.3.4.1).

6.3.6.3 Limiting Break Results

Results for the limiting break for each bundle design will be provided to the USNRC by the COL applicant (Subsection 6.3.3.7.3).

6.3.7 Reference

6.3-1 "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K", (NEDE-20566-P-A), September 1986.

Add: 6.3-2 MFN-16-059, Supplement 1, "Peak Cladding Temperature 2016 Annual Reporting Under 10 CFR 50.46 for the GE Hitachi Nuclear Energy Advanced Boiling Water Reactor (ABWR) Design Certification and the ABWR Design Certification Renewal Application" (October 12, 2016), Enclosure 1, "Advanced Boiling Water Reactor 2016 Annual Report Under 10 CFR 50.46(a)(3)(iii)."

Table 6.3-4 Summary of Results of LOCA Analysis

Break Location	Break Size* (cm^2)	Systems Available	PCT (°C)	Maximum Local Oxidation
Based on Appendix K evaluation models:				
Steamline Inside Containment	985	1HPCF + RCIC +2 RHR/LPFL + 8 ADS	552	0.03%
Feedwater Line	839	1 HPCF + 2 RHR/LPFL + 8 ADS	542	0.03%
RHR Shutdown Cooling Suction Line	792	1 HPCF + RCIC + 2 RHR/LPFL+ 8 ADS	542	0.03%
RHR/LPFL Injection Line	205	1 HPCF + RCIC + 1RHR/LPFL + 8 ADS	542	0.03%
High Pressure Core Flooder	92	RCIC+2RHR/ LPFL + 8 ADS	542	0.03%
Bottom Head Drain Line	20.3	1HPCF + RCIC + 2 RHR/LPFL + 8 ADS	542 (1)	0.03% (2)
Steamline Outside Containment	3939	1 HPCF + RCIC + 2 RHR/LPFL + 8 ADS	621	0.03%
Based on bounding values:				
Steamline Outside Containment	3939	1 HPCF + RCIC + 2 RHR/LPFL + 8 ADS	619	0.03%

* The most severe ABWR design basis LOCA calculations (Subsection 6.3.3.7.8) involve use of bounding worst-case values for key plant parameters - including an arbitrary 20% increase in the break flow rate. Even with these bounding assumptions, the LOCA analyses demonstrate that the ABWR design still retains large margins between predicted peak fuel clad temperatures and the criteria of 10 CFR 50, Appendix K.

Tolerances associated with fabrication and installation may result as-built break areas that could be 5% greater than these values. Based on the above conservatisms in the LOCA analyses, these as-built variations would not invalidate the plant safety analysis presented in Chapter 6 and Chapter 15.

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Insert Note 1 here.

NOTE: The core-wide metal-water reaction for this analysis has been calculated using method 1 described in Reference 6.3-1. This results in a core-wide metal-water reaction of 0.03%.

NOTE 1: The estimated adjustment of the PCT based on reporting under 10 CFR 50.46 is a potential increase of 114 °C, or a total of 735 °C. This result demonstrates continued compliance to the 1204 °C (2200 °F) regulatory acceptance criteria limit with ample margin. See Reference 6.3-2.

Section 19.3.1.3.1

condensate storage tank. Sufficient makeup water is available to enable these pumps to maintain adequate core cooling for all events except large or medium liquid LOCAs.

A motor driven feedwater pump is combined in series with a condensate pump in order to provide a higher pressure system. Therefore, this option also depends on the availability of makeup water and electrical power. Sufficient makeup water is available to enable this series of pumps to maintain adequate core cooling for the small steam LOCA and transient events.

The fire protection system has two pumps which take suction from the firewater tanks and inject into the RPV through an RHR line. One pump is driven by an electric motor which requires AC power. The other is driven directly by a diesel engine. Once the reactor system has been depressurized, either pump can provide enough makeup water to restore and maintain the RPV water level following any transient (including IORV) event. The analysis to support this conclusion assumes a full ADS blowdown begins within 15 minutes after the vessel water level has reached the level 1 setpoint. The subsequent reactor system depressurization allows injection from the fire protection system about 7 minutes after the start of the blowdown. The ability of the fire protection system to mitigate the consequences of LOCA events is conservatively ignored. For more information about the fire protection system refer to Subsection 5.4.7.

All calculated PCTs in 19.3 were conservatively updated as part of the ABWR DCD renewal to account for the estimated adjustments based on 10CFR50.46. See Reference 19.3-8.

It is conservative to use the 1204°C (2200 °F) PCT licensing limit as an acceptance criteria for the success criteria since tests have been performed which show that the core will remain in a coolable geometry with temperatures as high as 1482°C (2700°F).

A review of Table 19.3-2 shows that, for success, the inventory threatening events require the flow equivalent of only 1 RHR/LPFL or 1 HPCF pump available for large break cases and only 1 HPCF or 1 RHR/LPFL + 3 ADS available for small break cases. The resulting PCTs for the large break cases and transients were between 596°C(1100°F) and 707°C(1300°F) ~~482°C (900°F) and 593°C (1100°F)~~. For the small break cases with the flow equivalent of only 1 HPCF available the resulting PCTs were less than 652°C ~~538°C~~ and with 1 RHR/LPFL + 3 ADS available the maximum PCT was 1096°C(2000°F) ~~982°C (1800°F)~~.

Subsection 6.3.3.7.8 identifies the input parameters that significantly impact the LOCA results. If the above analyses were reanalyzed with these conservative input parameters, it is estimated that only the resulting PCTs for the small break cases with 1 RHR/LPFL + 3 ADS available are above 1096°C ~~982°C~~ (2000°F) ~~(1800°F)~~. For these cases the PCT is estimated to be about 1374°C (2500°F) ~~1260°C (2300°F)~~. However, even for these conservative LOCA calculations all the PCTs are less

19.3.5 References

19.3-1 "Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessment," NUREG/CR-3862, Idaho National Engineering Laboratory, May 1985.

19.3-2 "Advanced Light Water Reactor Utility Requirements Document, Volume II, Chapter 1; Appendix A: PRA Key Assumptions and Groundrules", Draft, Electric Power Research Institute, August 1988, p. D4.

19.3-3 "GESSAR II, 238 Nuclear Island, BWR/6 Standard Plant Probabilistic Risk Assessment," 22A7007, General Electric Company, March 1982.

19.3-4 "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", WASH-1400, NUREG-75/014, United States Atomic Energy Commission, October 1975.

19.3-5 "Failure Rate Data Manual for GE BWR Components", NEDE-22056, Rev. 2, Class III, General Electric Company, January 17, 1986.

19.3-6 A.D. Swain and H.E. Guttmann, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications", NUREG/CR-1278, August 1983.

19.3-7 "Analysis of a High Pressure ATWS with Very Low Makeup Flow", DOE/ID-10211, Idaho National Engineering Laboratory, October 1988.

19.3-8 MFN-16-059, Supplement 1, "Peak Cladding Temperature 2016 Annual Reporting Under 10 CFR 50.46 for the GE Hitachi Nuclear Energy Advanced Boiling Water Reactor (ABWR) Design Certification and the ABWR Design Certification Renewal Application" (October 12, 2016), Enclosure 1, "Advanced Boiling Water Reactor 2016 Annual Report Under 10 CFR 50.46(a)(3)(iii)."