AREVA

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Revision 0

ACE/ATRIUM 11
Critical Power Correlation RAIs

Topical Report

October 2017

AREVA Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

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INTRODUCTION

The ACE/ATRIUM™* 11 Boiling Water Reactor (BWR) critical power correlation topical report was provided to the U.S. NRC for review in Reference 1. One additional request for additional information is addressed in this report.

* ATRIUM is a trademark of AREVA, Inc.

RAI-C:

Over the course of the review of Topical Report ANP-10335P, Revison 0, the NRC staff became aware of a leaking fuel rod at the Kernkraftwerk Leibstadt (KKL) nuclear power plant in Switzerland, a boiling water reactor/6 (BWR/6) plant operating on yearly cycles. The leaker was believed to have resulted from excessive cladding oxidation due to dryout. Subsequent inspections found widespread occurrences of dryout in locations throughout the core. In the next cycle, steps were taken to increase the minimum critical power ratio operating limit and prevent future instances of dryout. However, further inspections revealed even more dryout indications after the compensatory measures were taken. Additional inspections found that dryout had occurred in several cycles before the leaking fuel rod was identified.

The dryout indications were characterized by visible, wedge-shaped areas of increased oxidation on the fuel rods. The size, shape, and material properties of these areas of increased oxidation indicate that dryout occurred over an extended period of time while the reactor was operating at steady-state conditions, with cladding temperatures remaining below 800°C. Though the shape of the oxidized areas was consistent, the dimensions of the oxidized area and the oxide thickness varied. Dryout is believed to have occurred only when the reactor operated at greater than 95% of rated total core flow and was only observed in first-cycle bundles that had been operated with a fuel assembly power greater than 7.4 megawatts. Within these bundles, dryout indications were only found on certain symmetric rod positions and always in the upper part of the bundle.

Dryout of the type observed at the plant was not observed in critical power testing at similar bundle flow rates and powers. At no point during KKL's operation did the analytical methods developed by the fuel's vendor predict that margin to dryout would be sufficiently degraded for dryout to occur. Currently, the underlying mechanisms that caused the dryout at KKL are still unknown. Given this operating experience, how does AREVA provide reasonable assurance that adequate critical power margin will be maintained during normal operation, including the effects of anticipated operational occurrences?

AREVA Response C:

AREVA applies U.S. NRC reviewed and approved methods for the thermal limits. A summary description of the thermal limits methodology is provided in Reference 2. The thermal limits methodology, THERMEX, consists of a series of related analyses which establish an Operating Limit Minimum Critical Power Ratio (OLMCPR). The OLMCPR is determined from two calculated values, the Safety Limit Minimum Critical Power Ratio

(SLMCPR) and the limiting transient ΔCPR. The overall methodology is comprised of four major segments: 1) reactor core hydraulic methodology, 2) a critical power correlation, 3) plant transient simulation methodology, and 4) critical power safety limit methodology.

The first part, reactor core hydraulic methodology, provides pressure drop and flow distribution in the core and is described in References 2 and 3.

The second part is the critical power correlation that calculates the power or heat flux at the onset of dryout. These correlations are generally fuel design specific. Approved correlations include Reference 4 (ATRIUM-9B, ATRIUM-10 fuel), Reference 5 (ATRIUM-10 fuel), and Reference 6 (ATRIUM 10XM fuel). The critical power correlation for AREVA's newest fuel design, ATRIUM 11, (Reference 1) has been submitted for NRC review. AREVA also applies an NRC reviewed and approved methodology for co-resident fuel (Reference 7).

The third part, the plant transient simulation methodology is applied to calculate the ΔCPR. The methodology and computer codes for AREVA BWR plant transient analyses are the XCOBRA-T code (Reference 8) and the COTRANSA2 code (Reference 9). The COTRANSA2 code is used to calculate BWR system behavior for steady-state and transient conditions. This behavior is then used to provide input to the XCOBRA-T and XCOBRA codes, from which critical power ratios are determined for limiting transients. The regulatory review of AREVA's new transient methodology, AURORA-B (Reference 10) is in progress.

The fourth part provides the methodology for the safety limit calculation. The calculations supporting this part of the overall methodology are implemented in the SAFLIM2 code (Reference 11) and most recently the SAFLIM3D code (Reference 12).

The RAI specifically mentions critical power test programs. The critical power experimental test program of AREVA has been examined as part of the critical power correlation development program and has been audited by multiple NRC inspectors as part of the regulatory review process for the critical power correlations.

Reference 2 on page 3 concludes:

> *"The methodology described herein is based upon a series of assumptions which overestimate the potential of boiling transition and, as such, is judged to be conservative in the establishment of reactor operating thermal margins for boiling water reactors."*

The work of establishing thermal limits for nuclear reactors is in compliance with 10 CFR 50 App. B and the methods collectively provide assurance that the regulatory requirements on fuel clad integrity are completely satisfied.

Further evidence of the adequacy of the methods used by AREVA to establish appropriate thermal limits comes from AREVA's fuel inspection programs. [

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* The ATRIUM 10XM fuel assemblies operated in KKL are [

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Table 1. [ ]

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AREVA cannot compare AREVA fuel features, operation, or performance with that of the fuel which was found to be defective due to lack of knowledge of the failed fuel's design and the complete details of the investigation.

References

1. ANP-10335P, Revision 0, "ACE/ATRIUM 11 Critical Power Correlation," February 2015.

2. XN-NF-80-19(P)(A) Volume 3 Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description," January 1987.

3. XN-NF-79-59(P)(A), "Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies," November 1983.

4. EMF-2209(P)(A), Revision 3, "SPCB Critical Power Correlation," September 2009.

5. ANP-10249P-A, Revision 2, "ACE/ATRIUM-10 Critical Power Correlation," March 2014.

6. ANP-10298P-A, Revision 1, "ACE/ATRIUM 10XM Critical Power Correlation," March 2014.

7. EMF-2245(P)(A), Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," August 2000.

8. XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," February 1987.

9. ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," August 1990.

10. ANP-10300P, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios," December 2009.

11. ANF-524(P)(A) Revision 2 and Supplements 1 and 2, "ANF Critical Power Methodology for Boiling Water Reactors," November 1990.

12. ANP-10307P-A, Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," June 2011.