Entergy

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Dale E. James
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1CAN050502

May 20, 2005

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Supplement to Amendment Request
To Support Use of M5 Fuel Cladding and BHTP Departure from Nucleate Boiling Correlation
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

REFERENCES: 1. Letter to the NRC dated September 30, 2004, "License Amendment Request to Support Use of M5 Fuel Cladding and BHTP Departure from Nucleate Boiling Correlation, and 10 CFR 50.46 and 10 CFR Appendix K Exemption Request" (1CAN090402)

Dear Sir or Madam:

By letter (Reference 1), Entergy Operations, Inc. (Entergy) proposed a change to the Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TSs) to: 1) permit the use of M5 advanced alloy for fuel rod cladding and fuel assembly structural components; and 2) modify TS 2.1.1.2 to permit the use of the BHTP correlation, which is needed to utilize the Framatome ANP (FANP) high thermal performance (HTP) spacer grid design. The original submittal also included requests for an exemption pursuant to 10 CFR 50.12 from 10 CFR 50.46 and 10 CFR 50, Appendix K.

On April 12, 2005, Entergy and members of the NRC staff held a call to discuss the peak cladding temperature that would be experienced during a large break and small break loss of coolant accident for a mixed core and uniform core. As a result of the call, one question was determined to need formal response. Entergy's response is contained in Attachment 1.

There are no technical changes proposed. The original no significant hazards consideration included in Reference 1 is not affected by any information contained in the supplemental letter. There are no new commitments contained in this letter.

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If you have any questions or require additional information, please contact Dana Millar at 601-368-5445.

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 20, 2005.

Sincerely,

DEJ/DM

Attachment:
1. Response to Request for Additional Information

cc: Dr. Bruce S. Mallett
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U. S. Nuclear Regulatory Commission
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Attachment 1

To

1CAN050502

Response to Request for Additional Information

Response to Request for Additional Information Related to License Amendment Request to Support Use of M5 Fuel Cladding and BHTP Departure from Nucleate Boiling Correlation

Question 1:

Please provide the following information in support of the use of M5 cladding at Arkansas Nuclear One, Unit 1 (ANO-1):

1. The calculated peak clad temperatures (PCTs) for both the M5 cladding (Mark-B-HTP fuel design) and the co-resident Zircaloy cladding.

2. A statement that the LOCA EM considers both the pre-LOCA and LOCA oxidation in demonstrating compliance with 10 CFR 50.46 requirements.

3. A statement that the non-M5 fuel cladding oxidation is bounded by a number which is less than or equal to the 10 CFR 50.46 acceptance criteria of 17%.

Response 1:

Analytical Methodology

Framatome ANP (FANP) performed loss of coolant accident (LOCA) linear heat rate analyses to support all co-resident fuel scheduled to be inserted into ANO-1 Cycle 20. This includes the fresh Mark-B-HTP fuel with M5 cladding and the Mark-B9 fuel with Zircaloy (Zr-4) cladding. The Mark-B-HTP analyses were performed using the NRC-approved BWNT LOCA Evaluation Model (EM) (BAW-10192P-A Rev. 0, Reference 1), the associated code or methods topical reports, and any changes implemented via 10 CFR 50.46 letters and reports since the EM was approved. Revision 4 of the RELAP5/MOD2-B&W code topical report (BAW-10164P-A Rev. 4, Reference 2) was approved after the completion of the Mark-B9 LOCA analyses. Therefore, the full-core Mark-B9 LOCA analyses were performed with Revision 3 (Reference 3), and the Mark-B-HTP and mixed-core LOCA analyses were performed with Revision 4. Documentation linkage between the EM and the NRC-approved Revision 4 of the RELAP5/MOD2-B&W code topical is made through Appendix U of BAW-10179P-A Rev. 5 (Reference 5).

LOCA Results

The LOCA analyses performed with the BWNT LOCA EM are fuel design specific and consider the entire lifetime of the fuel rod (burnup from 0 to 62 GWd/mtU) in determining the compliance to the first four criteria of 10 CFR 50.46. The fifth criterion of 50.46 relates to long-term core cooling, which is generally independent of fuel design. These LOCA analysis results for the first three criteria, which are defined by the short-term core cooling phase analysis of the limiting LOCAs, are summarized in Table 1 and Table 2. It should be noted that the Mark-B-HTP analyses, that were performed to show acceptance to 10 CFR 50.46, included the replacement once through steam generators (ROTSGs) and other plant input parameter changes. Therefore, the analyses were not intended to evaluate a single effect related to the introduction of M5 cladding. A description of the effect of M5 cladding may be found in the NRC-approved M5 cladding topical report (Reference 4).

The maximum local oxidation reported in Tables 1 and 2 is calculated based on the approved EM guidelines, which includes specification of a minimum pre-accident oxidation to maximize the predicted Peak Cladding Temperature (PCT). Therefore, the reported maximum local oxidation is the sum of the minimum pre-accident oxidation and the oxidation increase predicted during the LOCA transients (Large Break LOCA (LBLOCA) and Small Break LOCA (SBLOCA)) that provide the limiting PCTs.

Additionally, Appendix I of the M5 cladding topical report (BAW-10227P-A, Reference 4) commits FANP to consider realistic pre-accident oxidation to ensure that the 17% criteria would continue to be met. The realistic pre-accident oxidation is 5.5% for Mark-B-HTP fuel at 62 GWd/mtU and 13.5% for Mark-B9 fuel at 62 GWd/mtU. The maximum burnup reported corresponds to the maximum time in life considered in the LOCA analyses. When the realistic pre-accident oxidation is conservatively combined with the analyzed transient oxidation increase (maximum), the sum total also remains less than 17%. Therefore, this criterion is met for both the fresh Mark-B-HTP fuel with M5 cladding and the co-resident Mark-B9 fuel with Zr-4 cladding.

The fourth 10 CFR 50.46 criterion of coolable geometry is ensured when the combined effects of the fuel assembly disfiguration from the dynamic seismic plus LOCA loading and transient fuel rod swelling and rupture do not result in gross core flow blockage that prevents adequate core cooling. The analysis of the dynamic loads on the Mark-B-HTP spacer grids from a combined LOCA and seismic event has the acceptance criterion of no permanent grid deformation that alters the fluid coolant channels at the onset of the LOCA. In addition, the LOCA analyses predict that the assembly flow area reduction from the transient M5 cladding swell and rupture in the Mark-B-HTP assembly has considerable margin to the gross flow blockage criteria. Therefore, the calculated change in the Mark-B-HTP fuel assembly core geometry results in a fuel pin lattice that remains amenable to cooling.

The fifth acceptance criterion of 10 CFR 50.46 states that the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core. Successful initial operation of the emergency core cooling system (ECCS) mitigates the consequences of the LOCA during the short-term core cooling period by refilling and quenching the entire core, reducing the cladding temperatures to approximately the reactor coolant system (RCS) saturation temperature, and having the core decay heat energy removed without superheating the pumped ECCS injection that reaches the core. This is the point where the short-term code analyses are stopped and it is synonymous with the onset of the long-term core cooling phase.

Long-term core cooling is not fuel design specific. It is preserved by the operators while taking suction from the borated water storage tank (BWST) as they follow the emergency operating procedure (EOP) guidance in monitoring, aligning flow paths, and throttling the pumps or flow paths as necessary to maintain adequate to abundant ECCS flow reaching the core. The flow should match or exceed the flow rates used in the analyses of record for the plant. As the BWST empties, the operators successfully transfer the high pressure injection (HPI) pump source to the discharge of the low pressure injection (LPI) pumps (if required) and successfully transfer the suction of the LPI pump from the BWST to the containment emergency sump. The operators preserve a continuous core cooling mechanism by maintaining a flow alignment and managing the pump performance within acceptable operating ranges until the ECCS flow is no longer needed. The operators must also actively manage the core boron concentration to provide adequate shutdown reactivity and initiate active boron dilution paths that will prevent excessive boron concentrations that could lead to precipitation in the core.

Table 1: LBLOCA Analysis Results Demonstrating 10 CFR 50.46 Compliance [1]

	10 CFR 50.46 Limits	Whole Core		Mixed Core	
		Mark-B-HTP M5 Cladding	Mark-B9 [2] Zr-4 Cladding	Mark-B-HTP M5 Cladding	Mark-B9 [2] Zr-4 Cladding
PCT	2200 °F	2008.1 °F (@17.1 kW/ft)	2000 °F (@16.3 kW/ft)	1981.4 °F (@16.7 kW/ft)	2000 °F (@16.3 kW/ft)
Max Local Oxidation	17%	< 4%	< 3.1%	< 3%	< 3.1%
Whole Core H_2 Generation	1%	< 0.2%	< 0.3%	< 0.2%	< 0.3%
Coolable Geometry	Core coolable	Demonstrated [3]	Demonstrated [3]	Demonstrated [3]	Demonstrated [3]
Long Term Cooling	Core Temperature maintained and decay heat removed	Demonstrated [4]	Demonstrated [4]	Demonstrated [4]	Demonstrated [4]

Notes:

[1] The Mark-B-HTP analyses were performed with the ROTSGs installed based on Revision 04 of BAW-10164P-A (Reference 2), while the Mark-B9 analyses were originally performed with the currently installed once-through steam generators (OTSGs) based on Revision 03 of BAW-10164P-A (Reference 3).

[2] The Mark-B9 results with the currently installed OTSGs and 20 percent plugging are bounding for the 5 percent tube plugging with the ROTSG and other plant changes included in the Mark-B-HTP analyses. The Mark-B9 mixed-core results also neglect any beneficial flow diversion into the Mark-B9 assemblies from the higher resistance Mark-B-HTP assembly.

[3] A coolable geometry is demonstrated when LOCA plus seismic loads with any transient swelling and rupture do not result in gross core flow blockage.

[4] Demonstrated following initial core cooling by maintaining continuous pumped ECCS flows, managing core reactivity, and managing boron concentration to prevent precipitation.

Table 2: SBLOCA Analysis Results Demonstrating 10 CFR 50.46 Compliance [1]

	10 CFR 50.46 Limits	Mixed-Core ROTSG Results	
		Mark-B-HTP M5 Cladding	Mark-B9 [2] Zr-4 Cladding
PCT	2200 °F	1179.7 °F (17.5 kW/ft)	1179.7 °F (17.5 kW/ft)
Max Local Oxidation	17%	< 1%	< 1%
Whole Core H_2 Generation	1%	< 0.01%	< 0.01%
Coolable Geometry	Core coolable	Demonstrated [3]	Demonstrated [3]
Long Term Cooling	Core Temperature maintained and decay heat removed	Demonstrated [4]	Demonstrated [4]

Notes:

[1] The Mark-B-HTP analyses were performed with the ROTSGs installed based on Revision 04 of BAW-10164P-A (Reference 2), while the Mark-B9 analyses were originally performed with the currently installed OTSGs based on Revision 03 of BAW-10164P-A (Reference 3).

[2] The Mark-B9 PCTs were increased to account for the other plant changes included in the ROTSG analyses with the Mark-B-HTP fuel. The Mark-B9 mixed-core results neglect the small beneficial flow diversion into the Mark-B9 assemblies from the higher resistance Mark-B-HTP assembly.

[3] A coolable geometry is demonstrated when LOCA plus seismic loads with any transient swelling and rupture do not result in gross core flow blockage.

[4] Demonstrated following initial core cooling by maintaining continuous pumped ECCS flows, managing core reactivity, and managing boron concentration to prevent precipitation.

References

1. BAW-10192P-A Revision 0, "BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," June 1998.
2. BAW-10164P-A Revision 4, "An Advanced Computer Program for LWR LOCA and Non-LOCA Transient Analysis," November 2002.
3. BAW-10164P-A Revision 3, "An Advanced Computer Program for LWR LOCA and Non-LOCA Transient Analysis," July 1996.
4. BAW-10227P-A Revision 0, "Evaluation of Advanced Cladding and Structural Material in PWR Reactor Fuel," February 2000.
5. BAW-10179P-A Revision 5, "Safety Criteria and Methodology for Acceptable Core Reload Analysis," December 2004.