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1CAN090402

September 30, 2004

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

SUBJECT: 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
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests the following amendment for Arkansas Nuclear One, Unit 1 (ANO-1). The proposed change will modify ANO-1 Technical Specification (TS) 4.2.1, Fuel Assemblies, to permit the use of M5 advanced alloy for fuel rod cladding and fuel assembly structural components. In addition, Entergy proposes to modify TS 2.1.1.2, Safety Limits, to permit the use of the BHTP correlation which is needed to utilize the Framatome ANP (FRA-ANP) high thermal performance (HTP) spacer grid design. On December 18, 2002, the BHTP correlation was submitted for NRC review by FRA-ANP in BAW-10241P, *BHTP DNB Correlation Applied with LYNXT*. On the same date, FRA-ANP also requested approval of BAW-10179, *Safety Criteria and Methodology for Acceptable Cycle Reload Analysis*, Revision 5, to include use of the BHTP Correlation. FRA-ANP has received a draft Safety Evaluation (SE) from the NRC for BAW-10241P, which contains several conclusions that are addressed in Attachment 1.

This letter also requests an exemption pursuant to 10 CFR 50.12 from 10 CFR 50.46, *Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors* and 10 CFR 50 Appendix K, *ECCS Evaluation Models*. The M5 cladding is a proprietary zirconium-based alloy that is chemically different than Zircaloy or ZIRLO fuel cladding materials which are approved for use in these 10 CFR sections. Therefore, a plant specific exemption from these regulations is required to support the use of M5 cladding. Information supporting the exemption requests is contained in Attachment 4. Entergy has concluded that special circumstances defined by 10 CFR 50.12 exist to warrant the exemptions and that granting the exemption requests will not present undue risk to the public health and safety and is consistent with the common defense and security.

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The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that this change involves no significant hazards consideration. The bases for these determinations are included in Attachment 1.

The NRC has approved similar Technical Specification changes for other plants. In particular, fuel with M5 cladding is used at Oconee Units 1, 2, and 3, Three Mile Island, Unit 1, Davis Besse, and Crystal River, Unit 3, which are Babcock and Wilcox (B&W) plants similar to ANO-1. Crystal River, Unit 3 has also been approved to use the BHTP correlation in its current reload core design.

Entergy requests approval of the proposed amendment and exemption by September 30, 2005 in order to support fuel procurement and core design for the fall 2005 refueling outage. Once approved, the amendment shall be implemented within 60 days. Although this request is neither exigent nor emergency, your prompt review is requested.

The proposed change includes new commitments.

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 September 30, 2004.

Sincerely,

JSF/DM

Attachments:
1. Analysis of Proposed Technical Specification Changes
2. Proposed Technical Specification Changes (mark-up)
3. List of Regulatory Commitments
4. 10 CFR 50.46 and 10 CFR 50, Appendix K Exemption Request

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Attachment 1

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Analysis of Proposed Technical Specification Changes

1.0 DESCRIPTION

This letter is a request to amend Operating License DPR-51 for Arkansas Nuclear One, Unit 1 (ANO-1). The proposed change will revise the Operating License to permit the use of M5 advanced alloy by modifying Technical Specification (TS) 4.2.1, "Fuel Assemblies," and the use of the BHTP correlation by revising TS 2.1.1.2, Safety Limits. ANO-1 is planning to use an enhanced Framatome ANP (FRA-ANP) fuel design, which uses M5 material for fuel cladding and other assembly structural components, for the replacement fuel assemblies in future core reload designs starting with Cycle 20. The enhanced fuel design also utilizes a high thermal performance (HTP) spacer grid that requires the use of the BHTP correlation for departure from nucleate boiling (DNB) analyses.

In addition, an exemption from 10 CFR 50.46, *Acceptance criteria for emergency core cooling systems in light-water cooled power reactors* and 10 CFR Appendix K, *ECCS Evaluation Models* in accordance with 10 CFR 50.12 is requested (see Attachment 4). These exemption requests pertain to the proposed use of the M5 advanced zirconium alloy for ANO-1 fuel rod cladding and fuel assembly material.

2.0 PROPOSED CHANGE

The proposed change will add the allowance to use M5 advanced alloy fuel cladding to TS 4.2.1, thereby permitting the use of M5 cladding for replacement fuel assemblies in future core reloads. M5 fuel cladding is chemically different than Zircaloy, which is currently specified in TS 4.2.1.

The proposed change will also add a safety limit value for use of the BHTP correlation for DNB analyses in TS 2.1.1.2.

An exemption to 10 CFR 50.46 and 10 CFR Appendix K is also proposed in accordance with 10 CFR 50.12.

3.0 BACKGROUND

Currently, the ANO-1 fuel cladding is Zircaloy-4, which is allowed by TS 4.2.1. The fuel rod cladding is designed to maintain its integrity for the anticipated operating transients throughout core life. The effects of gas release, fuel dimensional changes, and corrosion- or irradiation-induced changes in the mechanical properties of cladding are considered in the design of fuel assemblies. The Zircaloy-4 cladding is designed to withstand strain resulting from combined effects of reactor pressure, fission gas pressure, fuel expansion, and thermal and irradiation growth. Materials testing and actual reactor in-service operation with Zircaloy cladding have demonstrated that Zircaloy-4 material has sufficient corrosion resistance and mechanical properties to maintain the integrity and serviceability required for design burnup.

In order to accommodate the high fuel rod burnups that are required for fuel management and core designs, FRA-ANP has developed the M5 advanced fuel rod cladding and fuel assembly structural material. M5 is an alloy comprised primarily of zirconium (~99 percent) and niobium (~1 percent). The elimination of tin in M5 has resulted in superior corrosion resistance and reduced irradiation-induced growth relative to both standard Zircaloy (1.7% tin) and low-tin Zircaloy (1.2% tin). The addition of niobium increases ductility, which is desirable to avoid brittle failures.

Results of test irradiations of M5 fuel rod cladding in commercial power reactors, in both the United States and Europe, have demonstrated that the maximum fuel rod cladding corrosion rate is 40 to 50% of that of Zircaloy-4. In addition, the hydrogen pickup rate is one-fourth of that experienced with Zircaloy-4 cladding. Therefore, the use of M5 cladding will provide a significantly improved margin to the 100 micron corrosion limit than the current margin associated with Zircaloy-4. The improvements in corrosion and hydrogen uptake are also applicable to fuel assembly structural components, such as guide tubes and spacer grids.

The same tests also illustrate that the M5 alloy exhibits significantly less irradiation-induced growth in fuel rods and fuel assembly guide tubes when compared to Ziracoly-4. The M5 cladding will provide additional margin to the fuel assembly and fuel rod growth limits for fuel assemblies with high burnups. Reduced fuel assembly growth will also help reduce irradiation-induced fuel rod bow and distortion, which can be detrimental to fuel handling activities. Fuel cladding creep collapse is furthermore greatly reduced for the M5 alloy relative to Zircaloy-4, which can benefit fuel rod internal pressure performance.

HTP correlations have been approved for use in DNB analyses for Westinghouse fuel since the early 1990s. NRC approval for the HTP correlation was extended to Combustion Engineering and other fuel designs in EMF-92-153(P)(A), *HTTP: Departure From Nucleate Boiling Correlation for High Thermal Performance Fuel*, Safety Evaluation dated December 28, 1993 (Reference 8). The HTP DNB correlation was used in conjunction with the XCORBRA-IIIC computer code to compare DNB predicted to measured heat fluxes.

FRA-ANP developed the methodology for the BHTP correction (use of the HTP correlation with the LYNXT computer code) and submitted the topical report BAW-10241P, *BHTP DNB Correlation Applied with LYNXT*, to the NRC for review and approval on December 18, 2002 (Reference 4). Framatome revised the topical report BAW-10179P, *Safety Criteria and Methodology for Acceptable Cycle Reload Analysis*, Revision 5 (Reference 7), to include the BHTP Correlation and submitted this to the NRC for review and approval on December 18, 2002. The LYNXT computer code was approved in BAW-10156-A, Revision 1, *LYNXT – Core Transient Thermal-Hydraulic Program* (Reference 9).

ANO-1 intends to use the BHTP correlation for Mark-B-HTP fuel, which utilizes the Siemens HTP spacer grid design combined with the FRA-ANP Mark-B series fuel design and analysis methods. BAW-10241P essentially combines two approved methods of DNB analysis (HTP DNB correlation and LYNXT) to accommodate the combination of fuel design features.

4.0 TECHNICAL ANALYSIS

Topical report BAW-10227P-A, Revision 1 *Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel*, approved by the NRC on June 18, 2003, provides the technical basis for the use of M5 fuel cladding material and structural material. The M5 cladding is an FRA-ANP proprietary material comprised primarily of zirconium (approximately 99%) and niobium (approximately 1%). As previously stated, M5 cladding, when compared to Zircaloy-4 cladding, provides for improvements in fuel cladding corrosion and hydrogen pickup, fuel assembly and fuel rod growth, and fuel rod cladding creep. The M5 alloy has been tested in both reactor and non-reactor environments to ascertain its mechanical and structural properties, as described in BAW-10227P-A.

In evaluating the properties of M5, FRA-ANP determined that the use of M5 would have either no significant impact or would produce a benefit for the following parameters and analyses:

- Fuel assembly handling and shipping loads
- Fuel rod internal pressure
- Fuel rod cladding transient strain
- Fuel centerline melting temperature
- Fuel rod cladding fatigue
- Fuel rod cladding creep collapse
- Fuel rod axial growth
- Fuel rod bow

Thus, FRA-ANP has determined that the M5 advanced alloy will perform acceptably at all normal operating conditions.

FRA-ANP has evaluated the performance of the M5 alloy for the Loss of Coolant Accident (LOCA) and non-LOCA accident scenarios in Reference 3. For the non-LOCA safety analyses performed using Zircaloy material properties, it was determined that these analyses apply equally to M5 cladding. Therefore, it is not necessary to recalculate any of the non-LOCA safety analyses solely because the cladding material is changed to M5.

Non-LOCA events in which the cladding material could affect DNB will be evaluated. For such accidents, a change from Zircaloy-4 to M5 fuel rod cladding should not produce any adverse consequences in DNB performance. This is to be expected, since both M5 and Zircaloy-4 have very similar heat transfer properties. In some cases, due to the reduced clad creep rate of M5, a DNB benefit can be produced since the reduced clad creep rate results in greater heat transfer surface area and, therefore, lower heat flux. For non-LOCA accident evaluations that do not involve DNB criteria, there is an effect of the M5 alloy if the transient involves the calculation of a detailed cladding temperature history with an excursion into the alpha-beta phase temperature range (approximately 700°C). For these transients, a small impact on temperature response is expected. This impact will be assessed in the Cycle 20 reload analysis in accordance with BAW-10179P-A, *Safety Criteria and Methodology for Acceptance Cycle Reload Analyses*, the analytical method reviewed and approved by the NRC for use at ANO-1 (refer to TS 5.6.5.b, Core Operating Limits Report). The results of these calculations are not expected to differ substantially from Zircaloy-4 based calculations and no limiting criteria are expected to be challenged.

Generic LOCA analyses for both B&W and Westinghouse plants and fuel have been performed using the specific material properties which accounted for the following:

- Lower alpha-beta phase transition temperature for M5 relative to Zircaloy-4
- Slower clad creep collapse for M5 relative to Zircaloy-4
- Slightly lower beginning of life yield strength for M5 relative to Zircaloy-4 (after 3 gigawatt days per metric ton of uranium (GWd/mtU), approximately 3 months of irradiation, the yield strength of M5 is equivalent to that of unirradiated Zircaloy-4).
- M5-specific clad swelling and rupture models determined through experimental measurements.

These generic LOCA analyses using M5-specific material properties have demonstrated that all five of the LOCA acceptance criteria mandated by 10 CFR 50.46 can be readily met in cores using M5 cladding. The analyses also demonstrated that there are no adverse LOCA-related issues that would prevent the acceptable use of M5 cladding. The cycle-specific reload report associated with Cycle 20 will include a plant specific-LOCA analysis prior to the use of the M5 alloy fuel assemblies at ANO-1. This LOCA analysis will be done in accordance with BAW-10179P-A and ANO-1 TS 5.6.5. The following paragraphs describe the methodologies that will be used to perform these analyses:

FRA-ANP will perform Mark-B-HTP LOCA analyses for ANO-1 using the NRC approved B&W Nuclear Technology (BWNT) LOCA Evaluation Model (Reference 1) using the blow down methods and models described in the NRC approved RELAP5/MOD2-B&W code (Reference 2). The RELAP5/MOD2-B&W code references the NRC-approved methods for applications of M5 cladding (Reference 3). The NRC-approved Evaluation Model (EM) blow down methodology states that the LOCA analyses will use the same critical heat flux (CHF) correlation that is used for the fuel pin DNB analyses. The BHTP CHF correlation (Reference 4) has been implemented into the RELAP/MOD2-B&W code for the analysis of the Mark-B-HTP fuel assemblies to support Cycle 20. The system reflooding phase of the large break LOCA (LBLOCA) analyses will be completed using the NRC-approved REFLOD3B code (Reference 5). The refill and reflood cladding temperature response will be completed with the NRC approved BEACH code (Reference 6).

LOCA analyses and evaluations for both mixed core and whole core configurations with Mark-B-HTP fuel will be performed to demonstrate compliance with 10 CFR 50.46 based on the NRC approved evaluation model. Beginning of life (BOL), middle of life (MOL), and end of life (EOL) cases with axial peaks simulated at the 2.506, 4.264, 6.021, 7.779, and 9.536 feet elevations will be included in a mixed-core and whole core Mark-B-HTP configuration.

The RELAP5/MOD2 blow down mixed core LBLOCA analyses that support Cycle 20 will conservatively place the Mark-B-HTP fuel with the higher form losses for the HTP grids in the hot channel and simulate the average channel with the Mark-B9 lower resistance fuel assemblies. The core bypass flow will be conservatively maximized in the mixed core analysis by simulating the core as though it were comprised entirely of higher resistance Mark-B-HTP fuel. The mixed core REFLOD3B analyses of the reflooding phase will also conservatively simulate the resistance of a full core of Mark-B-HTP fuel to increase the flow losses and minimize the core reflooding rate. The increase of flow diversion, flow losses and bypass flow conservatively reduces the fluid flow through the Mark-B-HTP assembly in the mixed core configuration.

Potentially limiting small break LOCA (SBLOCA) break sizes also will be analyzed with the NRC approved BWNT LOCA Evaluation Model (Reference 1) and NRC approved RELAP5/MOD2-B&W code (Reference 2) using the void-dependent core cross flow model and the BHTP CHF correlation that is used for the fuel pin DNB analyses. These cases will be analyzed in a mixed core simulation to demonstrate 10 CFR 50.46 compliance for the Mark-B-HTP fuel. The mixed core results will also be reported for the SBLOCA whole core results. The mixed core and whole core results are similar for SBLOCA transients because the quiescent core flow and lower core decay heat rate during the core uncovering phase of the transient do not result in substantial core flow diversion or changes in calculated peak centerline temperature (PCT).

The BWNT LOCA EM, associated code, and method topical reports have been approved for LOCA analysis for B&W 177 fuel assembly lowered-loop and raised-loop plant designs, as well as B&W 205 raised-loop plant types. Reference to the application of the BWNT LOCA EM for M5 cladding is made through Appendices N and U of BAW-10179P (Reference 7), with application of the DNB correlation (Reference 4) submitted in Appendix V of BAW-10179P. The ANO-1 plant-specific LOCA analyses and evaluations will be completed with a mixed core configuration input that conservatively represents Cycle 20 conditions. Analyses and evaluations will also be completed with a whole core configuration of Mark-B-HTP fuel for use in future cycles.

FRA-ANP has previously performed LOCA analyses to support all co-resident fuels. This includes the fresh Mark-B-HTP fuel with M5 cladding and the Mark-B9 fuel with Zircaloy-4 cladding. The analyses were performed using the NRC approved BWNT LOCA EM as described above. The LOCA reload analyses performed each cycle with the BWNT LOCA EM will consider the entire lifetime of the fuel rod in determining the limiting criteria with respect to 10 CFR 50.46.

When the realistic pre-accident oxidation is conservatively combined with the analyzed transient oxidation increase (maximum), the sum total will be validated to remain less than 17% to ensure this criterion is met for both the fresh Mark-B-HTP fuel with M5 cladding and the co-resident Mark-B9 fuel with Zircaloy-4 cladding.

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 predicts that there is no permanent grid deformation that alters the fluid coolant channels. 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.

FRA-ANP also determined that, for those accidents that result in radionuclide release (e.g., LOCA, control rod ejection accident, fuel handling accident), the use of M5 cladding and structural components will have no adverse impact on radiological doses. Again, this is due to the similar material properties and DNB performance of M5 and Zircaloy-4 during these accident scenarios.

The HTP correlation (Reference 8) has been used with the XCORBRA-IIIC thermal-hydraulic computer code (Reference 10) for the reload analyses of the HTP fuel designs. The incorporation of the HTP spacer grid into the Mark-B fuel design series reflects the integration of a Siemens developed spacer grid design into a Framatome developed fuel assembly design.

BAW-10241P provides the technical justification for using the HTP correlation with the LYNXT thermal-hydraulic code (Reference 4). The HTP data base has been evaluated using the LYNXT code and a 95/95 CHF design limit of 1.132 has been established. Although the original HTP CHF correlation form has been retained, the re-correlation has yielded changes to some of the coefficients that reflect the use of LYNXT. Since some of the coefficients have changed, the correlation has been given the distinct name of BHTP DNB Correlation.

The NRC has issued a draft Safety Evaluation (SE) for BAW-10241P which contains five conclusions related to the acceptability of the BHTP correlation for use in DNB analysis of the HTP fuels. The reload analysis will verify that the ANO-1 core is in compliance with these conclusions as listed below:

(1) Based on the data in Reference 3 [of the draft SER], the BHTP DNB correlation is applicable to fuels whose design characteristics fall within the correlation data base in Table 2 below.

(2) Based on the data in Reference 7 [of the draft SER], the application of the BHTP correlation for DNB analysis is restricted to the operating conditions given in Table 1, except as noted in Condition 5 below.

(3) The BHTP correlation limit is determined to be as stated in the subject TR [topical report] (Reference 2 [of the draft SER]).

(4) DNB penalty relative to DNB prediction for a full core of Mark-BHT fuel during transition core application shall be addressed in the plant-specific application.

(5) Actions for analyzing the operating conditions outside of the approved ranges of maximum pressure, but less than 2600 psia are acceptable in principle for this application (Reference 9 [of the draft SER]). Extrapolations below the minimum quality range using the process described in the TR are permitted with no lower limit. Any other extrapolation requires a plant-specific review.

Table 1
Range of Coolant Conditions for BHTP Correlation

Pressure (psia)	1775 to 2425
Local Mass Flux (Mlb/hr/ft^2)	0.897 to 3.549
Inlet Enthalpy (Btu/lb)	383.9 to 644.3
Local Quality	-0.130 to 0.344

Table 2
Range of Fuel Design Parameters for BHTP Correlation

Fuel Rod Diameter (in)	0.360 to 0.440
Fuel Rod Pitch (in)	0.496 to 0.580
Axial Spacer Span (in)	10.5 to 26.2
Hydraulic Diameter (in)	0.4571 to 0.5334
Heated Length (ft)	9.8 to 14.0

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met. Entergy has determined that the proposed change that allows the use of M5 fuel rod cladding material requires exemptions from 10 CFR 50.46, *Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors* and 10 CFR 50, Appendix K, *ECCS Evaluation Models*. Attachment 4 provides the basis and justification for relief from these regulations. The proposed change does not require relief from other regulatory requirements, other than the TS, and does not affect conformance with any General Design Criterion (GDC) differently than described in the Safety Analysis Report (SAR.)

5.2 No Significant Hazards Consideration

A change is proposed to the Design Features section of the Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TSs) to include the allowance to use M5 advanced alloy as a fuel rod cladding and fuel assembly structural material. Currently, Zircaloy-4 fuel cladding material is used as the fuel rod cladding material for ANO-1 fuel. An additional change is proposed to the Safety Limits section to allow the use of the high thermal power (BHTP) correlation for departure from nucleate boiling (DNB) calculations of reload cores containing the Mark-B-HTP fuel design.

Entergy Operations, Inc. (Entergy) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, *Issuance of amendment*, as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The NRC approved topical reports BAW-10227P-A, *Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel*, and BAW-10179P-A, *Safety Criteria and Methodology for Acceptable Cycle Reload Analyses*, provide the licensing basis for the Framatome ANP (FRA-ANP) advanced cladding and structural material, designated M5. The M5 material was shown in these documents to have equivalent or superior properties to the currently used Zircaloy-4 material. The cladding itself is not an accident initiator and does not affect accident probability. The M5 cladding has been shown to meet all 10 CFR 50.46 design criteria and, therefore, will not increase the consequences of an accident.

   The proposed safety limit value ensures that fuel integrity will be maintained during normal operations and anticipated operational occurrences (AOOs), and that the design requirements will continue to be met. The core operating limits will be developed in accordance with the new methodology. The proposed safety limit value does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. There is no impact on the source term or pathways assumed in accidents previously evaluated. No analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   Use of M5 clad fuel will not result in changes in the operation or configuration of the facility. Topical report BAW-10227P-A demonstrated that the material properties of the M5 alloy are similar or better than those of Zircaloy-4. Therefore, M5 fuel rod cladding and fuel assembly structural components will perform similarly to those fabricated from Zircaloy-4, thus precluding the possibility of the fuel becoming an accident initiator and causing a new or different type of accident.

   In addition, there will be no change in the level of controls or methodology used for processing radioactive effluents or handling solid radioactive waste. Since the material properties of M5 alloy are similar or better than those of Zircaloy-4, there will be no significant changes in the types of any effluents that may be released off-site. There will not be a significant increase in occupational or public radiation exposure.

   The proposed safety limit value does not change the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered. The BHTP correlation is not an accident / event initiator. No new initiating events or transients result from the use of the BHTP correlation or the related safety limit changes.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change will not involve a significant reduction in the margin of safety because it has been demonstrated that the material properties of the M5 alloy are not significantly different from those of Zircaloy-4. M5 alloy is expected to perform similarly or better than Zircaloy-4 for all normal operating and accident scenarios, including both loss of coolant accident (LOCA) and non-LOCA scenarios. For LOCA scenarios, where the slight difference in M5 material properties relative to Zircaloy-4 could have some impact on the overall accident scenario, plant-specific LOCA analyses will be performed prior to the use of fuel assemblies with fuel rods or fuel assembly components containing M5. These LOCA analyses, required by the ANO-1 TSs, will demonstrate that all applicable margins of safety will be maintained by the use of M5 alloy.

The proposed safety limit value has been established in accordance with the methodology for the BHTP correlation, to ensure that the applicable margin of safety is maintained (i.e., there is at least 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience DNB). The other reactor core safety limits will continue to be met by analyzing the reload for the mixed core using NRC approved methods, and incorporation of resultant operating limits into the Core Operating Limits Report (COLR).

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENCE

The use of the M5 advanced alloy material was previously approved for several other Babcock and Wilcox reactors. The amendments were issued as follows:

- Oconee Units 1, 2, and 3, Amendment No. 313 – June 21, 2000
- Davis Besse Nuclear Power Station, Unit 1, Amendment No. 239 – March 15, 2000
- Three Mile Island, Unit 1, Amendment No. 233 – May 10, 2001
- Crystal River Unit 3, Amendment No. 210 – October 1, 2003

The use of the BHTP DNB correlation was previously approved for one other Babcock and Wilcox reactor. The amendment was issued as follows:

- Crystal River, Unit 3, Amendment No. 211 – October 16, 2003

7.0 REFERENCES

1. BAW-10192P-A Revision 0, *BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam Generator Plants*, June 1988.

2. BAW-10164P-A Revision 4, *RELAP5 / MOD2 – B&W An Advanced Computer Program for LWR LOCA and Non-LOCA Transient Analysis*, November 2002.

3. BAW-10227P-A Revision 1, *Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel*, June 2003.

4. BAW-10241P, Revision 0, *BHTP DNB Correlation Applied with LYNXT*, December 2002

5. BAW-10171P-A Revision 3, *REFLOD3B – Model for Multinode Core Reflooding Analysis*, December 1995.

6. BAW-10166P-A Revision 5, *BEACH – Best Estimate Analysis Core Heat Transfer – A Computer Program for Reflood Heat Transfer During LOCA*, November 2003.

7. BAW-10179P Revision 5, *Safety Criteria and Methodology for Acceptable Core Reload Analysis*, December 2002.

8. EMF-92-153(P)(A) and EMF-92-153(P)(A), Supplement 1, *HTTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel*, March 1994

9. BAW-10156-A, Revision 1, *LYNXT – Core Transient Thermal-Hydraulic Program*, Babcock & Wilcox, Lynchburg, VA, August 1993.

10. XN-NF-75-21(P)(A), Revision 2, *XCORBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Operation*, Exxon Nuclear Company, January 1986.

Attachment 2

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Proposed Technical Specification Changes (mark-up)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the maximum local fuel pin centerline temperature shall be ≤ 5080 - (6.5 x 10^{-3} x (Burnup, MWD/MTU)°F) for TACO2 applications and ≤ 4642 - (5.8 x 10^{-3} x (Burnup, MWD/MTU)°F) for TACO 3 applications.

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio shall be maintained greater than the limits of 1.3 for the BAW-2 correlation, ~~and~~ 1.18 for the BWC correlation, and 1.132 for the BHTP correlation.

2.1.1.3 In MODES 1 and 2, Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the Variable Low RCS Pressure-Temperature Protective Limits as specified in the Core Operating Limits Report, so that the safety limits are met.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2750 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 or SL 2.1.1.2 is violated, be in MODE 3 within 1 hour.

2.2.2 In MODE 1 or 2, if SL 2.1.1.3 is violated, restore RCS pressure and temperature within limits AND be in MODE 3 within 1 hour.

2.2.3 In MODE 1 or 2, if SL 2.1.2 is violated, restore compliance within limits AND be in MODE 3 within 1 hour.

2.2.4 In MODES 3, 4, and 5, if SL 2.1.2 is violated, restore RCS pressure to ≤ 2750 psig within 5 minutes.

2.2.5 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

4.0 DESIGN FEATURES

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 177 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or M5 clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Assemblies

The reactor core shall contain 60 safety and regulating CONTROL ROD assemblies and 8 APSR assemblies. The CONTROL ROD assembly control material shall be a silver-indium-cadmium alloy and the APSR assembly control material shall be an Inconel alloy, as approved by the NRC.

Attachment 3

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List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
An evaluation of the performance of the M5 alloy for the Loss of Coolant Accident (LOCA) and non-LOCA accident scenarios that bound the accidents in the ANO-1 Safety Analysis Report will be performed in accordance with BAW-10179-A as part of the cycle-specific reload report associated with Cycle 20 prior to the use of the M5 alloy fuel assemblies at ANO-1.	x		
Beginning of life (BOL), middle of life (MOL), and end of life (EOL) cases with axial peaks simulated at the 2.506, 4.264, 6.021, 7.779, and 9.536 feet elevations will be included in a mixed-core and whole core Mark-B-HTP configuration.		x	
Potentially limiting small break LOCA (SBLOCA) break sizes will be analyzed to demonstrate 10 CFR 50.46 compliance for the Mark-B-HTP fuel.		x	
FRA-ANP will perform LOCA analyses to support all co-resident fuels. This includes the fresh Mark-B-HTP fuel with M5 cladding and the Mark-B9 fuel with Zircaloy-4 cladding. The LOCA analyses performed with the BWNT LOCA evaluation will consider the entire lifetime of the fuel rod in determining the limiting criteria with respect to 10 CFR 50.46.		x	
The reload analysis will verify that the ANO-1 core is in compliance with the conclusions of the Draft Safety Evaluation report for BAW-10241P.		x	

Attachment 4

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10 CFR 50.46 and 10 CFR 50, Appendix K Exemption Request

10 CFR 50.46 and 10 CFR 50, Appendix K Exemption Request

In accordance with 10 CFR 50.12, *Specific Exemptions*, Arkansas Nuclear One, Unit 1 (ANO-1) requests exemptions from the requirements specified in 10 CFR 50.46, *Acceptance Criteria for Emergency Core Cooling Systems for Light- Water Nuclear Power Reactors*, and 10 CFR 50 Appendix K, *ECCS Evaluation Models*, paragraph I.A.5, regarding the use of Zircaloy or ZIRLO as a fuel rod cladding material. These exemption requests pertain to the proposed use of the M5 advanced zirconium alloy for ANO-1 fuel rod cladding and fuel assembly material.

10 CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10 CFR 50 provided that: 1) the exemption is authorized by law, 2) the exemption will not result in an undue risk to public health and safety, 3) the exemption is consistent with the common defense and security, and 4) special circumstances, as defined in 10 CFR 50.12(a)(2) are present. The requested exemptions to allow the use of advanced zirconium alloys other than Zircaloy or ZIRLO for fuel cladding material for reloads at ANO-1 satisfy these requirements as described below.

1. The requested exemption is authorized by law.

   Transition to an alternate, but equivalent fuel product is not precluded by law. The fuel that will be irradiated at ANO-1 contains cladding material that does not conform to the cladding material designations explicitly defined in 10 CFR 50.46 and 10 CFR 50, Appendix K. However, the criteria of these sections will continue to be satisfied for the operation of the ANO-1 core containing M5 fuel rod cladding and fuel assembly structural material.

2. The requested exemption does not present an undue risk to the public health and safety.

   The M5 fuel rod cladding and fuel assembly structural material has been evaluated to confirm that operation of this fuel product does not increase the probability of occurrence or the consequences of an accident. The evaluation also concluded that no new or different type of accident will be created that could pose a risk to public health and safety. In addition, appropriate full-core and mixed-core safety analyses have been performed to demonstrate that this fuel type does not present an undue risk to the public health and safety. Entergy, in conjunction with Framatome-ANP (FRA-ANP), will utilize NRC approved methods for the reload design process for ANO-1 reload cores containing M5 fuel rod cladding and fuel assembly structural materials.

3. The requested exemption will not endanger the common defense and security.

   The M5 fuel rod cladding is similar in design to the current cladding material used at ANO-1. The special nuclear material in this fuel product will continue to be handled and controlled in accordance with approved procedures. It has been confirmed through evaluation, that M5 fuel rod cladding and fuel assembly structural material will not endanger the common defense and security.

4. Special circumstances are present which necessitate the request of an exemption to the regulations of 10 CFR 50.46 and 10 CFR 50 Appendix K.

   The special circumstance necessitating the request for an exemption to 10 CFR 50.46 and 10 CFR 50 Appendix K is that neither of these regulations allows the use of M5 fuel rod cladding material.

   The underlying purpose of 10 CFR 50.46 is to ensure that nuclear power facilities have adequately demonstrated the cooling performance of their Emergency Core Cooling System (ECCS). Framatome-ANP demonstrates in its topical report BAW-10227P-A, *Evaluation of Advance Cladding and Structural Material (M5) in PWR Reactor fuel*, approved by the NRC by letter dated February 4, 2000, that the effectiveness of the ECCS will not be affected by a change from Zircaloy fuel rod cladding to M5 fuel rod cladding. Normal reload safety analyses will confirm that the safety analyses performed to support the use of this fuel type will remain applicable for the ANO-1 cores. Consequently, the use of the M5 fuel cladding and fuel assembly structural material will not have a detrimental impact on the performance of the ANO-1 core under loss of coolant accident (LOCA) conditions.

   The underlying purpose of 10 CFR 50 Appendix K is to ensure that cladding oxidation and hydrogen generation are appropriately limited during a LOCA and conservatively accounted for in the ECCS evaluation model. Specifically, Appendix K requires that the Baker-Just equation be used in the ECCS evaluation model to determine the rate of energy release, cladding oxidation, and hydrogen generation. Framatome-ANP demonstrates in Appendix D of BAW-10227P-A, that the Baker-Just model is conservative in all post-LOCA scenarios with respect to the use of the M5 advanced alloy as a fuel rod cladding material, and that the amount of hydrogen generated in an M5 clad core during a LOCA will remain within the ANO-1 design basis.

Therefore, the intent of 10 CFR 50.46 and 10 CFR 50, Appendix K will continue to be satisfied for the planned operation with FRA-ANP M5 fuel rod cladding and fuel assembly structural material. Issuance of an exemption from the criteria of these regulations for the use of M5 fuel rod cladding and structural material in the ANO-1 core will not compromise the safe operation of the reactors.