ATTACHMENT (3)

SAFETY EVALUATION REPORT FOR USE OF ADVANCED ZIRCONIUM - BASED CLADDING MATERIALS IN CALVERT CLIFFS UNIT 1 BA''CH R LEAD FUEL ASSEMBLIES

MAY 1995

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SAFETY EVALUATION REPORT
FOR
USE OF ADVANCED ZIRCONIUM-BASED CLADDING MATERIALS IN CALVERT CLIFFS, UNIT 1 BATCH R LEAD FUEL ASSEMBLIES

MAY 1995

ABB COMBUSTION ENGINEERING NUCLEAR OPERATIONS

ABB

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Use of Advanced Zirconium-Based Cladding Materials in Calvert Cliffs Unit 1, Batch R Lead Fuel Assemblies

1.0 INTRODUCTION

ABB Combustion Engineering Nuclear Operations (ABB CENO) has developed an improved fuel design to meet the nuclear industry's continued need for greater fuel reliability, improved thermal margin, increased fuel discharge burnup, and more favorable fuel cycle economics. Four Lead Fuel Assemblies (LFAs) with improved fuel design features are planned for irradiation in Baltimore Gas and Electric's Calvert Cliffs Unit 1 reactor, beginning with Cycle 13. Currently these LFAs are scheduled for three cycles of irradiation. The final burnup achieved will exceed the current approved peak pin limit of 60 GWd/MTU (Ref. 1).

Several features of the LFAs unrelated to the exemption request that are being considered are: modification to the fuel rod design to allow increased uranium loading and higher burnups and incorporation of a new straight strip Zircaloy spacer grid design with improved rod support features and mixing vanes. These changes associated with the LFA will be implemented under 10 CFR 50.59. A primary feature of the LFA is the introduction and use of advanced zirconium-based alloys in the cladding with expected superior corrosion resistance. These advanced cladding materials do not strictly conform to the Zircaloy or ZIRLO designation. This report supports an exemption from the requirements 10 CFR 50.46, 10 CFR 50.44 and 10 CFR Part 50 Appendix K in order to allow the use of the advanced cladding materials.

1.1 Background

The corrosion performance requirements for nuclear fuel cladding are becoming more demanding with the continued trend in the nuclear industry towards increased fuel discharge burnups and longer exposure cycles. Under these more demanding operating conditions, the corrosion resistance of standard Zircaloy-4, the most common commercially used fuel cladding material, is not adequate to provide the necessary operational flexibility and performance margins. ABB CENO has developed process-optimized low-tin Zircaloy-4 (with tin content in the lower range of the ASTM specification for standard Zircaloy-4) for improved corrosion resistance to meet these needs. Improved corrosion performance of the optimized low-tin Zircaloy-4 has been demonstrated to 53 GWd/MTU in Palo Verde 1 reactor (Ref. 2) and irradiation is in progress to obtain data up to 60 GWd/MTU.

Although optimized low-tin Zircaloy-4 is believed to be capable of providing satisfactory performance for most of the current fuel management operational schemes, materials with a corrosion resistance superior to that of the optimized low-tin Zircaloy-4 may be needed for added operational flexibility at burnups beyond 60 GWd/MTU. In anticipation of this need, ABB CENO has developed new cladding materials with superior corrosion resistance. As part of this development program, several promising zirconium-based cladding alloys were included in two demonstration assemblies in Palo Verde Unit 3 Batch F (Ref. 3). These assemblies are scheduled for three cycles of irradiation, (via Cycles 4, 5 and 6) and include a total of [ ] fuel rods (located at the periphery of each assembly) clad with alloys with chemical composition outside conventional Zircaloy-4. The assemblies successfully completed [

] in comparison to optimized low-tin Zircaloy-4 (used as reference cladding).

The corrosion database on the advanced cladding alloys needs to be further expanded prior to the batchwide commercial application of non-Zircaloy-4 cladding materials. In addition, the optimum mix of enhanced corrosion resistance, acceptable mechanical properties, and ease of fabrication needs to be ascertained. To address these issues, fuel rods fabricated with five advanced cladding alloys are planned to be irradiated in lead fuel assemblies proposed for Calvert Cliffs Unit 1, Batch R reload fuel.

All the selected alloys are zirconium-tin alloys, like Zircaloy-4, with variations in other alloying elements. Some or all of these alloys will be used in the LFAs. Table 1 shows the maximum number of fuel rods using the advanced cladding variants in the proposed lead fuel assemblies. It is planned that a mixture of these alloys will be used in two of the LFAs. Fuel rods clad with optimized low-tin Zircaloy-4, which is the standard used for the rest of the Batch R fuel, are also included in these LFAs as controls. The other two LFAs will use all optimized low-tin Zircaloy 4 cladding. The cladding types to be used were chosen on the basis of improved corrosion properties demonstrated during extensive ex-reactor autoclave corrosion tests and some in-reactor tests. The mechanical properties and behavior of these alloys during postulated LOCA and non-LOCA transients and operational transients will be essentially the same as those of conventional Zircaloy-4.

The fuel rods containing the special cladding alloys are identical in design and dimension to the control fuel rods used in the LFAs. Specifically, all the rods will contain urania or urania-erbia fuel pellets of the same enrichment as the rest of the fuel rods in the LFAs. These assemblies will be placed in non-limiting locations in the core with predicted peak pin power not more than 0.95 of the predicted maximum peak pin power in the core. Since these assemblies will not be in the highest core power density locations, it is expected that the placement scheme and the similarity of the advanced alloys to Zircaloy-4 will assure that the behavior of the fuel rods clad with these alloys is bounded by the fuel performance and safety analyses performed for the Zircaloy-4 clad fuel rods.

Visual examinations and eddy current oxide thickness measurements are planned to confirm continued satisfactory behavior. Moreover, two Palo Verde Unit 3 Batch F LFAs are scheduled to complete their second cycle of irradiation (with maximum projected burnup of [ ] GWD/MTU) before the end of the first cycle of irradiation of these alloys in Calvert Cliffs Unit 1. The alloys used in the Palo Verde LFA's [ ] advanced cladding alloys proposed in Cycle 13 of Calvert Cliffs Unit 1. Irradiation performance in LWRs on the [ ] non-Zircaloy-4 cladding [ ] are already available (Ref. 4). These data support the expectation of superior material performance (i.e., reduced in-reactor growth and creep, and greater corrosion resistance). The reconstitutable upper end fitting feature of the fuel assemblies will provide access to the fuel rods for reconstitution in the unlikely event there is any indication of unsatisfactory performance following inter-cycle examinations and measurements.

1.2 Exemption From 10 CFR 50.46, 50.44 and Appendix K to 10 CFR Part 50 Requirements

The Code of Federal Regulations specifies standards and acceptance criteria strictly for fuel rods clad with Zircaloy or ZIRLO. Because the chemical compositions of [ ] of the six cladding alloys used in the fuel rods used in the LFAs are outside the Zircaloy and ZIRLO specifications, an exemption is needed to use the fuel rods clad with these alloys. Specifically, an exemption is required for variants of [ ]. This report describes the composition and properties of the selected alloys and provides a safety evaluation for the fuel rods clad with these alloys. For completeness, corrosion and in-reactor (PWR) performance of Zircaloy-2P are also addressed, where appropriate, although it is an acceptable cladding alloy under the existing regulations.

2.0 EVALUATION

2.1 Chemical, Mechanical, and Other Material Properties

2.1.1 Chemical Properties

The chemical compositions of the non-Zircaloy-4 cladding alloys (including Zircaloy-2P) are given in Table 2 along with alloy numbers designated for each cladding material. The chemical compositions of the cladding alloys previously approved by the Nuclear Regulatory Commission (NRC) for the demonstration assemblies included in Palo Verde Unit 3, Batch F are also given for comparison (Refs. 3 and 5).

All the cladding alloys used in the LFAs were selected on the basis of improved corrosion properties demonstrated during extensive ex-reactor autoclave corrosion tests. In the evaluation of ex-reactor autoclave test data, the corrosion resistance of the alloys has been compared against optimized low-tin Zircaloy-4. Optimized low-tin Zircaloy-4 has been used as the standard for comparison since this cladding has shown approximately 40 to 55% improvement in corrosion resistance over conventional Zircaloy-4 (containing ≈1.5-1.6% tin and a variation in processing history) when irradiated in Palo Verde Unit 1 reactor to 53 GWd/MTU (Ref. 2). Figure 1 shows the in-reactor data which demonstrate the beneficial effects of lowering the tin content and process optimization of Zircaloy-4, and thereby validate the use of optimized low-tin Zircaloy-4 as a high standard against which the other alloys are compared. Figure 2 compares the weight gain measured on several alloys relevant to the alloys of interest and compares with the data on optimized low-tin Zircaloy-4. Long-term autoclave corrosion behavior of conventional and optimized low-tin Zircaloy-4 are compared in Figure 3 to show that the 360°C water autoclave test used to evaluate the relative corrosion behavior of all the alloys, except for [ ] these alloys with respect to their in-reactor corrosion behavior.

In addition to the ex-reactor autoclave testing, all the cladding alloys except Alloy 4 have been tested in reactor and post-irradiation examination results are available on these alloys at least after the completion of [ ] of irradiation. [

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Available irradiation experience for the alloys of interest is summarized in Table 3. The table lists the maximum burnup for which in-reactor performance information is available for each of the proposed alloys. For each alloy, the host reactors, the number of fuel rods irradiated (or continuing irradiation) in each reactor and the scope of the post-irradiation examination performed, if any, are also shown. In general, the post-irradiation examination results on corrosion behavior confirm the trend observed in ex-reactor testing. The corrosion results are addressed below in conjunction with the discussions on chemical compositions that follow for each of the alloys. The alloys also showed either equivalent or superior in-reactor creep and growth behavior in comparison to Zircaloy-4, and these aspects are discussed later in the respective sections.

For Alloy 2 [ ], all elements except [ ] are within the range of conventional Zircaloy-4. The composition is practically identical to an alloy previously approved for PV-3F. The [

]. The objective of irradiating this alloy is to confirm the beneficial effect of [ ] under operating PWR conditions on additional fuel rods beyond those under irradiation in Palo Verde Unit 3, Batch F demonstration assemblies. Long-term autoclave corrosion data for 360°C water environment shown in

Figure 2 shows the beneficial effect of [ ] level on the corrosion resistance of Zircaloy-4. Oxide thickness measurements performed after the completion of the [ ] of irradiation in Palo Verde 3 to [ ] GWd/MTU showed [ ].

The inclusion of Alloy 3, Zircaloy-2P, as a cladding variant in the program is based on the observation of improved long-term corrosion performance of Zircaloy-2 as compared to Zircaloy-4 in autoclave tests (Figure 2). In addition, as shown in Table 3, ABB Atom has successfully irradiated Zircaloy-2P clad fuel rods on a large scale in Ringhals-3 and other European PWRs. Post-irradiation examination results on fuel rods irradiated for four cycles up to an average burnup of 44 GWd/MTU in Ringhals 3 were reported (Refs. 2 and 6). In comparison to Zircaloy-4, the Zircaloy-2 cladding showed 50 to 60% lower oxide thickness. Poolside oxide thickness measurements have recently been completed on these types of cladding with an [ ]. A preliminary examination of the data indicates that the [ ].

The hydrogen levels measured earlier on conventional high-tin Zircaloy-2 without a minimum requirement for silicon and conventional high-tin Zircaloy-4 and both irradiated to approximately 44 GWd/MTU were similar (Ref. 6). As discussed in Reference 2, a minimum silicon requirement was incorporated into the Zircaloy-2 specification (in addition to lowering tin content for improving corrosion resistance as in the case of Zircaloy-4) to lower hydrogen pickup while maintaining improved corrosion resistance. Zircaloy-2 with controlled tin and silicon contents and with processing optimized for PWR operation has been designated as Zircaloy-2P. Alloy 3 that is planned to be used in Calvert

Cliffs Unit 1 Batch R LFAs meets the minimum silicon requirement of Zircaloy-2P. Zircaloy-2P cladding [ ] is also included in Calvert Cliffs 2 Batch N LFAs that commence irradiation in Cycle 11.

A comparison of the corrosion performance of optimized low-tin Zircaloy-4 and Zircaloy-2P with similar silicon and tin contents irradiated in Ringhals-3 to 28 GWd/MTU indicates a 20% improvement in corrosion resistance of Zircaloy-2P with respect to low-tin Zircaloy-4 (Ref. 2). Oxide thickness measurements made on Palo Verde Unit 3, Batch F fuel rods after the [

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The composition of the Alloy 4 included in this program is based on considerations of [

]. Figure 2 shows the beneficial effects of [ ] on the corrosion resistance of zirconium based alloys. Although at a constant tin content the [ ] addition results in a slight degradation of the autoclave corrosion resistance, [ ] addition to zirconium-based alloys is known to reduce the [ ] of the alloys. Also, there are indications in the literature that [ ] addition improves in-reactor corrosion resistance and [ ] that are not apparent in out-of-reactor tests (Ref. 7). Furthermore, the specific composition of the alloy represents a further optimization for improving the mechanical strength of [ ] bearing alloys included in the two Palo Verde Unit 3, Batch F demonstration assemblies, namely Zirconium Alloys A and B, (Reference 3). [ ] fuel rods bounding the composition of this alloy showed satisfactory behavior after [ ] of irradiation in Palo Verde 3 reactor to [ ] GWd/MTU. Post-irradiation examinations recently completed on a subset of [ ] fuel rods irradiated in [ ] showed similar satisfactory results at approximately [ ] GWd/MTU burnup.

The composition of Alloy 5 is based on the long-term autoclave corrosion test results which have shown beneficial effect of [ ] in zirconium-based alloys. Data reported in Reference 3 showed significant improvement in corrosion resistance [ ] that could be obtained by reducing [ ] to values that are significantly [ ] than is normally present in Zircaloy-4. Supporting data demonstrating the beneficial effect of [ ] content have also been reported by others (Refs. 8 and 9). Since alloys containing such [ ] concentrations of [ ] are known to have lower mechanical strength, the [ ] content of the alloy was adjusted to provide adequate mechanical strength without degrading the corrosion resistance. Weight gain data on an alloy [ ] very similar in composition to the Alloy 5 and used in Palo Verde 3 Batch F are plotted in Figure 2. The data show this type of alloy [ ] to exhibit the least weight gain and suggest a factor of [ ] reduction in oxide thickness that is achievable in reactors.

In-reactor corrosion behavior on alloys with compositions close to that of Alloy 5 is available from at least [ ] reactors and is summarized in Table 3. Earliest in-reactor confirmation of significantly superior corrosion resistance for this class of alloys [

].

In reactor corrosion, creep and growth measurements made on fuel rods clad with an alloy with the [ ] within the range being tested in Palo Verde 3 (and close to the Alloy 5) have recently been reported (Ref. 11). Data are available on burnups up to [ ] GWd/MTU, and at the highest burnup showed approximately [ ] lower oxide thickness compared to Zircaloy-4 with reduced [ ] that was used by the investigators as a reference. The data obtained from Palo Verde 3, after the [ ] of irradiation to [ ] GWd/MTU are consistent with the above trend.

Alloy 6 [

]. In-reactor performance of this alloy has been reported in a number of publications [ ] and is summarized in Table 3. Except for [

] (Ref. 14). The alloy has been irradiated in test and power reactors to burnups and fluences of over [

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[ ] to meet the mechanical property requirements that are normally used for the Zircaloy-4 cladding. The upper limits for impurity concentrations [ ] have also been modified to closely correspond to limits generally specified for Zircaloys.

Cladding tubes fabricated to [ ] have been tested under an accelerated corrosion test [ ]. Figure 4 compares the weight gain with the optimized low-tin Zircaloy-4 reference. The data show [

]. For comparison, the weight gain data on [

] are also included in Figure 4. After [ ] days of autoclave testing, the measured weight gain of this alloy compares [

] with the data reported for [ ].

2.1.2 Mechanical Properties

Although the compositions of the cladding variants listed in Table 2 differ from the composition of Zircaloy-4, the mechanical properties in the as-fabricated state for all the variants to be tested were specified to meet the requirements of the fuel cladding specification used to procure commercial Zircaloy-4.

The mechanical properties of the as-fabricated tubes of all the variants to be tested were measured to assure compliance with the minimum strength and ductility properties of Zircaloy-4 both at room and elevated temperatures. [

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The addition of [ ] at PWR operating temperature (Ref. 9). Based on [ ]. Fuel rods clad with [ ] of various compositions (containing significantly [ ] than the lower limit for Zircaloys) have been successfully irradiated in a high-temperature PWR, Goesgen, up to 70 GWd/MTU and results have been published for burnups up to 60 GWd/MTU. Irradiations of similar cladding alloys were also reported to be in progress in another high-temperature PWR, Grohnde (Ref. 9).

[ ].

[ ].

Similarly, [ ] to levels above the upper limit for Zircaloy-4 to compensate for the reduction in strength associated with [ ]. -reactor creepdown measurements on fuel rods clad with an alloy similar to the Alloy 5 have recently been reported (Ref. 11). The data show essentially the same creepdown characteristics as that of Zircaloy-4 clad fuel rods irradiated in the same fuel assembly.

Summarizing, the new cladding materials listed in Table 2 are expected to show significant improvement in corrosion resistance compared to Zircaloy-4 under PWR operating conditions. The mechanical properties of these alloys are comparable to Zircaloy-4 and, therefore, their mechanical response is expected to be similar to that of Zircaloy-4 cladding for all anticipated operating and postulated accident conditions. Fuel rods using [ ] have been successfully irradiated in two high-temperature PWRs to burnups up to 70 GWd/MTU (Ref. 9).

2.1.3 Other Material Properties

All the other material properties of the selected cladding alloys relevant to the plant design bases are essentially the same as those of Zircaloy-4. This is a direct result of adjustment in the levels of different [ ] and appropriate changes in the [ ]. All cladding materials listed in Table 2 are certified to satisfy the requirements of the current Zircaloy-4 cladding specification. The following properties of the advanced alloys are similar to those of Zircaloy-4: room and elevated temperature yield strength, tensile strength and ductility; and surface roughness. In addition, the following properties are expected to be essentially the same as those of Zircaloy-4: modulus of elasticity, and hardness.

In-reactor growth measured on fuel rods in Palo Verde Unit 3, Batch F demonstration assemblies and other published data (Ref. 11) suggest that the growth behaviors of Alloys 2, 3, 4 and 5 are bounded by the observed in-reactor growth behavior of Zircaloy-4 clad fuel rods. Reported in-reactor growth of [

] compared to Zircaloy-4 (Refs. 4, 12 and 14). However, since a [ ] is generally considered beneficial with respect to the safety analyses, the growth behavior of the alloys is bounded by Zircaloy-4 behavior.

The total amount of alloying elements added to zirconium in each of the alloys listed in Table 2 is similar to that of Zircaloy-4. The thermal conductivity of the alloys to be tested is therefore expected to be within the range observed for Zircaloy-4.

2.2 Safety Analysis

2.2.1 Cladding Behavior Under LOCA Conditions

The behavior of the new cladding materials under LOCA transient conditions was evaluated. The two critical cladding material properties which affect fuel rod performance during the LOCA transient are high-temperature oxidation and deformation under transient conditions (ballooning). Most of the high temperature oxidation occurs in the β-phase since the diffusion coefficient for oxygen in the β-phase of zirconium is significantly greater than that in α-phase zirconium. Transient deformation (ballooning), on the other hand, mostly occurs in the high-temperature α-phase region prior to rupture. The following discussion presents a comparison of the expected behavior of the alloys listed in Table 2 with that of Zircaloy-4.

For Alloy 2 containing [ ] for Zircaloy-4 and Alloy 3 with [ ], compared to Zircaloy-4 composition, both the high-temperature oxidation behavior and ballooning strain behavior are not expected to be different from those of Zircaloy-4. The similarity in the measured properties and other structural features of these two alloys to that of Zircaloy-4 indicates that such [ ] variations will not cause any degradation in the behavior of the fuel rods clad with these two alloys from that of the Zircaloy-4 clad rods under the LOCA transient conditions. Therefore, LOCA models approved for Zircaloy-4 are considered applicable to these alloys. The high temperature oxidation and ballooning behavior of the remaining alloys are discussed below.

The extent of ballooning during a LOCA transient depends on the temperature at which maximum stress (and, therefore, rupture) is experienced by the cladding. For the majority of the LOCA-type transients, cladding rupture is predicted to occur in the [ ]. The relationship of this temperature to the $\alpha/(\alpha+\beta)$ phase transformation boundary temperature is important since the extent of the superplasticity elongation peak, which affects the potential for ballooning, depends on the presence of the β-phase as well as the extent of oxidation during deformation as described below. There is a superplasticity elongation peak near the $\alpha/(\alpha+\beta)$ boundary in Zircaloy-4 (Ref. 17). The magnitude of this peak depends on the extent of oxidation of the material prior to rupture. A higher rate of oxidation near the $\alpha/(\alpha+\beta)$ phase boundary region will decrease the magnitude of this elongation peak.

Among the different alloying elements added to the cladding alloys listed in Table 2, [ ] have significant solubility in α-zirconium. [ ] are α-phase stabilizers and [ ] is a β-phase stabilizer. [ ] have limited solubility in α-phase but both elements are β-stabilizers. As a result of these properties, changes in the relative concentrations of these alloying elements will change the $\alpha/(\alpha+\beta)$ phase boundary temperature. A decrease in [

] is expected to lower the $\alpha/(\alpha+\beta)$ transition temperature. This will affect both the oxidation rate in the high-temperature α-phase region and the ballooning behavior. Since oxygen diffusion in the β-phase is significantly faster than that in the α-phase, a lowering of the $\alpha/(\alpha+\beta)$ interphase temperature tends to increase the oxidation rate at the temperature of interest because of the proximity to the $\alpha/(\alpha+\beta)$ phase boundary with respect to conventional Zircaloy-4. With the lowering of the $\alpha/(\alpha+\beta)$ interphase temperature, the superplastic elongation peak is also expected to shift to lower temperatures. However, the increase in the elongation due to the shift of the superplasticity peak will be compensated for by the decrease in deformation due to a higher extent of oxidation. The net effect is that there is no significant change in the ballooning behavior of Alloy 4 compared to Zircaloy-4. Scoping studies were conducted to examine the burst behavior at high temperature and the results support this expected behavior (Ref. 18). The minor change in the oxidation rate near the $\alpha/(\alpha+\beta)$ phase boundary does not have a significant effect on the total extent of oxidation (which is mainly controlled by the extent of oxidation in the β-phase).

For Alloy 6, similar reasoning as for Alloy 4 applies. In this case, [ ] content overlaps the [ ] of the normal range for Zircaloy-4. A lowering of the $\alpha/(\alpha+\beta)$ transition temperature is expected due to addition of [ ] than Zircaloy-4. However, no significant change in ballooning behavior compared to Zircaloy-4 is expected because of the compensating effect of higher oxidation on the shift of the superplastic elongation peak to a lower temperature. It is also worth noting that [ ] has been approved by the NRC as an acceptable zirconium based cladding material with respect to meeting acceptance criteria under 10 CFR 50.44, 50.46 and Appendix K to 10CFR Part 50 regarding evaluations of emergency core cooling systems and combustible gas control [ ].

For Alloy 5, the decrease in the $\alpha/(\alpha+\beta)$ transition temperature due to [ ] is expected to be compensated by the increase in the $\alpha/(\alpha+\beta)$ temperature due to higher [ ]. Therefore, the $\alpha/(\alpha+\beta)$ temperature for this alloy is not expected to be different than that for Zircaloy-4. Moreover, the effect of [

] (Ref. 20). The net effect in Alloy 5 is predicted to be an insignificant change compared to the reference Zircaloy-4. Based on these considerations, the ballooning behavior of Alloy 5 is considered essentially the same as that of Zircaloy-4. High-temperature rupture behavior of [ ] under conditions representative of those imposed on the cladding during a typical loss of coolant accident transient was reported in Reference 11. The rupture behavior was found to be within the scatter of the reference Zircaloy-4 cladding used by the investigators as a control.

The extent of the total oxidation during the LOCA transient includes the oxidation prior to cladding rupture and the oxidation occurring after rupture. Since the latter part occurs mainly in the ß-phase region where the oxygen diffusion coefficient is high, it contributes a major fraction to the total oxidation. Therefore, oxidation in the ß-phase controls the extent of oxidation of the cladding during the LOCA transient. Based on the comparison of oxidation of Zircaloy-4 and [ ] alloys described below, it is concluded that [ ] alloys described in Table 2 [ ] to that of Zircaloy-4 in this high temperature region (up to ~1200°C). A comparison of the high temperature (1000 to 1850°C) oxidation of [ ] with that of Zircaloy-4 reveals that the oxidation rates of these materials are comparable and that the Baker-Just correlation conservatively overpredicts the oxidation of both types of materials (Ref. 21). (The Baker-Just correlation is used to calculate the extent of high-temperature oxidation of fuel cladding during the high-temperature transients, per the requirement of Appendix K to 10 CFR Part 50). The composition change from Zircaloy-4 to [

] containing alloys listed in Table 2. It is,therefore, concluded that the Baker-Just correlation will overpredict the oxidation behavior of the Alloys [ ].

The β-phase oxidation resistance of the Alloy 5 is expected to be as good as or better than that of Zircaloy-4. It is expected that the alloying element levels adjusted to improve the corrosion resistance of the α-phase of this alloy with respect to the α-phase of Zircaloy-4, will result in an improvement of the corrosion resistance of the β-phase for this alloy as well. It is, therefore, concluded that the β-phase oxidation rate of all the alloys listed in Table 2 is comparable to or lower than that of Zircaloy-4 and that the Baker-Just correlation will overpredict the β-phase oxidation of all the advanced alloys.

Summarizing, the behaviors of all the alloys proposed to be included in the Calvert Cliffs Unit 1 Batch R LFAs are essentially the same as that of conventional Zircaloy-4 under all conditions experienced during both normal operation and under the conditions existing during a LOCA transient. Therefore, the 10 CFR 50.44 and 50.46 criteria will be satisfied for these proposed alloys.

2.2.2 Cladding Behavior Under Non-LOCA Conditions

Consideration was also given to the behavior of the advanced cladding materials under non-LOCA conditions. These conditions include normal operation, Anticipated Operational Occurrences, (AOOs), and postulated accidents other than LOCA. Cladding properties/features that impact fuel behavior during non-LOCA conditions are:

[ ]

As mentioned in Section 2.1.3, material properties and characteristics of the non-Zircaloy cladding alloys at the operating clad temperatures for non-LOCA conditions are expected to be similar to those of Zircaloy-4. Therefore, the properties which could impact the Non-LOCA conditions shown in the table above, will be essentially the same as or better than [ ] the current Zircaloy-4 properties used in the licensing analyses.

The range of clad operating temperatures used for the design and licensing analyses for normal operation and AOOs is quite small compared to the range that is covered for LOCA analyses. For these conditions, the probability of fuel failure is exceedingly low because the DNB Specified Acceptable Fuel Design Limit (SAFDL) must be satisfied. The DNB SAFDL is established such that there is at least a 95% probability at a 95% confidence level that the limiting fuel rod in the core does not experience DNB. Clad surface temperatures during nucleate boiling (no DNB) can only be a few degrees above the coolant saturation temperature. Furthermore, the heat fluxes must be below the critical heat flux at which DNB would occur. Therefore, the inside clad temperature can be no more than

[ ] above the outside temperature. At these relatively low temperatures, no phase change in the zirconium alloy cladding is expected, further assuring that all important material properties will be similar to Zircaloy-4. [ ] in the creep rates could influence the time for the clad to creepdown on the fuel pellet, but this would have [ ] effect on the bounding values of maximum and minimum gap conductance that are used in the transient and safety analyses.

For postulated non-LOCA accidents the critical heat flux for DNB should not be affected by [ ], except as those differences could affect subchannel geometries, as in [ ], for example. Since the cladding properties that are expected to influence [ ] are similar to Zircaloy-4, there should be virtually no difference in [ ]. Consequently, the number of fuel failures predicted for the non-LOCA accidents would remain essentially unchanged and continue to be well below acceptance criteria even if a large region were to be composed of one or more of the proposed cladding variants. In the unlikely event that cladding failure occurs in the lead fuel assemblies, the nature and consequences of the failure occurring in the non-Zircaloy-clad fuel rods are no more adverse than those of Zircaloy-clad fuel rods. As a result, the environmental impact would remain unchanged and is bounded by previous assessments.

Based on the above considerations, cladding behavior under non-LOCA conditions is expected to remain essentially unchanged as a result of introducing a limited number of [ ] non-Zircaloy-clad fuel rods into the Calvert Cliffs Unit 1 Cycle 13 core.

3.0 EVALUATION CONCLUSIONS

The preceding discussions describe why the predicted chemical, mechanical, and material properties of the selected zirconium alloys fall within the range of the properties for Zircaloy-4 under all anticipated operating conditions, including those considered in the safety analysis. Therefore, it is concluded that the fuel rod design bases currently used for the design and analysis of the standard Zircaloy-4 clad rods will also be applicable to the fuel rods clad in the selected alloys which are to be included in the Calvert Cliffs Unit 1, Batch R LFAs. Furthermore, these LFAs and the fuel rods clad with the non-Zircaloy cladding materials will be placed in non-limiting core locations which experience no more than 0.95 of the highest core power density through the irradiation periods, as indicated in Section 1. Thus, the nominal fuel performance characteristics of the advanced alloys will be essentially the same as those observed for other fuel rods. Since the current design bases are applicable to the proposed non-Zircaloy cladding variants and the expected operating conditions are within those assumed for the standard clad rods currently licensed for Calvert Cliffs Unit 1, it is concluded that the licensing basis currently in effect will not be compromised by incorporating a limited number [ ] of non-Zircaloy-clad fuel rods.

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(19) Federal Register, Vol. 57, No. 169, Monday, August 31, 1992, page 39355.

(20) B. Burton, A.T. Donaldson and G.L. Reynolds, ASTM STP 681, 1979, pp561-585.

(21) V.F. Urbanic, ASTM STP 633, 1977, pp168-181.

Table 1

Advanced Cladding Variants to be Used in
Calvert Cliffs Unit 1, Batch R Lead Fuel Assemblies[a]

[ ]

Table 2

Chemical Compositions of Cladding Alloys For Use in Calvert Cliffs Unit 1, Batch R LFAs and Comparison with Zircaloy-4 and Previously Approved[a] Zirconium Alloys in Palo Verde Unit 3, Batch F Demonstration Assemblies

Alloy Number	Alloy Designation	Sn%	Fe%	Cr%	[ ]	O%	Ni%	Zr
1.								Bal.
								"
2.								"
								"
3.								"
4.								"
								"
								"
5.								"
6.								"
								"

[a] Per References [ ].

TABLE 3

SUMMARY OF IRRADIATION EXPERIENCE OF ADVANCED CLADDING

FIGURE 1

Oxide Thickness Measured on Palo Verde 1 Fuel Rods

(Effect of Tin Content and Fabrication)

FIGURE 2

Comparison of Corrosion Resistance of Zirconium-Based Alloys
With Optimized Low-Tin Zircaloy-4
(ASTM G-2, 360°C Water Static Autoclave)

[ ]

[ ]

FIGURE 3

Comparison of Corrosion Resistance of Zircaloy-4 Variants

[ ]

[ ]

—■— EPRI STD —▲— OPTIMIZED LOW-TIN Zr-4

FIGURE 4

Comparison of Corrosion Resistance of Alloy 6
With Optimized Low-Tin Zircaloy-4 and[ ]
(360°C Water With 70 ppm Li, Static Autoclave)

[ ]

-■- Alloy 6 -♦- Optimized Low-Tin Zr-4 (Alloy 1)

-▲- [ ]

ATTACHMENT (4)

PROPRIETARY AFFIDAVIT FOR ATTACHMENT (5)

AFFIDAVIT PURSUANT

TO 10 CFR 2.790

Combustion Engineering, Inc.)
State of Connecticut)
County of Hartford) SS.:

I, S. A. Toelle, depose and say that I am the Manager, Nuclear Licensing, of Combustion Engineering, Inc., duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conjunction with the application of Baltimore Gas & Electric Company and in conformance with the provisions of 10 CFR 2.790 of the Commission's regulations for withholding this information.

The information for which proprietary treatment is sought is contained in the following document:

> CEN 425-P, Revision 2-P, "Safety Evaluation Report for Use of Advanced Zirconium-Based Cladding Materials in Calvert Cliffs, Unit 1 Batch R Lead Fuel Assemblies," March 1995.

This document has been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by Combustion Engineering in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1. The information sought to be withheld from public disclosure, which is owned and has been held in confidence by Combustion Engineering, is the identification and chemical composition of zirconium-based alloys and the test and performance data on these alloys which demonstrate their superior corrosion resistance compared to Zircaloy-4.
2. The information consists of test data or other similar data concerning a process, method or component, the application of which results in substantial competitive advantage to Combustion Engineering.
3. The information is of a type customarily held in confidence by Combustion Engineering and not customarily disclosed to the public. Combustion Engineering has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The details of the aforementioned system were provided to the Nuclear Regulatory Commission via letter DP-537 from F. M. Stern to Frank Schroeder dated December 2, 1974. This system was applied in determining that the subject

documents herein are proprietary.

4. The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.790 with the understanding that it is to be received in confidence by the Commission.

5. The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.

6. Public disclosure of the information is likely to cause substantial harm to the competitive position of Combustion Engineering because:

   a. A similar product is manufactured and sold by major pressurized water reactor competitors of Combustion Engineering.

   b. Development of this information by Combustion Engineering required thousands of manhours and millions of dollars. To the best of my knowledge and belief, a competitor would have to undergo similar expense in generating equivalent information.

   c. In order to acquire such information, a competitor would also require considerable time and inconvenience to develop or discover the identification and chemical composition of zirconium-based alloys and the test and performance data on these alloys which demonstrate their superior corrosion resistance compared to Zircaloy-4.

d. The information required significant effort and expense to obtain the licensing approvals necessary for application of the information. Avoidance of this expense would decrease a competitor's cost in applying the information and marketing the product to which the information is applicable.

e. The information consists of the identification and chemical composition of zirconium-based alloys and the test and performance data on these alloys which demonstrate their superior corrosion resistance compared to Zircaloy-4, the application of which provides a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with Combustion Engineering, take marketing or other actions to improve their product's position or impair the position of Combustion Engineering's product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

f. In pricing Combustion Engineering's products and services, significant research, development, engineering, analytical, manufacturing, licensing, quality assurance and other costs and expenses must be included. The ability of Combustion Engineering's competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

g. Use of the information by competitors in the international marketplace would increase their ability to market nuclear steam supply systems by reducing the costs associated with their technology development. In addition, disclosure would have an adverse economic impact on Combustion Engineering's potential for obtaining or maintaining foreign licensees.

Further the deponent sayeth not.

S. A. Toelle

S. A. Toelle
Manager
Nuclear Licensing

Sworn to before me
this 26th day of April, 1995

Laurie J. White
Notary Public

My commission expires: 8-31-99