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SAFETY ANALYSIS REPORT

FOR

USE OF ADVANCED ZIRCONIUM BASED CLADDING MATERIAL

IN PVNGS UNIT 2 BATCH J

DEMONSTRATION FUEL ASSEMBLIES

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ABB COMBUSTION ENGINEERING

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Safety Analysis Report For Use of Advanced Zirconium Based Cladding Material in Palo Verde Unit 2 Batch J Demonstration Fuel Assemblies

1.0 INTRODUCTION

With the recent trends in the nuclear industry regarding increased fuel discharge burnups and longer exposure cycles, the corrosion performance requirements for nuclear fuel cladding are becoming more demanding. Added to this are desires for axial blankets, and increased core power. Under these more demanding operating conditions, Zircaloy-4, the commercially used fuel cladding material, may not be the best material to provide in the near future the desired operational flexibility and performance margins. To meet these needs, ABB CENO has developed new cladding materials with improved corrosion resistance. As part of this development program, several promising zirconium based cladding alloys were included in two Lead Fuel Assemblies (LFAs) in Batch F of Palo Verde Unit 3 and began irradiation in Cycle 4. Currently these LFAs assemblies are in their second cycle of irradiation and are planned for a third cycle of irradiation in Cycle 6. Further, two new LFA are being planned for insertion at the beginning of Cycle 7 in Palo Verde Unit 2 to test still more promising clad zirconium-based alloy.

This Safety Analysis Report (SAR) addresses insertion of two new LFAs using test fuel rods clad with [ ] more alloy in Palo Verde Unit 2 fuel Batch J. The alloy has shown significant promise in in-reactor and ex-reactor test programs. This [ ] and is designated in this report as Zirconium Alloy F. It was originally developed [ ].

This report describes the composition and properties of [ ]. It also presents irradiation experience with the alloy that supports expectation of superior material performance and provides a safety analysis for fuel rods clad with this alloy.

[ ] has shown superior in-reactor corrosion resistance, lower irradiation growth, lower hydrogen uptake and superior microstructural stability under irradiation. As a continuation of the clad development program, two LFAs containing up to [ ] fuel rods in each assembly are planned for insertion at the upcoming refueling at Palo Verde 2 in fuel Batch J.

Inclusion of 2 LFAs with [ ] fuel rods clad with non-Zircaloy-4 tubes in Unit 2 Cycle 7 is consistent with the Technical Specification (5.2.1, Fuel Assemblies) that allows limited inclusion of LFAs in non-limiting core locations. The composition of [ ], an alloy that the NRC has approved for use in PWRs.

Fuel rods using [ ] cladding are identical in design and dimension to other fuel rods in the core. These rods also contain UO_2 fuel pellets of the same enrichment as the Zircaloy-4 rods in the LFAs. The LFAs will be positioned in the core such that the rods will experience significant burnup and power density, but will not experience the highest core power density. That is, the LFAs will be placed in a non-limiting, core location.

This placement scheme, and the similarity of [ ] performance to Zircaloy-4, assures that the behavior of the lead fuel rods will be bounded by the fuel performance and safety analyses performed for Zircaloy-4 clad fuel rods. Visual examinations and eddy current oxide thickness measurements are planned to be conducted at the end of each operating cycle to obtain indications of any abnormal behavior. The reconstitutable upper end fitting feature incorporated in the fuel assembly allows reconstitution in the unlikely event that any indication of unsatisfactory performance is detected during the interim examinations.

2.0 EVALUATION

2.1 Alloy Composition

The major alloying elements in [ ] are given in Table 1. The composition of low-tin Zircaloy-4 is also given for comparison. With the exception of [ ], the composition is similar to Zircaloy-4. The upper limits for certain impurity concentrations [ ] have also been modified to correspond to limits specified for Zircaloys. [ ] alloy based on an alloy composition that was [ ]. Except for [ ]

2.2 Irradiation Experience

In-reactor performance of this alloy and ex-reactor test results have been reported in a number of publications (Refs. 2-8).

The alloy has been irradiated in test and power reactors to burnups and fluences of over [ ] respectively and shows good corrosion resistance combined with low irradiation growth, high in-reactor creep resistance, high residual ductility and a stable microstructure with respect to the effect of radiation damage. [ ] In Zircaloy-4, in-reactor growth and creep mechanisms are related to the formation of this type of dislocation loops at high burnups.

Approximately [ ]

Experimental fuel rods have also been irradiated under [ ]

In addition to these fuel rod irradiations, non-fueled samples have also been irradiated to investigate growth, creep and other properties. Specimens with composition identical to [ ] The ductility of this alloy was found to be superior to that of Zircaloy when specimens of the two alloys were irradiated in the [ ]. The pellet cladding interaction (PCI) resistance of the same alloy is expected to be better than that of Zircaloy-4 based on mechanical test results in [ ]. These data support the expectation of superior material performance of the new alloy compared to Zircaloy-4 (i.e., reduced in-reactor growth, and creep, and greater corrosion resistance). Data have also been published on the satisfactory growth, creep, and waterside corrosion behavior of ZIRLO to burnups of 46 GWd/MTU (Ref. 1 & 9). [ ]

Cladding growth, creep and waterside corrosion are the key performance concerns relative to normal fuel rod operation. Information available to address the performance concerns during normal operation is summarized below.

Growth

The irradiation induced growth of fuel rods clad with [ ] is expected to be [ ] than that for Zircaloy-4. Stress-free growth data obtained on tubular samples irradiated in [ ] The growth strain of these specimens is very well behaved and [ ] at the maximum exposure. In contrast, the stress free growth for Zircaloy-4 at a similar fluence is on the order of 1% $\Delta L/L$ (Ref. 10).

Creep

The creep strain of [ ] tubing has been evaluated in a test reactor under an internal pressure of [ ] This is less creep than would be expected for Zircaloy-4 in an out-of-reactor test at the similar internal pressure and deformation temperature. It is well known that in-reactor creep is greater than that from ex-reactor tests due to an additional irradiation component. The Zircaloy-4 shown in the figure was an early vintage that was fabricated in [.] than what is observed in Zircaloy.

Waterside Corrosion

In-reactor corrosion data on fuel rods irradiated in the [ ] conditions also demonstrate good corrosion resistance. Metallographic cross-sections were taken on several of these rods to characterize oxide film thickness. The rods examined have average burnups from [ ] days of operation. The average oxide film thicknesses measured at the maximum burnup locations ranged from approximately [ ] microns. It is important to point out that these rods ran at very high power during the in-reactor tests which would accelerate corrosion. The linear heat generation rates (LHGR) in the regions where these particular oxide measurements were made, ranged from [ ] Similar metallographic evaluations were performed on selected fuel rods taken from the [ ] and rods irradiated in a test reactor under [ ]

conditions. These investigations also showed good resistance to nodular corrosion. Nodular corrosion is a localized, accelerated corrosion phenomenon experienced in BWR's. It is believed that the corrosion resistance of this material under both BWR and PWR conditions is attributable to the [ ] that are formed during fabrication.

Non-fueled samples irradiated in the [ ] full power days showed corrosion resistance at least equivalent to Zircaloy. However, the accumulated in-reactor exposure at [ ] was too short to detect any significant differences in oxidation behavior. Measurements of hydrogen content showed that the pickup fraction was similar to that of Zircaloy-4.

Summarizing, in-reactor performance data available for [ ] demonstrate very good [ ] to high fluences. In-reactor corrosion data are available on fuel rods irradiated in the [ ] reactor to substantial burnups and non-fueled specimens irradiated at [ ]. These data show [ ] to have superior in-reactor corrosion properties. (It should also be noted that significant data have also been published on [ ]

2.3 Autoclave Corrosion Results

Results from ex-reactor corrosion tests are available which show that this alloy also has promising corrosion properties. [ ] tubing has also been fabricated to meet the ABB CENO specification requirements for Zircaloy-4 and subjected to an accelerated corrosion test [ ]. Figure 3 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 3. After 410 days of autoclave testing, the measured weight gain of this alloy [ ]. This particular ex-reactor corrosion test is recognized as being suitable for qualitatively ranking the relative in-reactor corrosion behavior of zirconium alloys containing [ ].

2.4 Mechanical Properties

The [ ] has been modified to meet the mechanical property requirements of the ABB CENO specification for the Zircaloy-4 cladding.

The mechanical properties of the as-fabricated tubes of [ ] are measured to assure compliance with the minimum strength and ductility properties of Zircaloy-4 both at room and elevated temperatures. The addition of [

] Moreover, the addition of [ ] is expected to improve the overall in-PWR performance of this alloy as a result of improved irradiated ductility (Ref. 11) and lower hydrogen pickup fraction. Improved irradiated ductility was measured on specimens of similar composition irradiated in [ ] reactor as discussed earlier (Ref. 3).

The addition of [

] at PWR operating

temperature (Ref. 12).

Based on irradiation data [ ], in-reactor creep rate for [ ].

2.5 Safety Analysis

2.5.1 Cladding Behavior Under LOCA Conditions

The behavior of [ ] 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 (i.e, 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 [ ] with that of Zircaloy-4.

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 high-temperature α-phase region around [ ]. 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. 13). 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 present in [ ] 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 [ ] compared to Zircaloy-4 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 decease in deformation due to a higher extent of oxidation. The net effect is that there is no significant change in the ballooning behavior of [ ] compared to Zircaloy-4. 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).

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 Part 50 regarding evaluations of emergency core cooling systems and combustible gas control (Ref. 14).

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 β-phase region where the oxygen diffusion coefficient is high, it contributes a major fraction to the total oxidation. Therefore, oxidation in the β-phase controls the extent of oxidation of the cladding during the LOCA transient. Based on the comparison of oxidation of Zircaloy-4 and Zr-2.5 wt% Nb alloys described below, [ ] 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 zirconium -2.5 wt% niobium alloy with that of Zircaloy-4 reveals that the oxidation rates of these materials are comparable and that the Baker-Just correlation conservatively over predicts the oxidation of both types of materials (Ref. 15). (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 Zr-2.5% Nb is [ ] It is, therefore, concluded that the Baker-Just correlation will over predict the oxidation behavior of [ ]

Summarizing, the behavior of [ ] proposed to be included in the Palo Verde 2J LFAs is expected to be superior to that of conventional Zircaloy-4 under all conditions experienced during both normal operation and under the conditions existing during a LOCA transient. (In any case the behavior would be at least equivalent to that of Zircaloy-4). Therefore, the 10 CFR 50.44 and 10 CFR 50.46 criteria will be satisfied for this alloy.

2.5.2 Cladding Behavior Under Non-LOCA Conditions

Consideration was also given to the behavior of [ ] 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:

Material properties and characteristics of [ ] 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 [ ] 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 [ ] temperatures, no phase change in the zirconium alloy cladding is expected, further assuring that all important material properties will be similar to Zircaloy-4. [ ] differences in the creep rates could influence the time for the clad to creepdown on the fuel pellet, but this would have [ ]that are used in the transient and safety analyses.

For the postulated non-LOCA accidents, the design limit that separates failed rods from non-failed rods is DNB. The critical heat flux for DNB should not be affected by small differences in cladding composition, except as those differences could affect sub-channel geometries, as in [ ] for example. Engineering factor such as the pitch and bow factor accommodate

large variations from " nominal design " values. Since the properties that are expected to influence [

] compared to Zircaloy-4, the expected rod [ ] fuel rods is expected to be [ ] than that for Zircaloy-4 rods. Consequently, the number of fuel failures predicted for the non-LOCA accidents would remain essentially unchanged (or decrease compared to Zircaloy-4) and continue to be well below acceptance criteria. In the unlikely event that cladding failure occurs in the lead fuel assembly, the nature and consequences of the failure occurring in the [ ] 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 result of introducing [ ] clad fuel rods into the a LWR core.

2.6 EVALUATION CONCLUSIONS

The preceding discussions describe why the predicted chemical, mechanical, and material properties and performance expectations of [ ] 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 are also applicable to the fuel rods clad in [ ]. Furthermore, LFA fuel rods clad with [ ] 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. Thus, the nominal fuel performance characteristics of the [ ] clad fuel rods will be essentially the same as those observed for other fuel rods. Since the current design bases are applicable to the proposed [ ] and the expected operating conditions are within those assumed for the standard clad rods currently licensed for Palo Verde Unit 2, it is concluded that the licensing basis currently in effect will not be compromised by incorporating a limited number, [ ] clad fuel rods.

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Table 1

Range Comparison of Major Alloying Elements in [ ] and Zircaloy-4

[ ]