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SAFETY ANALYSIS REPORT

FOR

USE OF ADVANCED ZIRCONIUM BASED CLADDING MATERIALS

IN PVNGS UNIT 3

LEAD FUEL ASSEMBLIES

AUGUST 1996

ABB COMBUSTION ENGINEERING

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Safety Analysis Report For Use of Advanced Zirconium Based Cladding Materials in Palo Verde Unit 3 Lead 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 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 for irradiation in Cycles 4, 5 and 6.

Currently these LFAs are in their third cycle of irradiation and some test rods from these assemblies clad with Zirconium Alloy A have shown excellent corrosion performance through two cycles of irradiation. Further, two LFAs were inserted at the beginning of Cycle 7 in Palo Verde Unit 2 to test in-PWR performance of another advanced clad alloy, Zirconium Alloy F [ ]. Lead test assemblies with several advanced cladding alloys, including Alloy F, are also currently undergoing their first irradiation cycle in Calvert Cliffs Unit 1 as part of Fuel Batch R.

This Safety Analysis Report (SAR) addresses insertion of two new LFAs, one comprised entirely of fuel rods clad with Zirconium Alloy A and the other comprised entirely of fuel rods clad with Zirconium Alloy F, in Palo Verde Unit 3 fuel Batch J, beginning with Cycle 7. Currently, these assemblies are scheduled for three cycles of irradiation, i.e., Cycles 7, 8 and 9. Peak rod burnup on these LFAs will be limited to 60 GWd/MTU. This SAR also addresses a third LFA, a Batch F carrier assembly, into which up to [ ] test rods clad with Zirconium Alloy A and up to [ ] test fuel rods clad with OPTIN™ which were irradiated during Cycles 4, 5 and 6 are planned to be transplanted. This third LFA will only be irradiated for one cycle, during Cycle 7 of Palo Verde Unit 3. The projected burnup of these [ ] transplanted test fuel rods in the third LFA will exceed the current burnup limit for Palo Verde fuel during Cycle 7. Examinations that will be performed on these test rods at the end of cycle (EOC) 6 for determining their acceptability for continued irradiation and an evaluation performed with respect to their exceeding the current burnup limit are presented in Appendix A. The discussions in the Appendix show that the critical performance parameters for the transplanted rods will remain within the currently approved limits (Refs. 1 and 2) for these parameters. No further discussions on the third LFA are included in the main text of the SAR.

OPTIN™ is a variant of Zircaloy-4 with tin content in the lower end of the specification range for Zircaloy-4 and fabricated with processing optimized to improve corrosion resistance in PWRs. This alloy has been developed by ABB CENO and is the current standard cladding used by ABB CENO for fabricating all PWR fuel. Superior corrosion resistance of OPTIN™ cladding in contrast to that of commercial Zircaloy-4 was demonstrated previously through several cycles of irradiation in Palo Verde reactors (Ref. 3).

Data obtained from the three LFAs is intended to support future licensing of Zirconium Alloy A and Zirconium Alloy F. Both alloys were selected for fabrication of all clad materials in two fuel assemblies (one full assembly with each alloy) because they have shown excellent performance in in-reactor and ex-reactor test programs.

Initially, eight (8) test fuel rods clad with Zirconium Alloy A were inserted in Cycle 4 of Palo Verde Unit 3 in 1992. A safety evaluation report was submitted to the NRC (Ref. 4), and on that basis, an exemption was authorized by the NRC (Ref. 5) for irradiation of 80 fuel rods clad with non-Zircaloy-4 variants in two Palo Verde Unit 3, Batch F LFAs. The Zirconium Alloy A test fuel rods were subsequently examined at poolside during the refueling outages after one cycle and two cycles of exposure. Data from these examinations, presented in Reference 6, supports the fabrication and irradiation of a new LFA consisting of up to 236 fuel rods clad with this alloy starting in Cycle 7 of Unit 3. Additional discussions regarding performance to date is included in Section 2.2 of this report. A representative number of Batch F test rods clad with Zirconium Alloy A will be examined at poolside at the end of the current cycle (EOC 6) to confirm the continued superior in-PWR performance of Alloy A.

Other testing of Zirconium Alloy F includes [ ] test rods which were inserted in two LFAs in Calvert Cliffs Unit 1, Batch R fuel and [ ] test rods which were inserted in two LFAs in Palo Verde Unit 2, Batch J fuel. SARs associated with the use of Zirconium Alloy F were submitted to the NRC for these two LFA programs (Refs. 7 and 8). The appropriate exemptions were authorized by the NRC via References 9 and 10.

This SAR presents new data on the two advanced alloys that became available since the submittal of the previous SARs to the NRC (Refs. 4, 7 and 8). This data supports in-PWR irradiation testing of a larger number of test rods in two LFAs for three cycles, as well as the continued irradiation of up to [ ] fuel rods clad with Zirconium Alloy A in a third LFA for a fourth cycle.

Irradiation of these LFAs clad with non-Zircaloy-4 materials is consistent with the conservative approach of verifying the performance of a larger number of fuel rods clad with the advanced alloys prior to a full batch application of the new alloys. The inclusion of three LFAs in the Palo Verde Unit 3 Cycle 7 core is also consistent with the Technical Specification, 5.2.1, Fuel Assemblies, that allows a limited number of LFAs in non-limiting core locations. The LFAs will be positioned in the core such that the rods will experience no more than [ ] of the highest core power density through the irradiation periods.

This placement scheme, and the similarity of Zirconium Alloy A and Zirconium Alloy F performance to that of Zircaloy-4, assure that the behavior of the non Zircaloy-4 fuel rods will be bounded by the fuel performance and safety analyses performed for Zircaloy-4 clad fuel rods. Visual examinations will be conducted at the end of each operating cycle to obtain indications of any abnormal behavior. These examinations may also be supplemented with eddy current oxide thickness measurements. 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 composition of the two advanced alloys were given in the SARs (Refs. 4, 7 and 8) previously submitted to the NRC and are repeated in Table 1. The composition of OPTIN is also given for comparison.

2.1.1 Alloy A Composition

Zirconium Alloy A has a [ ] content than OPTIN and also includes the addition of [ ]. The data presented earlier in Reference 4 and updated in Section 2.3 showed a significant improvement in corrosion resistance of [ ] that could be obtained by [ ] content to values that are significantly [ ] than is normally present in OPTIN. Since alloys containing [ ] concentrations of [ ] are known to have [ ] contents were also adjusted to optimize corrosion resistance.

2.1.2 Alloy F Composition

Zirconium Alloy F contains a significant addition of [ ]. [ ] content of Alloy F is also [ ] than OPTIN. Except for differences in the [ ].

2.2 Irradiation Experience

2.2.1 Irradiation Experience of Alloy A

[ ] test rods clad with Zirconium Alloy A were irradiated for two cycles in a European PWR, [

] (Ref. 12).

[ ] test rods clad with Zirconium Alloy A are currently being irradiated for a third exposure cycle in Palo Verde Unit 3. After two cycles of exposure (approximately [ ] GWd/MTU burnup), the in-PWR performance of these Zirconium Alloy A clad fuel rods is superior to that of OPTIN- clad rods. The corrosion oxide layer thickness data are presented in Figure 1. The measured oxide thickness on Zirconium Alloy A clad rods is about [ ] lower than that for OPTIN-clad rods. It is to be noted that although [

] was noted in the examinations conducted during the two previous outages. This view is also supported by the [

].

reactors (see Fig. 1). Palo Verde Unit 3 data are from [

].

The irradiation growth strain data presented in Figure 2 show that the Zirconium Alloy A clad rods grow about [ ] less than the OPTIN-clad rods. Irradiation creep strain data presented in Figure 3 show that, [

].

2.2.2 Irradiation Experience of Alloy F

In-reactor performance of Alloy F and ex-reactor test results have been reported in a number of publications and were discussed in detail in the previous SARs (Refs. 7 and 8). Additional growth strain data following irradiation to high fluences have recently been published (Ref. 13). These results show that Zirconium Alloy F has 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. The data base covers irradiations in [ ].
Typical growth and creep data obtained on Zirconium Alloy F are included as Figures 4 and 5. In Figure 5, a comparison to [ ] Zircaloy-4 is also made for creep characteristics.

Specifically, under the PWR conditions, approximately [

] under high heat rating.
Oxide thickness measurements were made on some test rods at positions of high linear heat rating [ ] to verify superior corrosion resistance under highly aggressive conditions. The oxide thicknesses measured at these locations ranged from approximately [ ] microns over a local burnup range of approximately [ ] GWd/MTU.

Based on the absence of any indication for [ ] occurring in the Palo Verde Unit 3, Batch F LFAs, which contained fuel rods clad with Alloy A and several other cladding alloys, [[ ] is expected with the use of an LFA clad with Alloy F. Examinations reported to date on fuel assemblies containing test fuel rods clad with Alloy F also do not reveal any evidence of [ ].

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. 14).

The pellet cladding interaction (PCI) resistance of Zirconium Alloy F is expected to be better than that of Zircaloy-4 based on mechanical test results in [ ]. Cladding alloys containing no [ ]. Under the same test, specimens of composition similar to Zirconium Alloy F did not [ ]. These data support the expectation of superior material performance of Zirconium Alloy F compared to Zircaloy-4. Data have also been published on the satisfactory growth, creep, and waterside corrosion behavior of ZIRLO to burnups of 46 GWd/MTU (Refs. 11 & 16). [ ].

Test rods clad with Zirconium Alloy F have also been irradiated in [ ].

The data summarized above and the detailed evaluations of the data presented in the referenced SARs (Refs. 7 and 8) demonstrate superior in-reactor performance of Zirconium Alloy F compared to Zircaloy-4 in a number of characteristics such as waterside corrosion, dimensional stability and post-irradiation ductility.

2.3 Autoclave Corrosion Results

Results from ex-reactor corrosion tests are available which show that both alloys also have promising corrosion properties. It was shown in Refs. 7 and 8, that in 360°C water autoclave with 70 ppm lithium, the corrosion resistance of Zirconium Alloy F was significantly superior to that of Zircaloy-4. This test is highly aggressive considering that the normal range for lithium content in

Palo Verde Unit 3 reactor coolant is from [ ] ppm and the coolant outlet temperature remains below [ ].

Another autoclave test used to correlate the in-reactor corrosion resistance of Zirconium alloys is the long term autoclave test at 360°C in high pressure (2700 psi) deionized water. A comparison of corrosion performance of Zirconium Alloy A and OPTIN exposed up to approximately [ ] days was presented in a previous SAR (Ref. 4). The data is now available for additional exposure under the same conditions and is shown in Figure 6. After a total of [ ] days of autoclave exposure, Alloy A shows more than [ ] improvement (lower post-corrosion test weight gain) over OPTIN.

2.4 Mechanical Properties

The fuel rod using Zirconium Alloys A and F cladding are identical in mechanical dimensions to other standard fuel rods in the core. These rods contain UO_2 fuel pellets of the same enrichment as the other fuel rods clad with OPTIN in the core. As discussed in previous reports (Refs. 4, 7 and 8), the mechanical properties of the as-fabricated tubes of Zirconium Alloy A and Zirconium Alloy F are measured to assure compliance with the minimum strength and ductility properties of OPTIN both at room and elevated temperatures. Moreover, the addition of [

] Improved irradiated ductility was measured on specimens of composition similar to Zirconium Alloy F irradiated in the [ ] test reactor (Ref. 15).

The addition of [

] at PWR operating temperatures (Ref. 20). Based on the data shown in Figure 5 and other supporting information (Ref. 21) for Zirconium Alloy F and Figure 3 for Alloy A, the in-reactor creep rate for both advanced alloys is expected to be lower or at least comparable to that of Zircaloy-4.

2.5 Safety Analysis

2.5.1 Cladding Behavior Under LOCA Conditions

The behavior of Zirconium Alloy A and Zirconium Alloy F under LOCA transient conditions was evaluated previously in References 4 and 8, respectively. It was concluded that the behavior of Zirconium Alloy A and Zirconium Alloy F proposed to be included in the Palo Verde Unit 3 LFAs is expected to be equivalent or superior to that of conventional Zircaloy-4 under all conditions experienced during both normal operation and under the conditions existing during a LOCA transient. The explanation for expected performance of both cladding alloys in LOCA condition is unchanged from what was presented in the SARs, (Refs. 4 and 8). Therefore, the 10 CFR 50.44, 10 CFR 50.46, and 10 CFR 50, Appendix K criteria will be satisfied for these alloys.

2.5.2 Cladding Behavior Under Non-LOCA Conditions

Consideration was also given to the behavior of Zirconium Alloy A and Zirconium Alloy F under non-LOCA conditions and the results are discussed in previous reports (Refs. 4, 7 and 8). It was concluded that the cladding behavior under non-LOCA conditions was expected to remain equivalent or superior to that of Zircaloy-4 cladding. The explanation provided to demonstrate equivalency in performance of both cladding alloys to Zircaloy-4 that was presented in the SARs continues to remain valid. Therefore, no additional failures need to be considered other than that already analyzed for Zircaloy-4.

2.6 Evaluation Conclusions

The preceding discussions, including the materials contained in the referenced SARs (Refs. 4, 7 and 8), have shown that the predicted and measured performance of Zirconium Alloys A and F are equivalent or superior to that of Zircaloy-4 cladding under all anticipated operating conditions, including those considered in the safety analysis. In several key performance characteristics, such as waterside corrosion and fuel rod growth, Zirconium Alloys A and F are superior to Zircaloy-4. Zirconium Alloy A is equivalent to Zircaloy-4 in creep characteristics. Zirconium Alloy F shows superior creep properties and post-irradiation ductility. Furthermore, LFA fuel rods clad with both advanced alloys will be placed in non-limiting core locations which will experience no more than [ ] of the highest core power density through the irradiation periods.

Since the predicted and measured performance of Zirconium Alloys A and F are comparable or superior to that of Zircaloy 4, and the expected operating conditions are within those assumed for the standard clad rods currently licensed for Palo Verde Unit 3, it is concluded that the licensing basis currently in effect will not be compromised by incorporating a limited number of LFAs (up to three) containing test fuel rods clad with advanced alloys in non-limiting core locations.

3.0 REFERENCES

1. "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWd/kgU for Combustion Engineering 16x16 PWR Fuels", CEN-386-P-A, August 1992.

2. "Report on the Implementation of a 1-Pin Burnup Limit of 60 MWd/kgU at PVNGS", ABB Combustion Engineering Proprietary Report, CEN-427-(V)-P, November 1995.

3. [ ]

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5. Exemption from 10CFR50.46, 10CFR Part 50, Appendix K, and 10CFR50.44, NRC Letter, July 17, 1992, Docket No. 50-530.

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8. Safety Analysis Report for Use of Advanced Zirconium Based Cladding Material in PVNGS Unit 2 Batch J Demonstration Fuel Assemblies, CEN-426-P, Rev. -01-P, September 1995.

9. Temporary Exemption from 10CFR50.44, 10CFR50.46 and Appendix K to 10CFR50 for Lead Fuel Assemblies- Calvert Cliffs Nuclear Power Plant, Unit No. 1, NRC Letter, November 28, 1995, Docket No. 50-317.

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13. [

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21. [ ]

Table 1
Range Comparison of Major Alloying Elements in Zirconium Alloy A, F and Zircaloy-4

Alloy Designation
Zirconium Alloy A
Zirconium Alloy F
OPTIN™

Figure 1

PV-3 Batch F Fuel Rod Cladding Corrosion

■ OPTIN ○ Alloy A

Figure 2

Growth of Palo Verde 3 Batch F Fuel Rods with Alloy A Cladding
Compared to Those with Zircaloy-4 Cladding

■ OPTIN   ○ Alloy A   —— Fuel Rod Growth Model

Figure 3
Plenum Strain Comparison
2 Cycle PV-3F Fuel Rods

[ ]

-■- OPTIN -○- Alloy A

Figure 4

Irradiation Induced Growth vs Neutron Fluence
Irradiation at 330-350°C

Figure 5

Creep Strain of Zirconium Alloys Versus Fluence
Irradiation at 300-350°C, 6 = 130 MPa (18,850 PSI)

FIGURE 6

Comparison of Corrosion Resistance of Alloy A and OPTIN
360°C Deionized Water Autoclave, ASTM G2

WEIGHT GAIN (MG/DM^2)

AUTOCLAVE EXPOSURE IN DAYS

APPENDIX A

Criteria to be Used and an Evaluation to Determine Acceptability for a Fourth Cycle of Irradiation of Transplanted Rods in the Third LFA

A.1 Measurements and Criteria for Transfer

As discussed in the text of the report, up to [ ] test fuel rods (up to [ ] rods clad with Zirconium Alloy A and up to [ ] rods clad with OPTIN) are planned to be transferred into a third LFA for a fourth cycle of irradiation of these test rods during Cycle 7 of Palo Verde Unit 3. The projected burnups of the test fuel rods will exceed the current burnup limit for Palo Verde fuel during Cycle 7. The end-of-life (EOL) burnup for the remaining rods in the third LFA will remain below the 60 GWd/MTU burnup limit. When the conservatively estimated end-of-cycle (EOC) 7 burnup (up to [ ] GWd/MTU for OPTIN and [ ] GWd/MTU for Zirconium Alloy A rods) of the [ ] test rods are averaged with the other [ ] standard rods in the assembly, the assembly burnup is conservatively estimated to be [ ] GWd/MTU.

The rods to be transplanted will be examined at the end of the current operating cycle (Cycle 6) to determine their acceptability for continued irradiation during Cycle 7. The examinations will include visual inspection to verify fuel rod integrity, measurement of fuel rod length and eddy current testing to measure oxide thickness. The purpose of the physical examinations is to confirm compliance with predefined criteria.

The measured values of maximum oxide thickness and fuel rod growth for each rod will be compared with the rod specific maximum permissible values at the EOC 6. Maximum permissible values for each specific rod are calculated using [ ] techniques for oxide thickness and fuel rod length, taking into account the measured in-reactor behavior of the fuel rods clad with OPTIN and Zirconium Alloy A from previous cycles. These maximum values are based on two criteria: (1) that the maximum EOC 7 circumferentially averaged oxide thickness projected for each rod transferred will remain below the approved oxide thickness limit of [ ] microns (Refs. A-1 and A-2) at 99.5% (+3σ) level, and (2) that adequate shoulder gap will exist at the EOC 7 for each rod using conservative assumptions for fuel rod and fuel assembly growth.

The OPTIN and Zirconium Alloy A clad rods that have been tentatively selected for irradiation during Cycle 7 may achieve maximum terminal burnups of up to [ ] MWd/MTU, respectively. However, based on the measurements made at the EOC 6 outage, the number of rods to be transferred as well as the targeted terminal burnups at the EOC 7 may be reduced to comply with the two criteria specified above.

It is of interest to note that a transfer similar to that proposed here, using similar criteria for cladding oxidation and fuel rod shoulder gap, was previously performed during the EOC 4 outage

of Palo Verde Unit 1, for the purpose of accumulating higher burnup in Batch D fuel rods clad with OPTIN (Ref. A-3). Poolside measurements performed on the transferred rods, following their additional irradiation during Cycle 5, showed that with respect to both criteria, adequate margins remained on these rods (Ref. A-4). The above experience provides support for the adequacy of this approach for evaluating acceptability of fuel rod transfer to accumulate higher burnup.

A.2 Other Evaluations

Ductility of fuel rod cladding is a performance parameter of interest for fuel rod burnup beyond the current approved limit of 60 GWd/MTU. Two burnup related parameters that influence ductility of irradiated cladding are fast neutron fluence and hydrogen embrittlement. Based on the measured trend of cladding ductility with fluence, as summarized in Reference A-2, sufficient ductility is expected to remain through Cycle 7 for the estimated fast fluence level accumulated by the transplanted rods. Hydrogen embrittlement is associated with the waterside corrosion of fuel rod cladding. Discussions presented in References A-1 and A-2 confirm that the fuel rod cladding maintains adequate strain capability with oxide thickness buildup up to [ ] microns.

A related topic of interest to cladding ductility is high burnup fuel failure that may be caused by reactivity insertion accidents (RIA). Potential for high-burnup fuel failure in Palo Verde reactors has recently been evaluated (Ref. A-2), taking into account the RIA failure data provided by NRC information notice 94-64 (Ref. A-5). Using conservative enthalpy insertion failure limits together with Palo Verde specific data, it has been shown that the fraction of fuel rods in the core that may fail due to high burnup RIA is less than reported in PVNGS UFSAR. A maximum of approximately [ ] fuel failure was estimated by using a highly conservative failure threshold. This estimated value is less than 9.8% fuel failure reported for CEA ejection analysis in PVNGS UFSAR. Additionally, if [ ] Batch F rods, representing less than [ ] of the total fuel rods in the core, are conservatively assumed to fail, the additional fuel failures will be insignificant. Therefore, the postulated failure of these [ ] Batch F rods will not result in consequences more adverse than previously presented in PVNGS UFSAR.

No other evaluations are necessary as the discussions presented above show that the critical performance parameters for the transplanted rods will remain within the approved limits during their irradiation in Cycle 7.

REFERENCES

A-1 "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWd/kgU for Combustion Engineering 16x16 PWR Fuel", CEN-386-P-A, August 1992.

A-2 "Report on the Implementation of a 1-Pin Burnup Limit of 60 MWd/kgU at PVNGS", ABB Combustion Engineering Proprietary Report, CEN-427-(V)-P, November 1995.

A-3 "Palo Verde Nuclear Generating Station (PVNGS), Unit 1, Docket No. STN 50-528, Fuel Rod Transfer, File 93-056-026", Letter from W. F. Conway, Arizona Public Service Company to US NRC, Mail Station P1-37, November 19, 1993.

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