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RADIATION EFFECTS ON CORROSION OF
ZIRCONIUM ALLOYS

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RADIATION EFFECTS ON CORROSION OF ZIRCONIUM ALLOYS

From the wide use of zirconium alloys as components in nuclear reactors, has come clear evidence that reactor radiation is a major corrosion parameter. The evidence emerges from comparisons of zirconium alloy corrosion behavior in different reactor types, for example, BWRs versus PWRs and in corresponding reactor loop chemistries; also, oxidation rates differ with location along components such as fuel rods and reactor pressure tubes. In most respects, oxidation effects on power reactor components are paralleled by oxidation behavior on specimens exposed to radiation in reactor loops.

INFLUENCE OF RADIATION ON ZIRCONIUM ALLOY CORROSION PHENOMENA

Systematic evaluation of radiation effects on zirconium alloy corrosion have produced a wide range of results:

- small radiation effects on specimens and reactor and fuel assembly components in low-oxygen aqueous coolants (e.g., PWR, PHWR, VVER, some loops), in lower fluence regimes;
- strong evidence for accelerated radiation-induced oxidation on non-heat-transfer specimens and components in low-oxygen aqueous coolants at higher fluences;
- controversy about radiation effects on fuel cladding in low-oxygen coolants, due to the complexity of separating heat transfer and radiation effects;
- accelerated uniform oxidation on specimens and reactor and fuel assembly components in oxygenated aqueous coolants;
- accelerated nodular oxidation on fuel cladding and fuel assembly components exposed to oxygenated coolants; less prevalent nodular oxidation on specimens in loops with oxygenated coolants;
- localized oxidation from a variety of causes, including:
 - dissimilar metal effects;
 - alloy element segregations for some alloys;
 - spacer effects on LWR fuel rods;

This summary addresses phenomena and characteristics of uniform and localized oxidation of zirconium alloys under irradiation. Associated hydriding phenomena also are summarized.

EFFECTS OF ALLOY COMPOSITION

Table 1 indicates zirconium alloys that have been subject to reactor service and/or reactor loop evaluations. Three areas are addressed in Table 1: fuel cladding, pressure tubes, and loop studies. Other service applications include BWR fuel channels and fuel assembly spacer materials, involving the same alloys indicated for fuel cladding and pressure tube applications. Table 2 briefly compares the oxidation and hydriding characteristics.

Alloys differ in response to chemical and radiation environments. These effects will be addressed in subsequent sections.

UNIFORM OXIDATION OF ZIRCONIUM ALLOYS UNDER IRRADIATION

Uniform oxidation, as used here, generally denotes growth of oxides that do not vary sharply in thickness across relatively large areas. However, some localized phenomena will be cited that do not involve marked differences in oxide thickness. On the other hand, oxides regarded here as uniform often vary gradually in thickness from one location to another on a given reactor material, for example, along the length of a fuel rod or a reactor pressure tube.

Defining radiation effects on oxidation requires comparisons on identical materials in similar water chemistries, with and without radiation. Reactor loops with specimen locations over a range of flux levels and outside the flux zone provided the basis to quantify radiation effects and to detect even subtle trends. Radiation effects become evident on reactor components by comparisons of metallographic specimens that represent locations corresponding to a variety of flux levels. Non-destructive techniques are also sufficiently precise to permit oxide thickness measurements on reactor components and fuel rods.

Studies on zirconium alloy specimens and reactor components with negligible heat transfer provide unambiguous evaluation of radiation effects. Evaluations on heat transfer surfaces, principally fuel rods, have the complexity of separating radiation and thermal effects.

Table 1. Summary of Zirconium Alloys

Alloy(a)	Evaluated Under Irradiation		
	Fuel Cladding	Pressure Tubes	Loop Studies
1. Zircaloy-2	✓	✓	✓
2. Low Ni Zircaloy-2	--	--	✓
3. Zircaloy-4	✓	✓ ?	✓
4. Zr-1.2Cr-0.08Fe	(b)	--	✓
5. Zr-1.2Cu-0.28Fe	--	--	✓
6. Zr-3Nb-1Sn	(b)	--	✓
7. Ozhennite	✓	--	✓
8. Zr-2.5Nb	(b)	✓	✓
9. Crystal bar zirconium	--	--	✓
10. Zr-1Nb	✓	--	✓
11. Zr-3.5Sn-08.Mo-0.8Nb(c)	--	(b)	✓
12. Scandalloys		--	✓
13. Zr-1.0Nb-1.0Sn-0.1Fe(d)	(b)	--	--

a Need to add East European experience.

b Demonstration basis.

c Designated Excel.(1)

d Designated ZIRLO.(2)

Table 2. Alloy Characteristics Under Irradiation

<u>Alloy*</u>	<u>Oxidation</u>	<u>Hydriding</u>
1. Zircaloy-2	Intermediate rates	Intermediate rates
2. Low Ni Zircaloy-2	Similar to other Zircalloys	Like Zircaloy-4
3. Zircaloy-4	Like Zircaloy-2	Generally less than Zircaloy-2
4. Zr-1.2Cr-0.08Fe	Like Zircalloys but susceptible to alloy segregation	Similar to Zircaloy-2 except when alloy segregation evident
5. Zr-1.2Cu-0.28Fe	Much higher than Zircalloys	Much higher than Zircalloys
6. Zr-3Nb-1Sn	Mildly lower than Zircalloys	Mildly lower than Zircaloy-4
7. Ozhennite 0.5	*	*
8. Zr-2.5Nb	Lower than Zircalloys*	Much lower than Zircalloys*
9. Crystal bar zirconium	Higher than Zircalloys	Much higher than Zircalloys
10. Zr-1Nb	*	*
11. Zr-3.5Sn-0.8Mo-0.8Nb	--	--
12. Scandalloys	--	--
13. Zr-1.0Nb-1.0Sn-0.1Fe	Lower than Zircaloy	Lower than Zircaloy-4 due to lower oxidation

* Need to add East European experience.

OXIDATION IN IRRADIATED LOW-OXYGEN COOLANTS

Data from three sources are presented:

- Oxide thicknesses on Zircaloy-4 PWR fuel cladding, determined by metallography (Figure 1),
- Oxide thicknesses from pressure tubes, determined by metallography, including Zircaloy-2 (Figures 2 and 3) and Zr-2.5Nb (Figure 3),
- Oxide weight gains on Zircaloy-2 specimens exposed in a test reactor loop; specimens weighed (Figures 4 and 5).

Test loop data provide the highest precision in judgments of radiation effects on oxidation. The loop specimens are weighed to ± 0.1 mg and sometimes include identical irradiated and unirradiated specimens exposed to the same coolant.

Oxidation and Hydriding on Fuel Cladding

Aqueous corrosion of Zircaloy-4 PWR fuel cladding has been represented by some investigators as showing no radiation enhancement.⁽³⁾ The metallographic approach to define corrosion rates may have missed the early, relatively subtle, radiation effects detected by weight gains on loop specimens. In fact, comparisons of the wide band of corrosion results on fuel cladding with nominal data from unirradiated specimens (Figure 1) suggest some radiation enhancement. More recent evidence on PWR fuel cladding at higher exposures suggests a radiation effect, above the enhanced oxidation from thermal effects on heat transfer surfaces.^(4, 5)

Oxidation Hydriding and Hydriding on Reactor Pressure Tubes

Zircaloy-2 pressure tubes have been examined periodically over the 25-y active life of the Hanford N-Reactor (Figure 2); Zircaloy-2 and Zr-2.5Nb pressure tubes have been examined after extended exposures in CANDU reactors (Figure 3). The tube examinations provide a relatively comprehensive evaluation of radiation effects:

- as a function of time for a series of tube examinations,
- as a function of flux and temperature along the tube length,

- inside (water-side) versus outside (generally moist helium or nitrogen) surface oxidation,
- azimuthal variations.

Details regarding the above effects are reported elsewhere.^(6, 7) Here we will focus on the major oxidation and associated hydriding results. Figure 2 summarizes maximum oxidation and hydriding results on a series of N-Reactor pressure tubes. Tubes with lower exposures demonstrate relatively low oxidation rates, typical of low-oxygen reactor environments. The data in Figure 2 indicate a transition to a higher rate that is sustained over the subsequent exposures (2500 to 4000 d). The two oxidation trend lines represent least squares analysis of the data. The transition to the higher rate is consistent with thick-film phenomena first observed in test reactor investigations during the late 1960's and early 1970's⁽⁸⁾ (see section on Thick-Film Effects). The accelerated oxidation appears on Zircaloy pressure tubes from the Hanford N-Reactor, Pickering-1 and -2, and NPD reactors (Figures 2 and 3), all with low-oxygen coolants.

The associated hydriding trends largely parallel the oxidation trends for Zircaloy-2 tubes (Figures 2, 3, and 6). Tube 2755 (Figure 2) represents an exception to the trend; the tube had the expected oxide thickness, but a hydrogen content an order of magnitude below the trend line. Laboratory investigations confirmed that material from Tube 2755 is more resistant to hydriding than material from similar tubes with in-reactor hydriding histories.⁽⁹⁾ Lot-to-lot differences in both oxidation and hydriding were observed for Zircaloy-2 specimens in test reactor investigations.⁽⁸⁾

Figures 3 and 6 compare oxidation and hydriding trends for Zr-2.5Nb pressure tubes. At exposures shown in Figure 3, oxidation on Zr-2.5Nb tubes has not begun to demonstrate the transition that occurred consistently on Zircaloy-2 specimens. Hydriding trends for Zr-2.5Nb tubes are consistently below trends for Zircaloy-2 (Figure 6), agreeing with results from test reactor specimens.

Oxidation and Hydriding in Test Loops

Oxidation of loop specimens as a function of alloy composition is summarized in overview in Table 2, indicating marked differences in oxidation characteristics.

Figure 4 shows oxidation characteristics as a function of neutron fluence on identical Zircaloy specimens irradiated in both low-oxygen and oxygenated test reactor coolants. Compared to end-of-life radiation levels on fuel cladding and pressure tubes ($>5 \times 10^{21}$ n/cm², >1 MeV) the radiation exposures are relatively low, representing initial behavior. Oxidation on low-oxygen specimens exposed as-etched was only slightly accelerated, compared to unirradiated controls. Oxidation on prefilmed specimens was accelerated, compared to unirradiated controls. Beyond oxide thicknesses of 12 to 13 μm , Zircaloy-2 specimens exposed to a low-oxygen ETR G-7 loop coolant moved into a regime of highly accelerated oxidation, discussed under Thick-Film Effects.

Figure 5 indicates that Zr-2.5Nb also had low oxidation rates in the low-oxygen test reactor coolant. In fact, oxidation rates were slightly lower than rates on unirradiated controls, in contrast to slight acceleration on Zircaloy-2 specimens. Further examination of the Zr-2.5Nb behavior suggested that the lower oxidation for irradiated specimens was due to radiation-induced aging of the Zr-2.5Nb material, that tends to optimize oxidation resistance.⁽¹¹⁾

An overview of hydriding of loop specimens as a function of alloy is also summarized in Table 2. Hydriding trends generally parallel oxidation trends, but exceptions have been observed, involving high oxidation rates and low hydriding rates.

Thick-Film Effects in Low-Oxygen Coolants

The upturn in Zircaloy-2 oxidation rate shown in Figure 2 represents a trend previously discovered in test reactor exposures in the late 1960's and early 1970's.^(8,10) The test reactor series demonstrated that rapid oxidation rates did not decrease when Zircaloy-2 specimens were transferred from an oxygenated aqueous coolant to a low-oxygen coolant if the oxide film thickness was 30 μm or greater.⁽⁸⁾ If the film thickness was 10 μm or less,

the oxidation rate decreased after the oxygenated+low-oxygen transfer.⁽¹⁰⁾ To further clarify the threshold thickness that appeared to define the onset of thick-film behavior, Zircaloy-2 specimens were preoxidized in a laboratory autoclave. Identical Zircaloy-2 specimens were exposed in the ETR G-7 loop with the following surface conditions:

- as-etched
- thin oxides (13 to 24 mg/dm²)
- thick oxides (192 to 471 mg/dm²)

Specimens were irradiated to three fluences: 9×10^9 , 1.0×10^{21} and 1.5×10^{21} n/cm², >1 MeV in a low-oxygen (<0.05 ppm O₂) loop coolant.

The as-etched specimens showed only slightly accelerated oxidation, compared to unirradiated controls, typical of Zircaloy behavior when oxides are thin (<12 μ m). The specimens exposed with thin prefilms showed mildly accelerated oxidation. The specimens exposed with thick films oxidized at highly accelerated rates. In fact, the rates on the thick-film specimens in the low-oxygen coolant were similar to rates on identical specimens exposed in the oxygenated coolant. The test reactor series on thick-film specimens in low-oxygen water demonstrated that at oxide thicknesses above ~12 μ m, oxidation rates are highly accelerated. Oxidation characteristics of the thick-film specimens lead to the Thick-Film Hypothesis:

Beyond a threshold oxide thickness (>12 μ m for test reactor loop conditions), aqueous radiolysis in oxide cracks and pores controls Zircaloy-2 oxidation rates under irradiation.

The radiolysis creates microenvironments inside the oxide film. Comparison of oxidation rates on identical specimens suggested that the microenvironment inside the thick-films on low-oxygen specimens was similar to the highly oxidizing environment for specimens irradiated in the oxygenated coolant.

OXIDATION IN IRRADIATED OXYGENATED COOLANTS

The following zirconium alloy reactor and fuel assembly components are utilized in oxygenated reactor coolants:

Pressure tubes - SGHWR and RBMK reactor types
Fuel cladding - BWR, SGHWR, RBMK reactor types
Fuel channels - BWR, ?
Fuel Spacers - BWR, ?

Addressed here are effects of radiation on corrosion of fuel cladding, pressure tubes, and specimens exposed oxygenated coolants under irradiation.

Fuel Cladding

The BWR and SGHWR coolants, without hydrogen addition, typically have dissolved oxygen contents of a few tenths of a ppm O_2 . BWR and SGHWR fuel cladding is subject to accelerated oxidation, promoted by the synergistic effects of oxygen and radiation. The accelerated oxidation has two components: uniform and nodular. The nodular component is addressed under Localized Phenomena.

Figure 7 indicates oxidation trends on BWR fuel cladding, including uniform and nodular components. As with PWR cladding, the oxidation varies markedly along the length of a given rod, reflecting variations in radiation and water chemistry.

Hydrogen pickup fractions tend to be lower in the oxygenated reactor coolants, but total oxidation frequently is higher than at corresponding fluences in low-oxygen environments, so total hydriding frequently is similar for BWR and PWR cladding.

Pressure Tubes

Data to be added from foreign sources.

Test Reactor Specimens

Figure 4 compares the oxidation characteristics of Zircaloy-2 specimens in oxygenated and low-oxygen loop coolants under irradiation. The flux/oxygen interaction results in oxidation rates between one and two orders of magnitude above rates on corresponding unirradiated specimens. To illustrate for Zircaloy-2 Lot CT specimens in the G-7 loop, the in-flux oxidation rate at $280^\circ C$ and a neutron flux of $1 \times 10^{14} \text{ n/cm}^2, >1\text{MeV}$, was $3 \text{ mg/dm}^2\text{d}$.⁽⁸⁾ The corresponding rate on unirradiated control specimens was $0.03 \text{ mg/dm}^2\text{d}$, a factor of 100 below the rate under irradiation.

Corresponding Lot CT specimens (300°C prefilm) exposed in the low-oxygen G-7 loop coolant oxidized at 0.34 mg/dm²d at 1×10^{14} n/cm², an order of magnitude below the rate in the oxygenated environment.

The oxidation/fluence relationship is linear for Zircaloy-2 (Figure 4)⁽⁸⁾ in the oxygenated coolant. A similar relationship also applied to accelerated oxidation on thick-film specimens exposed in the low-oxygen G-7 loop coolant (Figure 3.3).⁽⁸⁾

The accelerated uniform oxidation on loop specimens parallels the accelerated uniform attack on fuel cladding. On the other hand, the incidence of nodular attack on the loop specimens seems to be lower than on susceptible Zircaloy materials irradiated in BWRs and in SGHWR.

The data in Figure 4 for oxygenated conditions demonstrate that specimens with thicker prefilms have higher resistance to the aggressive conditions than specimens initially exposed as-etched or with thinner/prefilms.

Figure 5 summarizes oxidation data for Zr-2.5Nb exposed in two metallurgical conditions. In the oxygenated coolant, Zr-2.5Nb specimens in the 700°C anneal, cold-worked condition oxidized at rates, similar to Zircaloy-2. In the quenched, cold worked, aged condition, the Zr-2.5Nb specimens were initially highly resistant to the oxygenated coolant. In fact, the initial oxidation rates were below rates on unirradiated control specimens (attributed to radiation-induced aging as indicated elsewhere). Mild upturns in oxidation rate occurred for as-etched specimens and for specimens with thin prefilms. Oxidation rates on specimens with thick prefilms remained low over the range of the exposures shown in Figure 5.

(Zr-1Nb experience in RBMKs?)

LOCALIZED OXIDATION AND HYDRIDING

Several localized oxidation and hydriding phenomena have been observed on irradiated zirconium alloys, including the following:

- nodular oxide formation,
- phenomena related to alloy element segregation,
- dissimilar metal effects.

Nodular Oxide Formation

Lenticular oxides have been a prevalent phenomenon on high heat transfer surfaces (fuel cladding) and low heat transfer surfaces (fuel spacers, fuel channels) of the Zircalloys. The largely solid oxide nodules nucleate, grow to mature sizes, in some cases coalescing to form thick, uniform oxides.⁽¹³⁾

A similar nodular phenomenon was observed on the Zircalloys in autoclaves at elevated temperatures (e.g., 500°C) in high-pressure steam.⁽¹⁴⁾ The nodular attack followed a sequence similar to that observed on BWR and SGHWR materials, but sometimes proceeded beyond coalescence to complete specimen oxidation and disintegration in periods as short as one day at 500°C. A beta heat treatment (e.g., 15 min at 1000°C), combined with rapid cooling, completely suppressed the nodular attack.⁽¹⁴⁾ Similar heat treatments are now used to suppress the nodular attack on reactor materials.⁽¹⁵⁾

Zr-2.5Nb and Ozhennite were resistant to nodular attack in 500°C autoclaving.⁽¹⁴⁾ Zr-2.5Nb seems to also have better resistance than the Zircalloys to nodular attack on reactor materials.

While nodular attack has been a major characteristic of reactors with oxygenated coolants (BWRs, RBMKs, and SGHWR), occasional appearance of nodules on PWR cladding has correlated with uncharacteristic periods of elevated oxygen concentrations in the normally low-oxygen reactor environments.⁽¹⁵⁾

Consequences of the nodular attack have included development of thick spalling oxide on BWR fuel channels⁽¹⁵⁾ and crud-induced localized corrosion (CILC) that has resulted in fuel rod failures in several BWRs.⁽¹⁶⁾

Alloy Element Segregation

Phenomena on two of the ETR G-7 loop alloys could be explained by radiation-induced alloy element segregation, involving evident evidence for the Zr-Cr-Fe alloy⁽⁸⁾ and subtle evidence for Zr-2.5Nb.

The Zr-Cr-Fe alloy has shown excellent resistance to nodular attack in autoclaves at 500°C. Under irradiation, local pustules and gray areas corresponded to relatively rapid accelerated attack. The localized effects seemed more prevalent on specimens autoclaved (e.g., 400°C, steam) prior to

reactor exposure. Electron microprobe scans indicated elevated chromium concentrations in areas of localized attack. Zr-2.5Nb specimens developed numerous gray halos and small nodules.⁽⁸⁾ Oxide partings in the oxides offered an explanation for the gray halos, but the oxide thickness variations between black and gray oxides were relatively small. Radiation-induced aging seemed to be most readily explained by condensation of niobium in supersaturated solution on radiation damage sites.⁽¹¹⁾

Dissimilar Metal Effects

Two dissimilar metal couples were investigated in the ETR G-7 loop test series, involving Zircaloy-2/Inconel-600 explosively bonded couples; and Zircaloy-2 and Zr-2.5Nb coupons with welded platinum inserts.

Regarding dissimilar metal effects on zirconium alloys in reactor service, enhanced oxidation on Zircaloy fuel rods at locations near fuel rod spacers seemed to correlate with proximity to stainless steel spacers.⁽¹⁷⁾ Electrochemical effects were involved as the leading explanation.

Other localized effects on irradiated loop specimens have been observed, including corner effects, local oxygen barriers, and crevice effects.⁽⁸⁾

REFERENCES

1. Cheadle, R. A., R. A. Holt, V. Fidleris, A. R. Causey, and V. F. Urbanic. 1982. "High-Strength, Creep-Resistant Excel Pressure Tubes." In Zirconium in the Nuclear Industry: Fifth Conference, ASTM STP 754, pp. 193-207, D. G. Franklin, Ed., American Society for Testing and Materials.
2. Sabol, G. P., G. R. Kilp, M. G. Balfour, and E. R. Roberts. 1989. "Development of a Cladding Alloy for High Burnup." In Zirconium in the Nuclear Industry: Eighth International Symposium, ASTM STP 1023, L. F. P. Van Swam and C. M. Eucken, Eds., American Society for Testing and Materials.
3. Dalgaard, S. B. 1976. "Long-Term Corrosion and Hydriding of Zircaloy-4 Fuel Clad in Commercial Pressurized Water Reactors With Forced Corrective Heat Transfer." Paper presented at the 149th Meeting of the Electrochemical Society, Spring 1976, Washington, D.C.
4. Stehle, H., F. Garzarolli, A. M. Garde, and P. G. Smerd. "Characterization of ZrO₂ Films Formed In-Reactor and Ex-Reactor to Study the Factors Contributing to the In-Reactor Waterside Corrosion of Zircaloy." In Zirconium in the Nuclear Industry: Sixth International Symposium, ASTM STP 824, pp. 483-506, J. G. Franklin and R. B. Adamson, Eds. American Society for Testing and Materials.
5. Kaiser, R. S., R. S. Miller, J. C. Moon, and N. A. Pisano. 1988. "Westinghouse High Burnup Experience at Farley 1 and Point Beach 2." In Proceedings of American Nuclear Society Topical Meeting on LWR Fuel Performance, April 17-20, 1988, Williamsburg, Virginia, pp. 119-124.
6. Lanning, D. D., A. B. Johnson, Jr., D. J. Trimble, and S. M. Boyd. 1989. "Corrosion and Hydriding of N Reactor Pressure Tubes." In Zirconium in the Nuclear Industry: Eighth International Symposium, ASTM STP 1023, L. F. P. Van Swam and C. M. Eucken, Eds., American Society for Testing and Materials.
7. Urbanic, V. F., B. Cox, and G. J. Field. 1987. "Long-Term Corrosion and Deuterium Uptake in CANDU-PHW Pressure Tubes." In Zirconium in the Nuclear Industry: Seventh International Symposium, ASTM STP 939, pp. 189-205, R. B. Adamson and L. F. P. Van Swam, Eds., American Society for Testing and Materials.
8. Johnson, Jr., A. B. April 1985. Zirconium Alloy Oxidation and Hydriding Under Irradiation: Review of Pacific Northwest Laboratory's Test Program Results. EPRI-NP-5132, Electric Power Research Institute, Palo Alto, California.

9. Johnson, Jr., A. B. 1989. "A Laboratory Method to Predict Hydriding Properties of Zirconium Alloys Under Irradiation." In Zirconium in the Nuclear Industry: Eighth International Symposium, ASTM STP 1023, L. F. P. Van Swam and C. M. Eucken, Eds., American Society for Testing and Materials.
10. Johnson, Jr., A. B., J. E. LeSurr, and R. A. Proebstle. 1974. "Study of Zirconium Alloy Corrosion Parameters in the Advanced Test Reactor." In Zirconium in Nuclear Applications, ASTM STP 551, pp. 495-513, American Society for Testing and Materials.
11. Urbanic, V. F., J. E. LeSurr, and A. B. Johnson, Jr. 1973. "Effect of Aging and Irradiation on the Corrosion of Zr-2.5 Wt% Nb." Presented at Corrosion/73, March 19-23, 1973, Anaheim, California. Published in Corrosion, 1973, 31:15-20.
12. Sheppard, M. F., and C. Tyzack. 1977. "Corrosion Monitoring of Steam Generating Heavy Water Reactor Pressure Tubes." In Zirconium in the Nuclear Industry, ASTM STP 633, pp. 258-280, A. L. Lowe, Jr. and G. W. Parry, Eds. American Society for Testing and Materials.
13. Megerth, F. H., C. P. Ruiz, and U. E. Wolff. 1971. "Zircaloy-Clad UO₂ Fuel Rod Evaluation Program Final Report, November 1967 to June 1971," USAEC Report GEAP 10371, General Electric Co., San Jose, California.
14. Johnson, Jr., A. B. 1969. "Corrosion and Failure Characteristics of Zirconium Alloys in High-Pressure Steam in the Temperature Range 400 to 500°C." Applications-Related Phenomena for Zirconium and Its Alloys, ASTM STP 458, pp. 271-285.
15. To be supplied.
16. To be supplied.
17. Sumerling, R., A. Garlick, A. Stuttard, J. M. Hartog, F. W. Trowse, and P. Sims. 1979. "Further Evidence of Zircaloy Corrosion in Fuel Elements Irradiated in a Steam Generating Heavy Water Reactor." Zirconium in the Nuclear Industry (Fourth Conference), ASTM STP 681, American Society for Testing and materials, pp. 107-121.

FIGURE 1. Maximum oxide layer thickness of PWR fuel rods versus burnup. (4)

FIGURE 2. Maximum oxidation and hydrogen content of N-Reactor pressure tubes. (6)

FIGURE 3. Long-term corrosion of cold-worked Zircaloy-2 and Zr-2.5Nb pressure tubes. (7)

FIGURE 4. Corrosion of Zircaloy-2, Lot HT, as a function of fast neutron fluence in oxygenated and deoxygenated AIR ID loop coolants. (10)

FIGURE 5. Corrosion of Zr-2.5Nb alloy as a function of fast neutron fluence in oxygenated and deoxygenated AIR 1D loop coolants. (10)

FIGURE 6. Long-term hydriding of cold-worked Zircaloy-2 and Zr-2.5Nb pressure tubes. (7)

FIGURE 7. Uniform and Local Oxide Growth of Zircaloy-2 Cladding in BWRs (Most data collected by AEG) ()

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