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FAST FLUX
TEST FACILITY
PROJECT REPORT

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FFTF EXAMINATION FACILITY ATMOSPHERE
IMPURITY CONTROL REQUIREMENTS
(A REVIEW OF EXISTING INFORMATION)

January 1969

AEC RESEARCH &
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FFTF EXAMINATION FACILITY ATMOSPHERE
IMPURITY CONTROL REQUIREMENTS
(A REVIEW OF EXISTING INFORMATION)

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January 1969

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D. P. O'Keefe

ABSTRACT

Impurity control requirements for the protective atmosphere to be used in the FFTF Examination Cell were studied by means of literature search and consultation.

The review yielded information that permits a partial definition of the atmosphere impurity control requirements for stainless steel components and indicated areas in which research and development effort is required.

An argon atmosphere with limits for O_2 and H_2O of 25 ppm each, and N_2 in the 3 to 4% range as controlled by feed and bleed is sufficient protection for stainless steel components maintained at temperatures of 600 °F and below. If metal temperatures are permitted to reach 1000 °F, the possibility for minor nitriding exists.

Caustic stress corrosion is a possible source of trouble for specimens at temperatures above 1000 °F, sodium-covered, and under stress, which could be the case for uncleaned, high exposure fuel pins.

TABLE OF CONTENTS

ABSTRACT	iii
INTRODUCTION	1
SUMMARY AND CONCLUSIONS	3
THE PROBLEM ELEMENTS	5
Stainless Steel Behavior.	5
Temperature	6
Stress	7
Irradiation Effects	8
FIELD EXPERIENCE	9
DISCUSSION	17
REFERENCES	19
DISTRIBUTION	23

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D. P. O'Keefe

INTRODUCTION

The Inert Gas Cell Examination Facility will provide for the nondestructive examination of fuel assemblies and other core components irradiated in the Fast Test Reactor (FTR).⁽¹⁾ Visual examination of the fuel is required and for this, the sodium will have to be removed. The transfer from sodium to gas, sodium removal, and examination are to be accomplished without corroding or otherwise altering the irradiated hardware in a manner that would influence examination results or jeopardize the fuel performance if it is reinserted in the reactor.

The material of primary concern is Type 316 stainless steel, which is the cladding for the driver fuel. This alloy has seen service in sodium-cooled systems. Some data on its performance upon removal from the sodium are available. Considerably more use has been made of Type 304 stainless steel in sodium systems, and this information also applies because there is little, if any, difference in the corrosion performance of the two alloys.⁽²⁾ In general, the performance of Types 304 and 316 stainless steel fuel sheaths in sodium and NaK coolants has been good. Corrosion associated with the postirradiation process of sodium removal and exposure to air and other atmosphere has been of concern, but has not presented significant problems. In providing for interim examination of FTR fuel, however, a close look at corrosion possibilities during examination is warranted because high fission product decay heating creates the potential for more severe corrosion than has been the case with other liquid-metal-cooled reactors.

A review of existing information on the behavior of stainless steels under conditions that are relevant to the subject problem has been made. Reference is made to the fundamentals of stainless steel behavior, temperature, stress, media composition and other such parameters which influence the surface chemistry of the fuel cladding, as well as related field experiences.

SUMMARY AND CONCLUSIONS

By means of literature search as well as consultation, a study was conducted on impurity control requirements for the protective atmosphere to be used in the FFTF Examination Cell. The situation regarding stainless steel components was examined in detail. The review yielded information permitting a partial definition of the atmosphere impurity control requirements for stainless steel components and indicated areas in which research and development effort is required.

An argon atmosphere with limits for O_2 and H_2O of 25 ppm each, and N_2 in the 3-4% range as controlled by feed and bleed is sufficient protection for stainless steel components maintained at temperatures of 600 °F and below. If metal temperatures are permitted to reach 1000 °F, the possibility for minor nitriding exists; however, experimental determination of the degree of nitriding is needed before a change in the allowable N_2 content is warranted.

Caustic stress corrosion is a possible source of trouble for specimens at temperatures above 1000 °F, sodium-covered, and under stress, which could be the case for uncleaned, high exposure fuel pins.

Follow-on work is needed to examine possible effects of the examination cell atmosphere impurities on candidate cladding alloys other than austenitic stainless steels for advanced fuels.

THE PROBLEM ELEMENTS

STAINLESS STEEL BEHAVIOR

Some of the characteristics of stainless steel as they are described in the literature and which appear to be of importance to the subject problem are included below.

When an austenitic stainless steel is held for sufficient time in the temperature range of 800 to 1650 °F, a complex iron-chromium carbide precipitates along grain boundaries. In this condition, stainless steel is sometimes subject to an intergranular corrosion attack. It has been shown that this type of attack is related to galvanic cell action between constituents in the stainless steel. Stickler and Vinckier⁽³⁾ experimentally demonstrated that the degree of intergranular corrosion resistance is dependent on the morphology of the grain boundary carbide. They state that an electrochemical cell develops between the carbides and the stainless steel, and that intergranular corrosion can be explained without postulating grain boundary chromium depletion. Recent work by France and Green goes a step farther and defines the precise environmental conditions necessary for intergranular corrosion by measurement of the electrode potential of sensitized stainless steel in the corrosion media.⁽⁴⁾ This work demonstrates that the intergranular corrosion of sensitized stainless is an electrochemical process, and the fact that stainless steel is sensitized does not necessarily eliminate its corrosion resistance, particularly in a gaseous medium. Nearly all standard references consider the intergranular corrosion of sensitized stainless steel to be a process which occurs in fairly aggressive media. Its occurrence in mild media such as weak acids is characteristically associated only with very long exposures.⁽⁵⁾

Intergranular corrosion may also be caused by the presence of sigma phase, but it occurs only in a concentrated nitric acid solution. For Type 316, it takes boiling 65% nitric acid.⁽⁵⁾ Sigma phase, which can conceivably show up in the fuel cladding,⁽²⁾ is of no consequence to the corrosion performance of the Type 316 in the media of concern to this study.

Stress corrosion cracking is a recognized problem with the austenitic stainless steels. This has resulted in the failure of nonsensitized Type 304 stainless steel clad fuel rods in boiling water.⁽⁶⁾ Stress corrosion in sodium hydroxide solutions is a well documented phenomenon⁽⁵⁾ and conceivably applies in the FTR fuel handling. Studies of the caustic stress corrosion of Type 347 stainless steel⁽⁷⁾ have shown that pure anhydrous NaOH will cause cracking at 850 °F, but may not do so at lower temperatures. Concentrated aqueous solutions of NaOH will produce stress corrosion of Type 347 in short times at 650 °F, and in this medium the threshold stress is below two-thirds of the yield strength. Mixtures of 16 to 60% NaOH in sodium will also cause cracking.

TEMPERATURE

The most important parameter in the problem of fuel cladding performance is temperature. There is sufficient experience with the handling of stainless steel clad fuels at low temperatures to establish the ability of the material to remain inert (with some exceptions to be covered later) in nonaggressive media such as air or water, even though it is sensitized. The temperatures of concern in the examination cell result from absorption of the energy liberated as a result of the radioactive decay of fission products in the fuel. The FTR fuel subassemblies and the experimental subassemblies will have higher decay heat levels than have been associated with other fast reactor fuels. A preliminary estimation by

Bunch⁽⁸⁾ placed the decay heat generation level for a driver subassembly at 10 kW after five days of decay time. After 30 days, the subassembly is rated at 5 kW. Additional decay beyond this time is very slow. By comparison, the maximum decay heat encountered in handling the EBR-II fuel subassembly is 1 to 1.5 kW⁽⁹⁾ and at Dounreay, 1 kW.⁽¹⁰⁾ At these two sites they apparently have not experienced any damaging heating of the fuel during examination. However, there is one allusion to the difficulty of gas cooling an uncleaned 310 W subassembly at Dounreay.⁽¹¹⁾

In the examination cell, it is intended to provide cooling for the irradiated FTR fuel as necessary to control cladding temperatures below the operating temperature of the cladding.⁽¹²⁾ The target control temperature for the driver fuel cladding is 600 °F during examination.⁽¹³⁾ Somewhat higher temperatures can be expected for experimental fuels which will operate at cladding temperatures over 1000 °F. Preliminary data from experiments now underway indicate that the feasibility exists for meeting the temperature control requirements with axial and transverse gas flow through the fuel pin assembly.⁽¹⁴⁾ In practice, one may expect uncertainties to arise from nonstandard impediments to flow, such as warped pins and adhering sodium. No real information exists concerning the effect of such things on the temperature distribution, but it appears reasonable to expect peak temperatures to exceed 600 °F under some credible conditions, and that corrosion mechanisms at temperatures between 600 and 1200 °F should be examined.

STRESS

The FTR fuel cladding will be subject to pressure from fission gases, fuel moisture, sorbed gases and helium backfill. At goal exposure, this will be a maximum and will result in an

8500 psi loop stress at operating temperature. At the goal exposure of 50 to 70,000 MWd/ton, the cladding will also have been subject to a 0.2% strain from fuel swelling.⁽¹⁵⁾

IRRADIATION EFFECTS

Documented commentary are not in complete accord concerning the effect of irradiation on the corrosion resistance of stainless steel. The following quotes are from periodic reports on a stainless cladding program at General Electric:

- "The VBWR and Dresden test history with stainless steel cladding failures presents some evidence that changes in the material occur during irradiation which make the stainless steel increasingly susceptible to intergranular cracking. Although specific fuel rod claddings have been subjected to severe conditions from the beginning of irradiation testing, ... intergranular failures did not occur in any of the specimens until a moderate amount of exposure had been attained. This suggests that irradiation-induced alloy phase changes or changes in the physical and chemical properties of the grain boundaries may be occurring."⁽⁶⁾
- "Results of 24 hours corrosion tests on irradiated and unirradiated Type-304 stainless steel in boiling $\text{HNO}_3 + \text{Cr}^{+6}$ indicate that irradiation exposure increases susceptibility to corrosion. The weight loss of irradiated cladding samples was consistently higher than that of unirradiated control specimens by a factor of two."⁽¹⁶⁾

Perhaps more significant to the Examination Cell issue because of the mildly corrosive medium is the study conducted at Harwell. The effect of irradiation on the corrosion of 20 Cr-25 Ni stainless steel in CO_2 was examined at Harwell AERE. They found⁽¹⁷⁾ "... at all temperatures the irradiation had not affected the magnitude of attack or the percentage of nonadherent oxide to within a factor of 2, the limit of reproducibility."

FIELD EXPERIENCE

Field experiences which may be drawn on for predicting problem areas in the FTR examination case exist principally in reactor applications because this is about the only area where users have had concern for the corrosive performance of sensitized stainless in mildly corrosive media such as water or air.

Experience at the Hallam (HNFP) project included a case of severe corrosion on a Type 304 stainless steel clad mockup element in wash cell tests.⁽¹⁸⁾ In this case, the stainless steel cladding material had been sensitized during the fuel element fabrication, and after 40 cycles of alternate exposure to liquid sodium and steam, corrosion penetrated the cladding. It was concluded in this case that all of the elements necessary for the formation of an electrolytic cell existed at the surface of the Type 304 stainless steel and that the intergranular corrosion proceeded by a galvanic process. Willard and Sheffield,⁽¹⁸⁾ who investigated the occurrence, define the necessary conditions as:

1. A galvanic couple, austenitic steel (with nickel, chromium and carbon) versus chromium or chromium-iron carbides, with a potential difference of approximately 0.5 V.
2. An electrolyte (in this case, condensed steam of unknown purity from the wash process or the naturally occurring atmospheric condensate in the coastal community locale of the laboratory).
3. A depolarizing element such as dissolved oxygen in the electrolyte.
4. A halogen ion such as Cl^- .

In another HNPF case involving Large Component Test Loop environmental tests of moderator elements, corrosion-induced cracks developed after sodium exposure followed by air exposure. Here 0.016 in. thick Type 304 stainless steel cans were

heated isothermally and subsequently thermally shocked to produce thermal stresses in the cans.⁽¹⁹⁾ The sodium was then replaced with a nitrogen environment at temperatures between 750 °F and ambient, over a period of five days.⁽²⁰⁾ As a result of this exposure, cracks developed in the moderator sheaths. It was concluded that sensitization of the stainless steel, reduction of the Cr_2O_3 (free energy considerations indicate that at temperatures up to 1000 °F, Na_2O will be formed in preference to Cr_2O_3) and metal-nitrogen interaction left the surfaces of the moderator cans susceptible to corrosion. When the moderator cans were exposed to the corrosive anions of the laboratory atmosphere, corrosion and subsequent cracking took place. Stress acted as an accelerator of the corrosion process.⁽¹⁹⁾ In support of this investigation, tests conducted with small Type 304 stainless tabs exposed alternately to sodium and nitrogen provided the following information:⁽²⁰⁾

1. Intergranular corrosion did not occur in sodium.
2. Specimens exposed 11 1/2 hr at 900 °F did not sensitize.
3. Specimens exposed to 1100 °F sodium followed by 1100 °F nitrogen were nitrided, but did not corrode in 30 days in air.
4. Specimens exposed at 900, 1000, 1100 °F in sodium, steam cleaned, water rinsed, exposed in air 30 to 45 days did not corrode.
5. In these supporting tests, cracking similar to that of the moderator cans in the LCTL were not duplicated until the tabs were exposed to acidified copper sulphate.

At the Sodium Reactor Experiment (SRE), the documented postirradiation experience most nearly germane to the FTR examination case was the U-10 Mo fuel examination. The Type 304 stainless steel cladding on this element was sensitized prior to irradiation. Irradiation was in sodium at

870 °F outlet temperature and continued for three years. Three interim examinations were made in which the element was washed and examined in air. No corrosion was found in postirradiation metallographic examination of the 0.010 in. thick cladding.⁽²¹⁾

In out-of-reactor experimentation at the SRE, Gill and Bokros⁽²²⁾ established that nitriding of Type 304 stainless steel occurs in a sodium-nitrogen system at 1000 °F. The process is more rapid on the sodium side of the interface.

Experience at the EBR-II in the cleaning and inspection of irradiated fuel assemblies and other hardware most nearly approaches the conditions which will be encountered with FTR fuel. The driver fuel cladding is Type 304L stainless steel which operates in sodium at temperatures between 850 and 1050 °F. Sodium removal is accomplished in three steps:⁽⁹⁾

1. Moist air or argon purge.
2. Demineralized water.
3. Dry air.

This treatment has not resulted in any corrosion detectable by metallographic or other means in any of the Type 304L hardware removed from the reactor. The following types of items have been examined in detail after reactor exposure:

- The hexagonal tubing which contains the fuel bundle.
- Driver fuel cladding.
- Various experimental capsule cladding.
- Blanket element cladding. This operated at a lower temperature than fuel cladding but had a much longer residence in the reactor.
- Guide Thimble, a very high exposure piece of Type 304L.

All of the above were subjected to the normal cleaning procedure and the air cell atmosphere. No corrosion effects have been seen, and although no driver fuel pins have been reinserted in the reactor as yet, their condition in regard to effects from exposure of the cladding to air is acceptable for reinsertion.

A test is in progress to assess the effects of the standard washing and drying operations used for driver subassemblies in combination with reexposure to sodium. A driver subassembly which has received 1.2% burnup⁽²³⁾ is undergoing five cycles of exposure to primary sodium followed by washing and drying, after which it will be examined in detail. This test will provide good information on cladding corrosion resistance upon reinsertion.

At the Dounreay Experimental Reactor Establishment, the mixed oxide fuel pins are clad in Type M316L steel which is the same as AISI 316 L stainless steel. They operate in NaK at temperatures which will sensitize the stainless steel. Fuels have been reinserted in the reactor after interim examination and subsequently examined after completion of the irradiation. In one such case,⁽²⁴⁾ no evidence of corrosion of the stainless steel was seen in visual inspection during the final examination. This fuel had a burnup of 41,000 MWd/tonne at a maximum can temperature of 1059 °F. This element was allowed to cool in NaK after discharge for 88 days, and maximum measured temperatures within the assembly without forced gas cooling were in the vicinity of 700 °F. (The cooling gas flow was accidentally stopped for a period of 80 hr before cleaning.) The cell atmospheres were argon (120 ppm O₂ and 45 ppm H₂O) before cleaning and air after cleaning. Cleaning was accomplished with dry steam, wet steam and a water rinse.

In another examination of fuel after irradiating to 6.1% burnup, no cladding corrosion was observed. This fuel was handled in argon before cleaning and air after cleaning. Total decay heating (3 months after discharge) of the 77 pin assembly was predicted to be 130 W. This appeared to be confirmed by the wrapper surface temperature of 115 °C.⁽²⁵⁾

In documented descriptions^(26,27) of two other similarly conducted examinations of subassemblies irradiated to 5 and 7% burnup, no mention is made of corrosive effects. Presumably, there were none to be seen.

The policy regarding atmosphere control in the examination caves at Dounreay appears to be based on the reactivities of NaK and plutonium metal. No concern is implied about reaction of sensitized Type 316 stainless steel with cell atmospheres. The cells may be operated with argon, nitrogen or air atmospheres^(28,29) and have provision for the admission of an argon blanket as a fire extinguisher, if necessary, for NaK fires.

Some of the findings by F. A. Smith at Argonne National Laboratory (ANL)^(30,31) must be considered in view of Examination Cell Requirements. His studies of sodium - stainless steel interaction based on Strauss test treatment in conjunction with electrical resistivity measurements indicate "... the general desirability of not re-exposing Type 304L stainless steel to sodium after any storage in air."⁽³¹⁾ This is based on experimental results which indicate that exposure to sodium in the 1100 to 1200 °F range of temperatures lowers the corrosion resistance more than sensitization in vacuum. He postulates that the effect results from the penetration of sodium into grain boundaries where it is retained through normal cleaning procedures. Smith's test data indicate the penetration is enhanced by intermittent exposure to air.

The effect of intermittent sodium - air exposure has also been the subject of attention at General Electric, San Jose, where examination of piping after long term sodium exposure revealed localized attack up to 7 mils in Type 316 stainless steel after 25,000 hr of operation.⁽³²⁾ Metallographic and fractographic evidence suggested that the penetration resulted from an electrochemical reaction similar to caustic corrosion. A look at operating conditions encouraged this contention. As described by the authors,

"Examination of the loop operating procedures indicated that during each shut-down period the internal piping surfaces, coated with residual sodium film, were subjected to atmospheric contamination. Start-up conditions were such that substantial thermal stresses were introduced in the piping during the preheat cycle to reinforce the already present conditions that suggest caustic corrosion. It is tentatively concluded that the observed intergranular attack is associated with a film of Na_2O and NaOH , and occurs in a manner similar to caustic corrosion."⁽³²⁾

At Battelle-Northwest, recent experience with Type 304 stainless steel burst test specimens^(14,33) has demonstrated how caustic stress corrosion can become a very real influence when the conditions are right. In this case, residual sodium on the inner surface of the specimen was exposed to water in the cleaning process and probably converted to sodium hydroxide. The specimens were stressed by pressurizing with argon at 1400 °F. Pronounced corrosion resulted and the stress-rupture strength of the specimens was lower than for specimens not exposed to sodium.

At Fermi, components removed from the reactor are cleaned with high pressure argon, steam and water. The policy developed there concerning reinsertion is described by Bowersock and Studer in their trip report,

"APDA does not normally return steam cleaned sub-assemblies to sodium service. They have found

that the oxides and caustic formed during steam cleaning are not readily removed and accelerated corrosion has resulted in storage of components removed from liquid sodium service. These coupons are held in clean liquid sodium (750 °F) to remove any residual oxides prior to being returned to service."(34)

At Los Alamos, a facility is being prepared for the examination of irradiated LMFBR fuel materials as requested by the Fuels and Materials Branch of DRD&T. The atmosphere control in this facility is intended to avoid reaction with the sodium bond and fuel materials which will be exposed in the process of destructive examination of the fuels.⁽³⁵⁾ This is accomplished by containing the examination equipment within a 200 ft³ capacity box to be inserted in the concrete cell. The box can maintain, under static conditions, an argon purity of 4 ppm O₂ + H₂O.⁽³⁶⁾

At General Electric, San Jose, in a program to develop cleaning and interim inspection methods, evidence of sodium penetration into stainless steel components has been found.^(37,38) The evidence of this was, in part, the appearance of white, strongly basic, material on the surface of cleaned specimens exposed to air. They conclude that since sodium penetration, "... is an accomplished fact, there is a possibility that the mechanical performance of the metal will be compromised by the subsequent emergence of this sodium and its reaction with oxygen and moisture."⁽³⁷⁾

At the Applied Research Laboratory of U.S. Steel Corporation, a detailed examination was conducted on Type 304 stainless steel pipe which had been internally exposed to 572 to 1472 °F sodium for 7800 hr. They concluded that no diffusion of sodium into the stainless steel occurred. A lamellar phase 1 to 2 mils deep on the inner surface of the specimen was

found to be predominantly Cr_2N , and was not the result of sodium penetration. The source of nitrogen was taken to be nitrogen-containing impurities in the sodium.⁽³⁹⁾

In a literature survey on cover gas impurities in sodium systems, Maffei has reviewed an area of published data of some importance to the Examination Cell issue.⁽⁴⁰⁾ He found that there are no data which correlate any reactor damage to cover gas impurities. There was more than one reference to the occurrence of nitriding, and Maffei suggests that partial pressures of nitrogen in FFTF primary cover gases could lead to nitriding of stainless steel components.

DISCUSSION

The state-of-the-art regarding the handling of reactor components after sodium exposure is mainly limited to stainless steel material and to temperatures below 700 °F. From this information, it may be concluded that stainless steel components maintained at temperatures below 600 °F will be adequately protected by the FFTF fuel examination atmosphere discussed in Reference 1, namely argon with nominal impurity contents of 25 ppm H₂O, 25 ppm O₂ and 3 to 4% N₂. Even with handling in nominally dry air, no corrosion effects have been observed on cladding where temperature has remained below 600 °F.

The instances in which corrosion of stainless steel has occurred at temperatures below 600 °F after removal from sodium service, all have to do with exposure to comparatively aggressive media such as moist air, air plus water plus halogen ions or alternating steam and sodium exposure cycles. These experiences are of tangential interest in establishing a background of what can happen to sensitized stainless steel, but the environments involved are too far removed from the Examination Cell case to be of concern. In its performance in a dry atmosphere, the fact that the stainless steel is in a highly sensitized condition is probably not of particular importance.

If surface temperatures are permitted to reach 1000 °F, the possibility for a minor amount of nitriding appears to exist, but further experimental or other studies are needed to establish if there is reason to maintain the N₂ fraction below the level attainable by the feed and bleed method.

In this literature there was survey reference to only one highly damaging corrosion process that might be a possibility in argon containing only minor amounts of reactive

constituents. This was the caustic stress corrosion phenomenon, which is an area for more detailed consideration in the event sodium is to be left on specimens and the temperature is permitted to exceed 1000 °F. The hazard should be the greatest for unvented, high exposure fuel pins with cladding stressed from internal pressure. The atmosphere impurity control under these conditions would have to restrict water and oxygen reaction with the sodium — probably a less practical goal than complete sodium removal or controlling the temperature below 1000 °F.

A primary area for additional study that could not be covered in this survey concerns the protective atmosphere requirements for advanced fuel employing cladding alloys other than the austenitic stainless steels. Some of these, such as the vanadium-base alloys with a high affinity for nitrogen, may require a lower nitrogen fraction in the cell gas than can be maintained by the feed and bleed system. This will likely have to be determined by experimentation.

Activities in the Examination Facility interface with the procedures for transferring fuel into the cell. The handling system for accomplishing the transfer will have its own atmosphere controls based on its own unique impurity control requirements. This study was directed toward only the requirements of the Examination Facility, not the fuel handling and reactor refueling systems.

REFERENCES

1. E. B. Ramsey and D. J. Meyers. Conceptual System Design Description for the Inert Gas Cell Examination Facility No. 71, A-0024-R. Pacific Northwest Laboratory, March 18, 1968.
2. R. R. Studer. On the Selection of Constructional Materials and Fabrication Processes for FFTF Closed Loops, BNWL-1805. Pacific Northwest Laboratory, July 31, 1968.
3. R. Stickler and A. Vinckier. "Morphology of Granic-Boundary Carbides and Its Influence on Intergranular Corrosion of 304 Stainless Steel," Trans., Am. Soc. Metals, vol 54. 1969.
4. W. D. France and N. D. Greene. "Predicting the Intergranular Corrosion of Austenitic Stainless Steels," Corrosion Sci., vol. 8. 1968.
5. F. L. LaQue and H. R. Copson. Corrosion Resistance of Metals and Alloys, Reinhold Publishing Corp., N.Y., 1963. p. 383.
6. T. J. Pashos. Stainless Steel Failure Investigation Program, GEAP 4915, General Electric Company, Schenectady, N.Y. July 1965.
7. G. C. Wheeler and E. Howells. Caustic Stress Corrosion of Type 347 Stainless Steel, KAPL-M-6911, Knolls Atomic Power Laboratory, March 1968.
8. W. L. Bunch. Decay Heat Deposition from Irradiated FTR Fuel Bundle, BNWL-648. Pacific Northwest Laboratory, December 1967.
9. V. G. Eschen. Unpublished Data. Argonne National Laboratory, June 12, 1968. (Personal Communication)
10. K. M. Swanson. Non-Destructive Testing of Fuel in the Dounreay Alpha-Gamma Hot Cells (Paker No. 5), International Symposium on Working Methods in High Activity Hot Laboratories, vol. 1, pp. 155-165. Organization for Economic Cooperation and Development, Paris, 1965.
11. K. M. Swanson et al. The Examination of DFR 53 Sub-assembly at 5% Burnup. TRG-Memo-3032(D). Dounreay Experimental Reactor Establishment.

12. R. J. Hening. Functional Testing Requirements for the FFTF, A-0030-R. Pacific Northwest Laboratory, November 1967.
13. D. R. Dickinson. Gas Cooling of FFTF Fuel During Examination and Disassembly Preliminary Feasibility Study, BNWL-524. Pacific Northwest Laboratory, July 1967.
14. E. R. Astley. Fast Flux Test Facility Monthly Informal Technical Progress Report, BNWL-830, p. 4.41. Pacific Northwest Laboratory, June 1968.
15. FFTF, Conceptual Component Design Description for the First Core Fuel Assembly, A-0035-R, Pacific Northwest Laboratory, January 1968.
16. T. J. Pashos. Second Quarterly Progress Report, Stainless Steel Failure Investigation Program, GEAP-4968, General Electric Company, Schenectady, N.Y. February-June, 1965.
17. M. J. Bennett et al. "The Effect of Fission Neutron Irradiation upon the Oxidation of a 20 wt% Cr, 25 wt% Ni/NG Stabilized Stainless Steel by Carbon Dioxide Between 580 and 840 °C," J. Nucl. Mater., vol. 25, p. 260-270. 1968.
18. R. M. Willard and G. S. Sheffield. Corrosion of the HNPF Wash Cell Mock-Up Element Cladding, NAA-SR-TDR-7571. Atomics International, June 30, 1962.
19. E. C. Phillips. LCTL Test of HNPF Moderator Cans, NAA-SR-TDR-5442. Atomics International, June 29, 1960.
20. L. Newcomb. Final Report on HNPF Moderator Element Development, NAA-SR-6508. Atomics International, June 15, 1964.
21. J. L. Arnold, K. J. Miller, and R. M. Peterson. U-10 wt% Mo Fuel Element Irradiation in SRE, NAA-SR-11121. Atomics International, August 31, 1965.
22. J. J. Gill and J. C. Bokros. Nitriding of Type 304 Stainless Steel in a Sodium-Nitrogen System, NAA-SR-6162. Atomics International, May 30, 1961.
23. Reactor Development Program Progress Report, ANL-7382. Argonne National Laboratory, September 1967.

24. K. M. Swanson, J. K. Butler, and G. Sutherland. The Examination of D.F.R. 53 Sub-Assembly at 5% Burnup, TRG-Memo-3032(D)Revised. United Kingdom Atomic Energy Authority, Dounreay, Caithness, Scotland, November 1966.
25. J. K. Butler. Post Irradiation Examination of the MkIB 77-Pin Sub-Assembly Experiment, TRG-Memo-3728(D). United Kingdom Atomic Energy Authority, Dounreay, Caithness, Scotland.
26. K. M. Swanson et al. The Examination of D.F.R. 166 at 5% Maximum Burnup, TRG-Memo-4097. United Kingdom Atomic Energy Authority, Dounreay, Caithness, Scotland.
27. K. M. Swanson et al. Mark II Sub-Assembly Report on the Examination of D.F.R. 114 at 7.3% Maximum Burnup, TRG-Memo-4073. United Kingdom Atomic Energy Authority, Dounreay, Caithness, Scotland.
28. K. M. Swanson. "Reactor Laboratory's Success Means Simpler Pu Fuel Inspection," Spectrum Magazine. 1965.
29. R. C. Alexander and S. A. Cottrell. "DERE Highly Active Pu Examination and Reprocessing Caves," J. Brit. Nucl. Energy Soc., vol 3., p. 215. July 1964.
30. ANL Reactor Development Program Progress Report, ANL 7391. Argonne National Laboratory, October 1967.
31. ANL Reactor Development Program Progress Report, ANL 7399. Argonne National Laboratory, November 1967.
32. F. A. Comprelli, F. J. Hetzler, and R. S. Young. "Observation Concerning Localized Corrosion in Pumped Sodium Systems," Trans. Am. Nucl. Soc., vol. 8, p. 411. November 1965.
33. FFTF Monthly Informal Progress Report for July, 1968, BNWL-863. Pacific Northwest Laboratory, August 9, 1968.
34. R. R. Studer and R. V. Bowersock. Unpublished Data. Pacific Northwest Laboratory. (Trip Report, Visits to APDA, MSA, ANL; May 29, 1968.
35. LASL Quarterly Status Report on the Advanced Plutonium Fuels Program July-September 1967, LA-3820-MS. Los Alamos Scientific Laboratory, November 1967.
36. J. W. Schulte. Unpublished Data. June 1968. (Private Communication).

37. Sodium Component Development Program Mass Transfer Investigation in Liquid Metal Systems. 1st Quarterly Report, March-May, 1967. GEAP-5508. General Electric APED, San Jose, California.
38. Sodium Component Development Program Mass Transfer Investigation in Liquid Metal Systems. 2nd Quarterly Report, June-August, 1967. GEAP-5546. General Electric APED, San Jose, California.
39. G. A. Ratz and K. G. Brickman. "Effect of Liquid Sodium on Welded AISI Type 304 Stainless Steel Pipe, Nuclear Applications, vol. 4. March 1968.
40. H. P. Maffei. Cover Gas Impurities in Sodium Cooled Reactors: Literature Survey, BNWL-849. Pacific Northwest Laboratory, August 5, 1968.

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