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GEAP - 5087

NUCLEAR SUPERHEAT PROJECT

Final Report

By
H. D. Ongman

December 1965

Atomic Power Equipment Department
General Electric Company
San Jose, California

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NUCLEAR SUPERHEAT PROJECT

FINAL REPORT

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December 1965

U. S. Atomic Energy Commission
Contract AT(04-3)-189
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ATOMIC POWER EQUIPMENT DEPARTMENT

GENERAL ELECTRIC

SAN JOSE, CALIFORNIA

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1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This is the final report covering the progress and results of conceptual designs, economic evaluations, and research and development work performed by the General Electric Company as part of Contract AT(04-3)-189, Project Agreement No. 13. This program was started in July 1959, and was terminated in September 1965.

This final report is intended to give a concise presentation of all published results of the program. To accomplish this, the titles and publication numbers of all reports are presented together with the summary section of each report according to the separate tasks of the program.

In addition to this compilation of reports, results obtained since publication of the Twenty-Fourth Quarterly Report (GEAP-4932) are covered in the quarterly report format.

1.2 SUMMARY

The most significant results on each task for the report period since GEAP-4932 are summarized as follows:

Task A - Program Planning and Coordination

Work on this task was confined to project reporting and administration associated with closing out the project.

Task C - Materials Development

Capsule SC-10 operated satisfactorily through the report period.

Tensile tests at 78 to 1300°F were carried out on coupons irradiated in SC-9. In general, decreases in elevated-temperature post-irradiation ductility were found in all alloys tested. The results are presented along with the results from the SC-8 specimens for comparison.

Specimens from Capsules MT-1 and MT-2 have been tensile tested at 78, 1100, and 1300°F. These tensile specimens have achieved a fast neutron exposure of 3.5×10^{20} nvt, > 1 MeV, in the EBR-II core. A reduction in post-irradiation ductility was shown in all alloys as the tensile test temperature was increased. The decrease in ductility was similar to that found in the SC-8, ESH 1.1, and SC-9 experiments which were performed in substantially equal fast and thermal flux.

Task E - Coolant Chemistry

Testing of Inconel-625 under heat transfer at a calculated uniform metal temperature of 1500°F was completed. The tabulated and plotted data are presented. The compositionally disturbed layers showed no apparent decrease of the layer thickness with time as was noted on layers developed up to 1300°F.

The study of the effect on the oxide and disturbed metallic surface layer of Incoloy-800, pre-exposed at 932°F before exposure at 1150°F, has been completed to an exposure of 3000 hours. The improved condition of the pre-exposed specimens has remained although the data is not sufficient to establish whether the effect is a variation in the initial or the long-term corrosion rate.

The remaining alloys being tested in superheated steam for 10,000 hours were terminated during the period. The assumption of a linear rate after 1000 hours appears reasonable for Hastelloy-F and Incoloy-804 at exposure temperatures of 1050 and 1150°F and for Inconel-718 and René-62 at 1050°F.

Task F - Heat Transfer Tests

Heat transfer and flow friction performance of boundary layer turbulence promoters have been established for superheat fuel applications. The influence of promoter height and promoter spacing has been determined. This investigation has been carried out in the fuel geometry of interest; i.e., a parallel rod array, to avoid uncertainties associated with transformation techniques.

Task L - Mixed Spectrum Superheat Critical Experiment

Measurements in the MSCA facility were completed during this period, and the final report of the task is in preparation.

2.0 TASK A - PROGRAM PLANNING AND COORDINATION

2.1 THE TASK

This task was originally called Conceptual Design Studies and Program Evaluation; thus, in addition to quarterly reports, the design study reports are listed below.

2.2 DOCUMENTATION

GEAP-3290 - First Quarterly Progress Report, July - September 1959.
GEAP-3371 - Second Quarterly Progress Report, October - December 1959.
GEAP-3468 - Third Quarterly Progress Report, January - March 1960.
GEAP-3538 - Fourth Quarterly Progress Report, April - June 1960.
GEAP-3581 - Fifth Quarterly Progress Report, July - September 1960.
GEAP-3686 - Sixth Quarterly Progress Report, October - December 1960.
GEAP-3724 - Seventh Quarterly Progress Report, January - March 1961.
GEAP-3785 - Eighth Quarterly Progress Report, April - June 1961.
GEAP-3877 - Ninth Quarterly Progress Report, July - September 1961.
GEAP-4024 - Tenth Quarterly Progress Report, October - December 1961.
GEAP-3924 - Eleventh Quarterly Progress Report, January - March 1962.
GEAP-4075 - Twelfth Quarterly Progress Report, April - June 1962.
GEAP-4126 - Thirteenth Quarterly Progress Report, July - September 1962.
GEAP-4153 - Fourteenth Quarterly Progress Report, October - December 1962.
GEAP-4243 - Fifteenth Quarterly Progress Report, January - April 1963.
GEAP-4332 - Sixteenth Quarterly Progress Report, May - July 1963.
GEAP-4419 - Seventeenth Quarterly Progress Report, August - October 1963.
GEAP-4565 - Eighteenth Quarterly Progress Report, November 1963 - January 1964.
GEAP-4619 - Nineteenth Quarterly Progress Report, February - April 1964.
GEAP-4703 - Twentieth Quarterly Progress Report, May - July 1964.
GEAP-4755 - Twenty-First Quarterly Progress Report, August - October 1964.
GEAP-4790 - Twenty-Second Quarterly Progress Report, November 1964 - January 1965.
GEAP-4861 - Twenty-Third Quarterly Progress Report, February - April 1965.
GEAP-4932 - Twenty-Fourth Quarterly Progress Report, May - July 1965.

Early in the project the plant economic studies were conducted under this task. These studies were made to establish the economic incentives for nuclear superheat, and to provide direction for the project studies which followed. The plant studies were presented in the following reports:

Pennington, R. T., "Nuclear Superheat Project - Steam Cooled Reactor Study - Economic Study for 300 MW_e Separate Superheater Reactor," GEAP-3589, November 1960.

SUMMARY

This report provides the preliminary design and cost estimate for the nuclear portion of a 300-MW_e Boiling Water Separate Superheater Plant to be in operation by 1967. Fuel cycle costs and the research and development programs required to insure plant feasibility, its cost, and schedule also are presented. In addition, the potential for the Boiling Water Separate Superheater concept which

may be realized in a plant to be in operation in 1975 is assessed in terms of projected plant capacity, technical improvements and potential for cost reduction.

The scope of work has been limited to the accumulation of design, performance is derived from work performed under Task A of the Nuclear Superheat Project under Contract AT(04-3)-189, P.A. 13.

"Nuclear Superheat Project Steam-Cooled Reactor Study Economic Study of the Mixed Spectrum Superheater," GEAP-3590, November 1960.

SUMMARY

Plant Description

The Mixed Spectrum Superheater (MSS) is a nuclear reactor contained within a pressure vessel in which both boiling of water and superheating of the resulting steam takes place. The reactor core consists of four annular regions: an inner unmoderated superheating region, an inner unmoderated buffer region, an outer moderated buffer region, and a boiling region which is very similar to that in a standard boiling water core. The MSS received its name from the fact that the neutron spectrum varies from a thermal spectrum in the boiling part of the core, to an intermediate spectrum between the boiling and superheating sections, to a fast spectrum in the middle of the unmoderated superheater. Water is boiled in the outer boiling core region. The steam-water mixture passes up through parallel sets of radial-type steam separators. The separated steam passes through steam dryers and then travels down through an offset pipe to the inner vessel which houses the unmoderated superheating core. Water surrounding the offset pipe reduces radiation streaming to the vessel head. After passing through the superheating core the superheated steam leaves the pressure vessel through the pipe which penetrates the vessel bottom. Beneath the superheating core within the pipe are a series of shielding plugs which reduce radiation streaming to the vessel bottom and a venturi at the outlet which serves to limit mass flow rate in the event of pipe rupture external to the vessel.

Recirculating water leaves the reactor vessel through three outlet nozzles, and is returned by three parallel mechanical-seal-type pumps through three inlet nozzles.

The reactor is pressurized to 1550 psia and superheated steam is supplied to the turbine at 1415 psia and 950°F. Feedwater is returned at a temperature of 495°F to the reactor vessel. A reactor thermal power of 815 MW of which 30 percent is supplied by the superheater is required to produce 326 MW of gross electrical power at a gross cycle efficiency of 40.0 percent.

The boiler power is controlled manually by bottom-mounted poison elements and the superheater power by top mounted control elements containing fuel. To attain a desired power ratio between the boiling and superheating sections, use is also made of the bottom mounted control (poison) elements in the water moderated buffer region. System steam pressure is regulated automatically by the turbine control valves and superheater outlet temperature by manual operation of the control elements in the superheating section. Load rejection will be absorbed using full capacity bypass valves to the condenser.

Initial conceptual design work on the MSS was performed by the General Electric Company under the company sponsored Advance Engineering program in the Atomic Power Equipment Department. Results of this work indicated that the MSS might have significant advantages over the alternate thermal reactor superheater systems being investigated under AEC programs by General Electric and other companies.

As a result of the high potential indicated in this initial study, G.E. proposed to the Atomic Energy Commission that the work already being performed on superheat for the Commission be extended so as to include a more detailed conceptual design and feasibility study of the MSS. The AEC authorized the beginning of such a study in April 1960.

"Nuclear Superheat Project - Steam Cooled Reactor Study - Economic Study for 300 MW_e Once-Through Superheater Reactor." GEAP-3633, January 1961.

SUMMARY

A conceptual design study of a once-through superheat reactor plant based on experimental heat transfer results is presented.

A preliminary design of a once-through superheat reactor complex and associated reactor equipment of a 300-MW_e power plant which could be in operation in 1967 was developed in sufficient detail to determine the probable reactor plant costs, nuclear fuel cycle costs, and operation and maintenance costs. The research and development program required to ensure plant feasibility, its cost, and schedule are also presented.

Schmidt, R.A., Armour, S.F., and Clancey, W.R., "Design Study of a 600 MW_e Boiling Water Separate Superheat Reactor Plant." GEAP-4057, September 1962.

SUMMARY

This report provides a final design and cost estimate for a 607-MW_e Boiling Water - Separate Superheat Reactor Plant.

The conclusions drawn from this study are:

1. The technical feasibility of this plant depends on the same two factors on which all direct cycle superheat reactors depend; these are the problems of steam-cooled fuel cladding material and direct-cycle system radioactive contamination.
2. Besides these two factors, this superheat reactor plant is mechanically feasible.
3. The superheat reactor is safe.
4. The superheat reactor has good control and operating flexibility.

5. The superheat reactor will require minimum extension of technology gained from the base development programs and from AEC-ESADA-EVESR operating experience.
6. The plant and reactor have the power costs of 5.46 mills/kWh.

3.0 TASK B - FUEL TECHNOLOGY

3.1 THE TASK

The work of this task was concerned with the design, fabrication, and evaluations of fuel elements for the Superheat Advance Development Experiment (SADE) loop in the Vallecitos Boiling Water Reactor (VBWR).

3.2 DOCUMENTATION

The following topical reports were issued to cover the work under this task:

Lees, E. A., Boyle, R. F., and Spalaris, C. N., "Nuclear Superheat Project - Fabrication, Irradiation, and Evaluation of Superheat Fuel Elements SH-1 and SH-2," GEAP-3387, March 1960.

SUMMARY

The following summary is based on the experimental information obtained:

1. The maximum steam temperature obtained during irradiation was 825°F, 1000 psi, thus verifying the feasibility of the nuclear superheat concept. Total irradiation exposure of fuel segments was approximately 440 MWD/T, determined from radiochemical analysis of the irradiated UO_2 .
2. The materials and fabrication process used for the construction of the experimental nuclear superheat fuel elements were satisfactory, with the exception of the expansion bellows which failed in service and the ZrO_2 which powdered under irradiation. The fuel cladding was Types-304 and -347 stainless steel, and the fissionable material was UO_2 in high density compacted and sintered form.
3. Even though the bellows failed during an unintentional fuel temperature transient, the UO_2 pellets were found in good condition after a continuous exposure of about 1 month. During this time, steam entered the interior of the fuel assembly, thus coming into contact with UO_2 .
4. The fission gas release rate from the failed fuel element bellows was estimated at 200 $\mu\text{Ci/sec}$.
5. Hot-cell techniques were developed in the course of these experiments, which permitted:
a.) identification of fuel defect at the bellows, and b.) measurement of accurate power distribution (from gamma scans) along a given fuel element including the flux peaking at active zone discontinuities (middle inert spacer).
6. Carryover of radioactivity was found in one of the assemblies (SH-1) at the vicinity of steam exit.
7. Hairline cracks were found in SH-2 (bellows failure) at the outside diameter of the inner annular cladding. A remnant of the tubing used in SH-2 was examined using nondestructive

and destructive techniques, but no cracks were found similar to those observed in the irradiated fuel element. At present, there is no explanation as to the occurrence of the cracks in the Type-347 stainless steel internal tube of the SH-2 annular fuel element, other than the possibility that these cracks may have been present initially.

8. The ZrO_2 pellets, used in this experiment as inert spacers, were not entirely satisfactory in their performance in that they powdered during irradiation of SH-2 (bellows failure). Other forms of ZrO_2 (stabilized) may serve the purpose if inert spacers are necessary in the future.
9. The maximum capabilities of UO_2 pellets were not utilized in these exploratory experiments. Raising the fuel element specific power is a worthwhile goal.
10. Oxidation rates of stainless steel in reactor superheated steam were comparable to the rates experienced in nonnuclear superheated steam.
11. Recent experience from tubing procurement (stainless) points out the need of careful non-destructive testing of the incoming tubing. The techniques in this area are not well defined, and further development will be necessary especially if thin-wall tubing is to be used in future work.

Raymont, W. R., and Daniels, D. T., "Manufacture of the Adhesive Bonded AEC Superheat Critical Fuel," GEAP-3591, December 1960.

SUMMARY

Superheat fuel elements were designed and fabricated for a critical experiment to be conducted as part of the AEC Nuclear Superheat Development Project.

The fuel element design was based on the bayonet fuel superheat concept, embodying a two-pass annular fuel element. The fuel closure area and steam flow passages were formed by stainless tubes adhesively bonded to aluminum fittings.

Adhesive bonding was selected as the method of fabricating the element to take advantage of the savings offered in material and fabricating costs. A test program was carried out to assure the design adequacy of an adhesive-bonded fuel element, and provide information necessary for the manufacture of the critical fuel.

Loctite, a polyester-type resin, was selected as the reference adhesive for bonding the critical fuel elements. It was shown that it could be used to join aluminum fittings to stainless steel tubes with shear strengths in excess of 450 psi. Adhesive-bonded prototype fuel cells of the critical fuel design showed no evidence of structural failure or water leakage after being exposed to a water pressure of 200 psi for 2 hours. A long-time submergence test (4 months) of a prototype fuel cell under an 8-foot head of water showed the possibility of the pickup of small quantities of water. It was shown that if this problem developed, a small fillet of an epoxy resin on the outside of a joint could become an effective water barrier.

A total of 288 critical fuel elements was manufactured. Pellet fabrication was carried out utilizing conventional powder metallurgy equipment and practices. A production pellet yield in excess of 91 percent was achieved. Fuel element fabrication was carried out in special racks constructed for assembly and bonding the fuel element components. Inspection of incoming material as well as the fuel elements produced was performed using standard manufacturing techniques. Helium mass spectrometer leak checking was the main criterion for fuel element acceptance.

Spalaris, C. N., Comprelli, F. A., and Siegler, M., "Erosion Experiments of Powder Compacted Uranium Dioxide under Dynamic Steam Flow," GEAP-3698.

SUMMARY

Erosion tests of purposely-defected unsintered UO_2 powder-containing fuel elements show that under certain conditions UO_2 powder can escape through defected cladding into the coolant stream. It is too early to assign a cause for the loss of powder in some samples with large cladding defects, whereas in other cases under similar conditions the powder was adequately contained within the purposely-defected fuel specimens. In all cases studied, oxidation of UO_2 to higher oxides was identified both by metallograph and x-ray methods. Specifically, U_4O_9 , as well as U_3O_8 , was found in all defected specimens after the loop exposures of 780 and 946 operating hours. Because the steam (when pure) is a reducing rather than an oxidizing medium, it was concluded that the small amounts of dissolved oxygen added to simulate reactor steam chemistry were chiefly responsible for the observed oxidation of UO_2 . The presence of the oxides identified after the runs are sufficient to reduce the UO_2 bulk thermal conductivity such that thermal performance of the fuel elements would be impaired if the oxidation were to take place in actual fuel elements. It was demonstrated that the escaping UO_2 can be collected in suitable traps and thus prevent contamination of the rest of the system. The specially designed holder was effective enough to contain all of the escaping powder in a predetermined area. The UO_2 escaping from the specimens was oxidized to U_3O_8 as it was determined by the x-ray analysis. The extent of oxidation was about 5 percent.

The arc-fused UO_2 powder presently available is contaminated with nitrogen to concentrations of approximately 2000 ppm. The slight weight gain experienced with some samples was attributed to oxidation of the dioxide and that of the nitride. In the same tests, pellets suffered no weight loss, but rather a slight gain which can be explained because of oxidation to U_3O_8 determined by x-ray and metallographic analysis. Although the available data are very limited, an extrapolation of the results on hand to actual fuel elements in a reactor environment show that excessive downstream contamination may be possible in cases where large-size splits occur in the cladding. On the other hand, fuel cladding defects which are known to have occurred in stainless fuel elements are comparable to those studied in these tests. Under these conditions, downstream contamination would be insignificant, especially if traps are provided into the fuel bundles or other parts of the piping. Moreover, small defects will cause the UO_2 to oxidize, thus permitting the escape of the fission gases in the coolant stream at higher rates than when the UO_2 is unoxidized. The greater fission gas escape rates may, in time, force an unscheduled reactor shutdown, and thus limit the useful fuel lifetime; on the other hand, the higher fission gas release rates will, in fact, permit an early detection of the cladding failure, and facilitate its subsequent identification and removal from the core, thus preventing excessive UO_2 erosion from the defect. Additional work is now being planned to study further the erosion behavior of powder compacted fuel elements.

These erosion tests were conducted out of reactor; consequently, there was no in situ sintering simulating conditions the fuel would see under power performance. How much this sintering action would improve fuel erosion performance under cladding defect conditions is not known. Only in-reactor tests will supply this information.

Esch, E. L., "Development of Process Tubes for Nuclear Superheat Applications,"
GEAP-3784, November 1961.

SUMMARY

Out-of-reactor experiments to establish basic feasibility of two low-cross-section, insulated process tubes were carried out. Two concepts were investigated:

1. An annular, thin-wall stainless steel configuration with Al_2O_3 powder vibratory compacted in the annulus.
2. A Zircaloy-2 tube with a thin-wall stainless steel thermal liner insert.

Fabrication parameters for the annular, stainless steel- Al_2O_3 concept were determined while building 12 prototypes for test purposes. The thermal conductivity of the composite tube wall (stainless steel cladding plus Al_2O_3) was measured and used to estimate in-reactor heat losses with this configuration. At-temperature pressure tests showed that the thin-wall stainless steel cladding would not buckle or deform if the annulus was filled with powder of sufficient compaction density. An inner cladding strengthening mechanism was developed which gave the configuration a differential pressure collapse strength greater than 250 psi with a defect in the outer cladding.

Three Zircaloy-2 simulated process tubes were each exposed for 1000 hours in a dynamic superheated steam loop. One tube was exposed directly to the superheated steam flow; the other two had thin-wall stainless steel thermal liner inserts. All tubes had a uniform, adherent oxide coating after loop exposure and hydrogen pickup was comparable to that incurred in pressurized boiling water. Both insulated tubes had an inside diameter weight gain rate which was extrapolated to predict a useful in-reactor life of 7 years. Fretting of the Zircaloy-2 occurred only where excessive spacer clearances allowed large vibrational effects. One thin-wall thermal liner distorted elastically to block the steam flow passage; a redesigned liner operated successfully.

Both concepts appear to be basically feasible for water-moderated, superheated steam reactor applications. Either should be superior to a solid stainless steel (uninsulated) process tube with respect to thermal neutron economy and reduction of heat losses to the moderator.

Spalaris, C. N., Boyle, R. F., Evans, T. F., and Esch, E. L., "Design, Fabrication, and Irradiation of Superheat Fuel Element SH-4B in VBWR," GEAP-3796, September 1961.

SUMMARY

Irradiation testing of SH-4B, a fuel element designed and fabricated for use in superheat steam applications, yielded valuable information necessary for improving fuel design. The reactor exposure in the VBWR superheated steam loop was terminated when high activity release was detected emanating from the fuel compartment.

Examination of the fuel element, as well as operational data, revealed evidence to show that the jacket failure was due to stress corrosion cracking. Although all conditions necessary to promote stress corrosion cracking were present during irradiation, only the high chloride concentration in the reactor water and water seepage in the loop were unusually high or abnormal and not anticipated in planning the experiment. The test revealed limiting operating conditions in regard to coolant chemistry and provided data such as UO_2 -jacket interactions, dimensional cladding stability, inadequacy of spacer design and method of attachment, stability of UO_2 under severe conditions, heat transfer, and mechanical behavior in a reactor superheated steam environment.

The fuel element tested was 0.028 inch thick, Type-304 stainless steel cladding with an annular geometry cylindrical configuration. The coolant flows downward on the outside, reverses itself at the bottom, then enters the second (inner annulus) pass to exit on top. The UO_2 was in the form of sintered pellets 94 percent minimum theoretical density. The total test period in reactor was 617 hours, with a maximum cladding temperature of 1150°F and maximum steam temperature of 900°F maintained for 80 hours.

Upon post-irradiation examination, the fuel element was found to have performed satisfactorily from the dimensional point of view, but two jacket defects penetrating through the 28-mil outer cladding were found. In addition to these defects, other cracks were found, both in the outside and inside jackets. Cracks were predominantly intergranular in nature, except where sensitization of Type-304 stainless steel was less prevalent, where some transgranular attack and cracks were present.

The uranium dioxide near the defects was oxidized, and the interior of the cladding had a corrosion deposit diminishing with distance away from the cladding defect. These observations, together with the activity release during reactor operation point to the fact that SH-4B operated in the loop with a jacket defect for about 5 to 7 days.

Based on data now available, it may be concluded that abnormally high water leakage in the loop and the concurrent high chloride concentration in the reactor water (1 to 3 ppm maximum for 8 hours) were responsible for the 10,000 parts of chloride per 10^6 parts of iron ions found in the surface deposit. The long-term, wet-dry mechanism for chloride buildup on heat-producing surfaces cannot be precluded as another possible source of high chloride surface concentration.

Information relative to the UO_2 -jacket interaction indicated that no axial expansion of UO_2 occurs relative to the fuel jacket during irradiation. Instead, slight contraction of the fuel column was detected. This information is invaluable in designing thin cladding fuel elements.

Hazel, V. B., Lyons, M. F., and Comprelli, F. A., "Studies of UO_2 -Cladding Mechanical Interactions (Short Time Irradiations in the GETR Trail Cable Facility)." GEAP-4088, October 1962.

SUMMARY

Repetitive irradiations using UO_2 filled capsules were undertaken for the purpose of providing additional data on the diametral growth versus heat flux correlation reported in GEAP-3739. Irradiation was accomplished in the GETR Trail Cable Facility, either in one single step or in repetitive steps. To follow the movements of UO_2 due to thermal expansion, many capsules were pressurized, following the irradiation step, in an autoclave at room temperature. The results from these repetitive capsule irradiations are in general agreement with the results reported in GEAP-3739; i.e., the external cladding is subjected to cyclic plastic strain due to UO_2 growth during irradiation and UO_2 cladding shrinkage under external pressure back to approximately their pre-irradiation dimensions following irradiation. A comparison of the first and second irradiation diametral growth magnitudes observed in the three pellet-filled capsules provides additional evidence for the influence of the capsule internal fuel-cladding gap on cladding expansion.

External cladding growth due to irradiation was observed for compacted powder-filled capsules. One powder-filled capsule was repetitively irradiated, with results similar to those for the pellet-filled capsules above. The data are not sufficient to determine if the growth magnitude is different from that experienced with pellet-filled capsules.

Six solid, cylindrical pellet-filled, 0.485-inch outside diameter capsules were irradiated, and diametral and axial growth were measured. Two capsules with swaged cladding (small initial gap ~ 1 mil) and the two with ~ 3 -mil initial gaps exhibited diametral growth, while none was observed in the two capsules having 9-mil initial gap. Axial growth was observed in the four unswaged capsules and essentially zero growth or slight shrinkage in the two swaged capsules. Axial growth* in three of the four unswaged capsules was about 20 mils.

All of the annular capsule axial growth measurements were analyzed in search of a correlation with the capsule fabrication parameters and/or operating conditions. Both axial growth and shrinkage were observed as a result of irradiation. Empirical correlation attempts with cladding strength; initial, external fuel cladding gap; measured external cladding, diametral growth; and operating heat flux indicated only random variations. Data on the initial axial gap incorporated in the capsules was only approximate, and could not be used for correlation purposes.

* Total fuel element growth from end plug to end plug. Differential UO_2 cladding growth was not measured in these experiments.

Spalaris, C. N., Rayment, W. R., Lyons, M. F., Evans, T. F., Boyle, R. F., Murdock, T. B., and Weiss, N. D., "Fuel Element Experiments in SADE-VBWR Nuclear Superheat Loop," GEAP-4240, May 1963.

SUMMARY

Fuel element experiments were conducted at G.E. - APED from July 1961 to January 1962, with the following objectives:

1. Test improved design and fabrication process,
2. Obtain additional data on failures brought about by irradiation, and
3. Obtain additional information on operational characteristics of the loop.

The experimental results presented in this report confirm earlier data obtained during the SH-4B fuel element test.* Leakage through the flange separating the inlet steam flow from the boiling water moderator resulted in high deposit accumulation on fuel superheat surfaces. Eliminating the flange leakage by seal welding resulted in less crud accumulation. The moisture present in the inlet steam was found to be sufficient to cause some crud accumulation which was associated with the area of the fuel element SH-5A found to be extensively cracked.

All cracks found by metallographic examination were intergranular, with some cracks extending into grain twins. Secondary cracks, adjacent to or away from the main cracks, were found to emanate from the steam side. Grains were dislodged from the cladding matrix exhibiting "holes" with rounded edges; an indication that the corrodant steam and/or impurities played a leading role in dissolving grain boundaries or adjacent grain matrix.

Cracks were highly localized and branched. Areas distant (0.1 inch) from the main cracks were found to be intact and ductile. Element SH-4C grain deformation was found at the edges of the crack, showing that mechanical deformation (stresses) played a leading role in crack propagation.

Crud analysis showed the presence of chloride ions in relatively high concentrations. Quantitative analysis for chloride ions was difficult because of size of crud sample; therefore, a material balance between crud deposit, coolant flow, and moisture carry-over was not possible.

Dimensional measurements showed 0.028-inch-wall fuel cladding to collapse uniformly around the UO_2 at the high heat flux region. Inside diameters showed no significant changes in the 0.028-inch-thick cladding fuel. The 0.016-inch cladding fuel showed uniform collapse both on the inside and outside diameters. The plenum design support was successful. This design was based upon the information obtained from SH-4B regarding UO_2 -cladding relative movement. The "gum drop" in SH-4C confirmed earlier findings that the UO_2 charge did not move toward the plenum, but rather contracted slightly. The over-all fuel length (end plug to end plug) did not change for SH-5A, SH-4, or SH-4B. Measurements for SH-4C show growth, but in view of results to the contrary for elements SH-5A, SH-4, SH-4B, and NUSU, it is suspected that pre- or post-irradiation measurements of SH-4C are in error.

*Spalaris, C. N., Boyle, R. F., Evans, T. F., and Esch, E. L., "Design, Fabrication, and Irradiation of Superheat Fuel Element SH-4B in VBWR," GEAP-3796, September 1961.

Busboom, H. J., and Snyder, M. E., "Fabrication and Testing of Dissimilar Metal Joints for Thermal Spectrum Superheat Reactors," GEAP-4756, December 1964.

SUMMARY

The application of Zircaloy-2 for process tubes in thermal spectrum superheat reactors presents the problem of joining Zircaloy-2 to austenitic stainless steel alloys. A review of literature related to the problem of joining these metals revealed that the most promising bonding method was hot tandem extrusion.

Diffusion bonding experiments were considered, but were not carried out since literature data showed this method to be less favorable for large-diameter tubes (greater than 1.0 inch) because of problems in temperature control and accurate alignment of parts. The large intermetallic layers formed during bonding are unfavorable for ductile joints.

Hot extrusion experiments demonstrated that well-bonded, large-diameter tubular joints (1.75-inch o.d. $\times$ 0.065-inch wall) can be fabricated between Zircaloy-2 and Incoloy-800 or Type-304L stainless steel. It was also demonstrated that long Zircaloy-2 tubes can be extruded with these joints, thus allowing the fabrication of a complete full-size process tube, resulting in a substantial cost reduction.

Ultrasonic testing and metallography revealed that these joints were 100 percent bonded, and that they would have adequate strength and elastic stability. The joints were found to have adequate corrosion resistance in a simulated superheat reactor environment where steam and boiling water temperatures, pressures, and flow conditions were maintained for 3472 hours. Measurements made during the corrosion run indicated that the stagnant steam gap which was maintained between the process tube and liner provided excellent insulation to loss of heat from the superheat to boiling side of the process tube. Thermal cycling between 100 to 600°F did not appear to be detrimental to joint integrity. The elastic mechanical behavior of these joints was similar to that of the base metals. The plastic mechanical properties were found to be limited by the Zircaloy-2 rather than the joint material.

Reproducibility of fabrication quality from joint to joint was found to be excellent.

4.0 TASK C - MATERIALS DEVELOPMENT - C. N. Spalaris, H. J. Busboom, S. Y. Ogawa, and G. F. Rieger

4.1 MATERIALS IRRADIATIONS - THERMAL FLUX

These tests are being conducted in the GETR. Three separate experiments have been or are being conducted at temperatures ranging from 1000 to 1300°F. All experimental capsules contain tensile specimens of various alloys selected for superheat fuel cladding applications. Alloys which have received heavy attention and are being investigated in depth are Incoloy-800 and Inconel-625. Hastelloy-X, Incoloy-801, Incoloy-825, and other high nickel alloys were also included for study.

Capsule SC-8 was discharged from the GETR on August 1964. The accumulated exposure for the coupons was 10^{20} nvt. The results were reported in the last two monthly reports.

Capsule SC-9 was discharged from the GETR in August 1965. The data obtained are reported in this report.

Capsule SC-10 was loaded into the GETR in August 1965. It contains burst as well as tensile coupons.

4.1.1 Irradiation of Capsule SC-10 in the GETR

Capsule SC-10 operated satisfactorily during the reporting period, and completed one cycle (No. 68) on September 26. Irradiation is continuing with all of the thermocouples reading the desired temperatures of 1000 to 1300°F.

4.1.2 Capsule SC-9 Irradiated in the GETR

Capsule SC-9 was irradiated in the GETR for nine cycles (59 through 67). The tensile coupons tested accumulated a fast neutron exposure of 2.0×10^{21} nvt, >1 MeV. The irradiation period was 5200 hours, and only the coupons exposed at 1300°F were tested. Selected tensile test temperature ranged from 78 to 1300°F. The strain rate was held at 0.02 min^{-1} , making these data comparable to earlier SC-8 data.

Results Obtained

The results from tests performed on the irradiated coupons are compared in Table 4-1 with results from control coupons having the same thermal history. Also included are results from the SC-8 experiment for comparison. In general, decreases in elevated temperature post-irradiation ductility were found in all alloys tested. The minimum ductility, as measured by percent elongation, for Incoloy-800 was found to be 7 percent at 1300°F. The general, stress-strain behavior for this alloy was not changed by the irradiation exposure. The total elongation was reduced, indicating that the fracture mode rather than plastic behavior was affected by the fast neutron exposure. Age hardening reactions are known to occur at the exposure temperature for the other alloys tested: Inconel-625, Incoloy-801, Incoloy-825, and Hastelloy-X. As can be observed from the yield stress values for Inconel-625, the irradiated specimens were weaker and more overaged than the control coupons at all test temperatures, suggesting that fast neutron exposure enhances precipitation in this alloy. Enhanced precipitation caused more overaging, and thus a reduction in strength. This reduction in post-irradiation ductility was more severe in Inconel-625 than Incoloy-800 (~1 percent net at 1300°F), and was probably due to a combination of radiation damage and precipitation.

TABLE 4-1

EFFECT OF NEUTRON EXPOSURE* IN THE GETR ON THE TENSILE PROPERTIES
OF SELECTED SUPERHEAT ALLOYS

Material	Exposure Time** (h)	Fast Neutron Exposure (nvt)	Tensile Test Temperature (°F)	Yield Stress (ksi)	Ultimate Stress (ksi)	Percent Elongation	
						Uniform	Total
Incoloy-800	5200	-	78	31	81	44	50
	5200	2.0×10^{21}	78	31	78	39	45
	630	-	1100	24	55	31	38
	5200	-	1100	20	57	34	41
	630	3.5×10^{20}	1100	21	48	19	22
	5200	2.0×10^{21}	1100	21	47	13	14
	630	-	1300	21	37	18	47
	5200	-	1300	20	38	19	71
	630	3.5×10^{20}	1300	20	35	12	25
	5200	2.0×10^{21}	1300	18	32	7	7
	630	-	1100	49	85	18	22
	630	3.5×10^{20}	1100	40	41	-	1
Incoloy-801	630	-	1300	45	54	9	32
	5200	-	1300	39	54	12	45
	630	3.5×10^{20}	1300	38	41	-	1
	5200	2.0×10^{21}	1300	35	36	-	<1
	630	-	1300	36	54	11	46
	5200	-	1300	41	60	10	49
Incoloy-825	630	3.5×10^{20}	1300	37	51	6	6
	5200	2.0×10^{21}	1300	34	46	3	3
	5200	-	78	120	169	31	32
	5200	2.0×10^{21}	78	111	154	23	23
Inconel-625	630	-	1100	91	146	22	23
	5200	-	1100	104	146	19	20
	630	3.5×10^{20}	1100	74	122	8	8
	5200	2.0×10^{21}	1100	92	114	3	3
	630	-	1300	87	106	8	20
	5200	-	1300	89	110	7	23
	630	3.5×10^{20}	1300	75	90	4	4
	5200	2.0×10^{21}	1300	80	88	1	1
	5200	-	1100	56	111	20	28
	5200	2.0×10^{21}	1100	59	93	4	5
	5200	-	1100	56	111	20	28
	5200	2.0×10^{21}	1100	59	93	4	5
Hastelloy-X	5200	-	1100	56	111	20	28
	5200	2.0×10^{21}	1100	59	93	4	5

* The fast-to-thermal flux ratio for these experiments was approximately 1.

** At 1300°F in inert atmosphere.

It is interesting to note that the room temperature properties are only slightly affected by the neutron exposure, as it was with Incoloy-800.

Drastic decreases in post-irradiation ductility, ~1 percent at 1300°F, were also observed for Incoloy-801 similar to what was observed for this alloy in the earlier SC-8 experiment. It appears that the damage causing this elevated temperature embrittlement is well established at relatively moderate neutron $\sim 10^{20}$ nvt, >1 MeV, and remains to relatively high exposures of $>10^{21}$ nvt. The post-irradiation mechanical exposures of behavior of Incoloy-825 is similar to the other alloys which show precipitation with only 3 percent elongation remaining at 1300°F.

In summary, it appears that all high nickel alloys of interest in the superheat program lost high-temperature ductility as a result of irradiation. For a given exposure, alloys which have age hardening reactions (precipitation occurring during the reactor exposure) exhibited lower values of post-irradiation ductility. As the tensile test temperature is increased, the reduction in post-irradiation ductility is also increased.

The observations show that the cladding temperature should be held as low as possible, and that alloys which are relatively stable metallurgically are best suited for superheat cladding.

4.2 MATERIALS IRRADIATION PROGRAM - FAST FLUX EBR-II

The objective of this program is to investigate the effects of fast neutron radiation on the elevated temperature properties of superheat cladding without the side effects of thermal neutron transmutations. The three alloys selected for investigation were Incoloy-800, Inconel-625, and Hastelloy-X.

4.2.1 Capsules MT-1 and MT-2

These capsules were irradiated in EBR-II at $\sim 1000^\circ\text{F}$ for 200 hours. Tensile coupons tested from these capsules achieved a fast neutron exposure of 3.5×10^{20} nvt, >1 MeV, in the EBR-II core. These exposure values have been determined from chemical analysis of wire monitors included in the capsules during fabrication. Earlier values reported were preliminary, and the present indicated exposures supersede them. Selected tensile test temperatures were 78, 1100, and 1300°F. Most of the tests were performed at a strain rate of 0.02 min^{-1} (slow), but a few tests were performed on Incoloy-800 at 2.0 min^{-1} to determine the effect of strain rate on post-irradiation ductility at elevated temperatures. Results obtained are shown in Table 4-2.

Since Incoloy-800* does not show an age-hardening reaction, the results for this alloy were not complicated by precipitation as were the results from the other two alloys which do show an age-hardening reaction. In general, all alloys showed a reduction in post-irradiation ductility, measured by total elongation, as the tensile test temperature was increased. This decrease in ductility was similar to that found in the SC-8, ESH-1.1, and SC-9 experiments. The latter three irradiation experiments were performed in reactors having substantially equal fast and thermal flux. It appears that fast neutron exposure alone plays an important role in the reduction of post-irradiation ductility of these alloys.

*With low aluminum and titanium contents.

TABLE 4-2
EFFECT OF FAST NEUTRON EXPOSURE ON THE TENSILE PROPERTIES OF
SELECTED SUPERHEAT ALLOYS

EBR-II Fast Neutron Damage Studies

Material	Test Temperature (°F)	Strain Rate (min ⁻¹)	Fast* Neutron Exposure (nvt)	P.L. (ksi)	0.2 Percent Yield (ksi)	Ultimate (ksi)	Percent Elongation	
							Uniform	Total
Incoloy-800	78	0.02	-	36.0	38.7	83.9	35.7	38.1
	78	0.02	3.3×10^{20}	33.0	43.5	87.7	33.0	34.0
	1100	0.02	-	25.0	30.2	61.6	28.0	32.6
	1100	0.02	3.3×10^{20}	23.0	25.3	55.3	20.1	22.3
	1100	2.0	-	24.0	27.0	64.5	26.9	28.6
	1100	2.0	3.3×10^{20}	29.3	31.7	55.7	19.6	21.9
	1300	0.02	-	23.0	24.3	37.5	17.3	63.2
	1300	0.02	3.3×10^{20}	18.3	22.3	30.9	7.6	12.8
	1300	2.0	-	21.0	22.4	50.9	29.7	39.1
	1300	2.0	3.3×10^{20}	26.0	38.7	48.6	19.5	26.1
Inconel-625	78	0.02	-	62.0	66.3	141.0	48.7	50.0
	78	0.02	3.3×10^{20}	97.3	102.5	163.9	38.7	30.0
	1100	0.02	-	46.0	51.5	119.0	47.9	49.5
	1100	0.02	3.3×10^{20}	69.0	75.8	124.5	13.8	14.1
	1300	0.02	-	46.0	48.1	96.4	24.3	36.0
	1300	0.02	3.3×10^{20}	64.9	73.2	92.5	5.2	5.8
Hastelloy-X	78	0.02	-	40.0	43.5	99.3	50.1	54.1
	78	0.02	3.3×10^{20}	56.5	65.3	137.0	15.5	16.0
	1100	0.02	-	24.0	26.5	76.0	54.5	57.3
	1100	0.02	3.3×10^{20}	43.0	52.3	102.9	14.4	14.7
	1300	0.02	-	22.0	26.7	61.7	40.9	49.8
	1300	0.02	3.3×10^{20}	41.3	48.3	62.0	5.4	8.4

* Irradiation was performed in EBR-II where thermal flux is negligible. Assume all flux was >50 keV. Exposure temperature was 1000° F for ~200 hours.

The large increase in post-irradiation yield stress found in Hastelloy-X and Inconel-625, but not in Incoloy-800, indicates that accelerated precipitation had occurred due to the very high fast flux, $\sim 4 \times 10^{14}$ nvt, > 1.34 MeV, of EBR-II. At the exposure temperature, $\sim 1000^\circ\text{F}$, one would normally not expect this degree of precipitation to occur in 200 hours without the presence of the fast neutron flux. The lower ductility found for these alloys was probably related to a combination of precipitation and neutron damage accumulated during the reactor exposure.

An increase in ductility from 13 percent to 26 percent total elongation was found by increasing the strain rate at 1300°F for the irradiated Incoloy-800. It appears that the deformation mechanism also plays an important role in elevated temperature ductility of irradiated austenitic alloys.

4.2.2 Capsules MT-3 and MT-4

These capsules are similar to MT-1 and MT-2, and contain 72 tensile coupons to be irradiated in EBR-II to a target exposure of 10^{22} , or greater, > 1 MeV. They are presently being considered for insertion in a long-term irradiation experiment being conducted by ANL. The expected irradiation is for 2 to 3 years in the EBR-II core starting this month.

4.3 OUT-OF-REACTOR HIGH-TEMPERATURE MECHANICAL TESTS

4.3.1 Stress Rupture Test Program

A series of stress rupture tests of Inconel-625 tubing is in progress. Some of these specimens were made from as-received tubing and others from tubing which had been preconditioned in the steam corrosion loop. Pertinent data of specimens tested at 1350°F are present in Table 4-3.

TABLE 4-3
MAXIMUM DIAMETRAL EXPANSION

Specimen	Initial Hoop Stress (psi)	Loop Sheath Identification	Loop Operating Temperature ($^\circ\text{F}$)	Loop Exposure Time (h)	Maximum Diameter Expansion	
					Time (h)	Diameter Change (in/in, percent)
4-18-A*	20,000	as received	-	-	1016	1.00
4-19-A	11,880	IC-10	1135 to 1196	618	1131	0.21
4-19-B	11,880	IC-10	1208 to 1281	618	1131	0.21
4-19-C	11,880	IE-4	1330 to 1364	2052	1131	0.62
4-19-D**	11,880	IE-4	1364 to 1398	2052	1283	17.6
4-19-E	11,880	IE-2	922 to 991	3316	184	0.09
4-19-F	11,880	IE-2	1004 to 1054	3316	184	0.61

* At approximately 1000 hours. 4-18-A developed a small leak at the end plug weld, and was removed from test at 1016 hours.

** Failure.

4.3.2 Intergranular Fracture Evaluation Tests

This work is to quantitatively evaluate, at creep conditions, the time-dependent magnitude of potential microleaks through cladding which may occur prior to catastrophic rupture of the cladding. Test specimens are subjected to internal helium gas pressures at temperatures in excess of 1200°F, and are located within a vacuum chamber so that low helium leakage rates through the cladding may be monitored.

Fabrication of the test equipment and the initial calibration and debugging operations are essentially complete. The final checkout of the system must await delivery of the special recorder which has been shipped from the vendor and is expected during the first week in October.

4.3.3 Multi-Axial Creep and Deformation Programs

Rupture and creep tests on Incoloy-800 (commercial) tubing are in progress. Test results obtained to date are presented in Table 4-4.

TABLE 4-4
CREEP AND RUPTURE DATA

Specimen	Outside Diameter (inch)	Wall (inch)	Temperature (°F)	Initial Hoop Stress (psi)	Maximum Diameter Expansion	
					Time (h)	Diameter Change (in/in, Percent)
5-8-A	0.450	0.020	1200	15,800	580	0.38
5-8-B	0.450	0.020	1200	15,800	580	0.38
5-8-C	0.450	0.016	1200	20,000	580	0.58
5-8-D	0.450	0.016	1200	20,000	580	0.44
5-8-E*	0.450	0.012	1200	26,800	<60	20.9
5-8-F*	0.450	0.012	1200	26,800	60	-
6-9-A	0.450	0.020	1100	15,800	658	0.04
6-9-B	0.450	0.020	1100	15,800	658	0.09
6-9-C	0.450	0.016	1100	20,000	658	0.09
6-9-D	0.450	0.016	1100	20,000	658	0.09
6-9-E	0.450	0.012	1100	26,800	658	2.50
6-9-F	0.450	0.012	1100	26,800	658	2.50
9-2-A**	0.450	0.016	1300	13,600	59	0.04
9-2-B**	0.450	0.016	1300	13,600	59	0.09
9-2-C	0.450	0.016	1300	13,600	59	0.11
9-2-D	0.450	0.016	1300	13,600	59	0.09

* Both specimens failed. -E by a slow leak which occurred within 5 hours of shutdown, and -F by rupture at 60.4 hours.

** These specimens were instrumented with strain gages.

In addition to the constant load tests reported in Table 4-5, cyclic tests are continuing. In cyclic tests, tubular specimens are pressure cycled by subjecting the specimen to an external pressure, and then alternating the internal pressure to values above and below the external pressure. The specimens contain internal mandrels which provide diametral gaps of 4 to 10 mils as measured at room temperature. The primary purpose of the mandrel is to limit the tubing deformation during the excess external pressure portion of the cycle so that the tubing would not collapse because of elastic instability. However, the fabricated gap will not necessarily prevent collapse if the tubing undergoes creep deformation during the test sufficient to develop a locally unstable condition, such as would occur during the formation of a longitudinal wrinkle.

In all tests the pressure is cycled. In some, the temperature is maintained at a constant value, in others during the external pressure excess portion of the cycle, the temperature is allowed to decrease exponentially to approximately 550 to 600°F (requires from 1 to 2 hours), and is then increased to the listed test temperature during the internal pressure excess portion of the cycle. The former will be designated by C, and the latter by V.

At present, effort is concentrated on Incoloy-800 and Inconel-625 materials. However, previous tests not reported to date have utilized some other potential materials of interest. For completeness, these are listed in Table 4-5. The pressures are described in terms of initial hoop stress values in the cladding. The initial stresses and the stresses at time of failure are always tensile. The time to failure reported is the cumulative time the specimen has been subjected to tensile hoop stress, and not the total test time.

TABLE 4-5
CYCLIC CREEP TESTS

Specimen	Material	Temperature (°F)	Outside Diameter (inch)	Wall (inch)	Initial Hoop Stress		Number Cycles to Failure	Adjusted Time to Failure
					Tensile (psi)	Compression (psi)		
3-5-A	Type 304 Stainless Steel	V-1200	0.498	0.010	12,250	24,500	5	26
3-6-A	Type 304 Stainless Steel	V-1200	0.498	0.010	12,250	24,500	4	22
3-7-A	Incoloy-1	V-1200	0.500	0.010	20,100	24,500	5	26
3-8-A	Hastelloy-X	V-1200	0.501	0.010	28,200	24,500	72	428
3-9-A	Type 310 Stainless Steel	V-1200	0.500	0.010	12,250	24,500	41	242
3-10-A	Type 304 Stainless Steel	V-1200	0.501	0.010	12,250	24,500	3	11
3-11-A	Incoloy-2	V-1200	0.450	0.020	16,200	10,800	52	307
3-12-A	Incoloy-2	V-1200	0.450	0.020	24,840	5,400	19	189
3-13-A	Incoloy-2	C-1000	0.450	0.020	24,840	5,400	177	1057
3-14-A*	Incoloy-2	C-1000	0.450	0.020	16,200	10,800	170 ⁺	1020 ⁺
3-16-A**	Incoloy-2	C-1350	0.450	0.020	6,680	10,750	41 ⁺	246 ⁺

*Test interrupted prior to failure.
**Test in progress.

Note: Incoloy-1 is commercial Incoloy-800, and Incoloy-2 is the G.E. modification of Incoloy-800.

4.4 DOCUMENTATION

Lyohs, M. F., Comprelli, F. A., Hazel, V. E., and Townsend, H. E., "Plastic Strain in Thin Fuel Element Cladding Due to UO_2 Thermal Expansion," GEAP-3739, July 1961.

SUMMARY

A total of 36 fuel-bearing capsules has been irradiated and examined to date in the program with the following generalized results:

1. The diameter of the capsule external cladding increased significantly as a result of irradiation for almost every capsule.
2. The over-all length of the capsules increased with only one or two exceptions, and the length increase was found to vary in a consistent pattern around the periphery of the capsule.
3. No significant change was observed in the diameter of capsule internal cladding with one possible exception.

Analysis of the external cladding dimensional changes has shown the magnitude of the change to be a function of the restraining forces imposed on the UO_2 by the cladding as well as a function of the capsule heat generation and the initial pellet-cladding gap. These functional relationships are stated in general terms as follows:

1. The diameter change, or growth, decreases with increased cladding strength.
2. The diameter change increases essentially linearly with increasing capsule heat generation.
3. The diameter change decreases somewhat with increased initial pellet-cladding gaps. However, the UO_2 appears to thermally expand sufficiently to close any of the initial gaps within the range studied - 0 to 11 mils - and produce cladding growth.
4. For the particular capsule configuration and operating conditions employed in these tests, the maximum expected outside diameter increase is approximately 10 mils, equivalent to 0.8 percent total strain.

The plastic straining of the cladding observed as the result of a single cycle of irradiation in these tests will probably be cyclic with changes in power generation when the cladding is nonfree-standing under the existing external pressure. It was experimentally determined that the UO_2 shrinks back to approximately its original dimensions when power generation ceases.

Analysis of the observed capsule length changes has not been completed. The changes in length appear to be the result of action of the pellets against the adjacent cladding rather than direct axial expansion since changes in the initial axial gap did not significantly affect the increase in length. The length increases varied sinusoidally around the capsule periphery, indicating that the relatively heavy end plugs were tilting with respect to each other, probably as a result of circumferentially nonuniform heat generation within the capsule.

Relatively large amounts of gas - $6 \times 10^{-2} \text{ cm}^3/\text{gm}$ - were found to evolve during irradiation from the conventionally processed, sintered UO_2 pellets used in the initial capsules. By modifying the capsule design and improving the capsule fabrication procedure, the magnitude of the gas evolved in subsequent capsules was reduced significantly and internal gas pressure eliminated as a possible contributor to the observed cladding growths. Comparison of subsequent growth results with those obtained previously indicated several cases where internal gas pressure was probably a factor. These capsules were not considered in data analyses or in forming the experimental conclusions.

No difference was observed within the experimental error between swaged and unswaged capsules in producing outside cladding growth other than that attributable to generally smaller initial gaps present in swaged capsules.

New equipment was designed and built, and a completely new irradiation technique was developed to permit the rapid conduct of these capsule irradiations in the GETR Trail Cable Facility and the post-irradiation dimensional measurements in the GETR canal.

Spalaris, C.N., Comprelli, F.A., and Douglass, D.L., "Materials for Nuclear Superheat Applications. A Literature Survey," GEAP-3875, January 1962.

SUMMARY

A survey was conducted to examine the information available in the literature which may be applicable in selecting one or more candidate metal alloys for use in high-temperature nuclear superheated steam environments. The survey was divided into four principal categories:

1. Practical experience of material performance in high-temperature steam service conditions.
2. Oxidation and corrosion behavior of materials at high temperatures.
3. High-temperature mechanical properties.
4. Effect of irradiation upon the physical properties of high-temperature alloys.

Information which may lead to the clear-cut selection of one or a group of candidates for use in nuclear superheat environments has not been located. Instead, the available data permitted selection of several alloys which, when properly studied in the environment desired, might safely be used in commercial superheat reactors. The development work necessary for selecting promising alloys for reliable performance consists mainly of the broad task of experimentally determining their operating limits in nuclear superheated steam environments, and then of reducing to fabrication and design practices the means of using these alloys as cladding.

This was not intended to be an exhaustive and all-inclusive literature survey; instead, information included is that which is most pertinent and directly applicable to the subject, and which could be assimilated and integrated within the time allowed.

Marburger, I. L., Haglund, E. A., Hafer, D. E., Liffengren, D. J., and Armour, S. F.,
"Design Report Superheat Strain Cycle Capsule," GEAP-3933, January 1963.

SUMMARY

The Superheat Strain-Cycle task was undertaken to investigate low cycle fatigue properties of 300-series stainless steel nuclear fuel cladding material in a reactor environment at 1300°F.

Capsules, each containing three strain-cycle test specimens, were designed, fabricated, and irradiated. Mandrels internal to the capsule produced hoop strains of 2, 1, 1/2, 1/4, 1/8, and 1/16 percent of the specimen outside diameter. Gas gaps, strain gaps, and mandrel thicknesses were individually sized to account for anticipated gamma heat variances during irradiation.

Each capsule was irradiated in a core position in the General Electric Test Reactor (GETR), Vallecitos Atomic Laboratory, Pleasanton, California, for one irradiation cycle of approximately 23 days at nuclear power. Some specimens were cycled only during irradiation, whereas other specimens were irradiated and then transferred to the reactor fuel storage canal for fatigue-limit cycling at 1300°F. Cycling was carried out on 30-minute intervals.

Cycling was accomplished by pressurizing alternately an external and an internal annulus with a 1000-psig helium supply. A special console located on the third floor of the reactor provided a manual control for proportioning helium and nitrogen gas mixtures used for specimen temperature control. The console provided individual and independent temperature control for the three specimens in one capsule, a timer which governed the action of the pressurizing solenoids, and a specimen-failure sensing system.

Post-irradiation cycling employed the same apparatus used for in-core cycling. Heater power supply and temperature controls were provided by a pyrometer saturable reactor system.

From the irradiation and dynamic cycling of four capsules containing a total of 12 specimens, it is concluded that:

1. The apparatus operates successfully both in the laboratory for the collection of pre-irradiation data, and in a reactor high temperature (1300°F) neutron environment simulating superheat conditions;
2. The capsules generated low-frequency, low-cycle fatigue data during and after irradiation on Type 304 stainless steel tubular specimens suitable and available for fuel element mass production; and
3. The apparatus can demonstrate its usefulness in testing other alloys as well.

Reynolds, Myron B., "Slow Cycle Strain Fatigue in Thin Wall Tubing: Preliminary Report," GEAP-3983, July 1962.

SUMMARY

A technique has been developed for slow cycle strain fatigue testing using specimens of thin-wall tubing of the type under consideration for use as superheat fuel cladding. Data on Type-304 stainless steel has been obtained under radiation and in the absence of radiation. The strain cycle fatigue life of this alloy at 1200 to 1300°F is decreased threefold by the presence of a neutron flux of 1.5×10^{14} >1 MeV. Out-of-reactor data has been obtained on Inconel at 1300°F.

Klepfer, H.H., Douglass, D.L., Cummings, W.V., Reynolds, M.B., and Antony, K.C., "Remarks Concerning Materials for Nuclear Superheat Fuel Cladding," GEAP-4060, September 1962.

SUMMARY

Metallurgical and corrosion work performed by many sites including research performed by General Electric on the Nuclear Superheat Project and on other programs were reviewed to comment on alternate possibilities to Type-304 stainless steel as superheat fuel cladding. It was concluded that the emphasis being placed on Inconel and Incoloy-800 in the Nuclear Superheat Project was well founded. Alloys of zirconium, niobium, molybdenum, beryllium, and chromium were all eliminated as practical alternates. Ferritic alloys were considered too weak for the application. Precipitation hardening, high-nickel alloys were shown to be seriously limited by metallurgical instability, particularly in the presence of neutron radiation at the proposed service temperature. Further research on the solid solution, high-nickel alloys of the Inconel and Incoloy type was recommended with emphasis on modification of alloy composition and closer control of chemical contaminants in the coolant to improve corrosion performance.

Kirby, R.F., "Evaluation of 300 Series Stainless Steel Tubing for Fuel Cladding," GEAP-4095, October 1962.

SUMMARY

The burst strength of Type-304 stainless steel tubing was evaluated by burst-testing artificially defected tubing samples. Tubing geometry was 0.875-inch o.d. $\times$ 0.016-inch wall thickness, and two metallurgical conditions, annealed and 1/8 hard, were tested. Test specimens were artificially defected by spark machining techniques to give defect depths ranging from 10 to 30 percent of the wall thickness. The specimens were burst tested at temperatures ranging from 1250 to 1350°F, and the results of pressure versus time were automatically plotted using an X-Y recorder.

Tensile tests were performed at 1300°F on Type-304 stainless steel tubing to determine the yield strengths and ultimate tensile strengths. The procedure used was in accordance with ASTM testing practices. Tubing tested was of 1.25-inch o.d. and various wall thicknesses, including 0.016 inch, 0.018 inch, and 0.028 inch.

Included are samples of tubing from four different vendors and in several diameters, wall thicknesses, and cold-work conditions. Tubing was ultrasonically tested to locate defects greater than 5 percent of the wall thickness and thus allow a correlation between number of defects and tubing parameters such as size, degree of cold work, and manufacturer.

Defects having a depth magnitude equivalent to 20 percent of the tube wall thickness were found to affect the tubing strengths, as determined by burst tests. In addition, a sampling of "as-received" tubing from several fabricators exhibited only one defect in excess of 10 percent of the wall thickness in 50 feet of tubing (regardless of material and geometry).

Based on this information, a specific inspection limit was established for incoming tubing which is in line with practical tubing fabrication processes and not likely to result in significant cost penalties. This limit is that tubing with defect sizes in excess of 10 percent of the wall thickness shall be rejected.

The acceptance limit has been established on the basis of laboratory test data and not on fuel element performance. A correlation between defects and fuel performance will require an extensive development program, certainly outside the scope of this task. Using the ground rule that if a tubing defect exists it offers a potential fuel element rupture, there remains for one to define the method of detection and the size of defect that can be acceptable. The imposition of a limit of acceptance must be in line with the present tubing fabrication practice and the required fuel element performance. A convenient method for establishing such a limit is the burst strength of tubing as a function of defect size. Although this limit may not correspond to a statistical number of fuel element defects tolerable during a given fuel cycle, it can be a practical limit from the tubing acceptance standpoint which can be further improved when reactor fuel performance information becomes available. It is believed that the methods used for establishing such a limit are realistic from the tubing burst strength standpoint in the unirradiated condition.

Rieger, G. F., "Applications of Strain Cycling Considerations to Superheat Fuel Design," GEAP-4244, June 1963.

SUMMARY

A potential performance limitation of superheat fuel is the susceptibility of the fuel cladding to low cycle fatigue failure. Two simplified analytical methods are presented to estimate the cyclic life-time of circular superheat fuel cladding.

One failure relation, based on a displacement method, is of the form

$$N_f = 0.33 \left[\frac{C}{\Delta \epsilon_p} \right]^n \quad (1)$$

where

N_f = Cycles to failure,

$\Delta \epsilon_p$ = Plastic strain range of the power cycle, and

C, n = Material property constants.

The other failure relation, based on a stress method, is of the form

$$N_f = \frac{1}{12} \left[\frac{EC}{S - S_e} \right]^2 \quad (2)$$

where

- N_f = Cycles to failure.
 S = Elastic stress amplitude of the power cycle, and
 E, C, S_e = Material property constants.

These relations were compared with data from the literature, and with data involving radiation damage obtained by Reynolds. * A recommended design procedure involving the relations is presented. The technique was applied to the SADE 4B experiment with moderate success.

These cycling relations involve only mechanical damage imposed by cycling, with a modification for additional damage caused by radiation; they do not include any other potential performance limiting mechanisms, such as stress corrosion, which are normally factored into the over-all fuel design.

Comprelli, F.A., MacMillan, D.F., and Spalaris, C.N., "Materials for Superheat Fuel Sheaths - Relative Performance of Alloys in Superheated Steam Environments," GEAP-4351, July 1963.

SUMMARY

Details

1. The effect of irradiation in superheated steam on mechanical properties (tensile) were investigated for Types-304, -316, -347 stainless steel, AISI 406, RA330, and Incoloy. These alloys were exposed at 700 to 800°F for an accumulated exposure of 6×10^{19} nvt with neutron energies above 1 MeV. This level of exposure did not have a significant effect upon the tensile properties of the alloys studied, except for AISI 406. The ductility of 406 decreased from 11.4 percent to 4.4 percent elongation as a result of radiation exposure.
2. Stress coupons were exposed to 1050°F steam to investigate uniform scale buildup (corrosion) as a function of time. The alloys investigated were Incoloy, Inconel, Ni-O-Nel, Hastelloy-X, Hastelloy-N, Types-304, -310, -347 stainless steel, RA330, AISI 406, and Types 2-1/4-Croloy and 5-Croloy Ti. The 300-series stainless steel alloys exhibited average oxide buildup rates of less than 1×10^{-3} inch in 4000 hours of exposure. Type AISI 406 exhibited 1.20×10^{-3} inch in 4000 hours. The Croloys exhibited a scale buildup of 4×10^{-3} inch in 3000 hours, whereas the high-nickel alloys exhibited less than 1×10^{-4} inch for 4000 hours. All of the alloys obeyed the parabolic rate law, the rate of oxide buildup diminishing considerably after 1000 hours of exposure.

* Reynolds, M.B. "Slow Cycle Strain Fatigue in Thin Wall Tubing," Preliminary Report, GEAP-3983, July 1962.

3. Unstressed coupons in 1050° F steam exhibited lower corrosion oxide buildup for Inconel and AISI 406 than the corresponding coupons under stress. This has been attributed to the formation of intergranular oxide fingers associated with the stress-induced precipitation of grain boundary voids. All other alloys exhibited no differences in oxide buildup rate between stressed and unstressed conditions (within an experimental error). These alloys included Incoloy, Ni-O-Nel, and Hastelloy-X.
4. Coupons were exposed up to 1160° F with chloride salt surface contamination to investigate the effects of these salts upon the corrosion oxide buildup of the alloys. The alloys tested included Type-304 stainless steel, AISI 406, Inconel, Incoloy, Hastelloy-X, RA330, Hastelloy-N, and Ni-O-Nel. Three alloys, RA330, Hastelloy-N, and Inconel, failed under the stresses applied (0.1 percent creep in 1000 hours). The failures of Inconel were attributed primarily to the applied stresses rather than localized corrosive attack. Uniform scale thickness buildup was measured as follows:
 - a. Incoloy, Hastelloy-X, and Ni-O-Nel scale buildup was 30×10^{-5} inch in 1500 hours.
 - b. Inconel and AISI 406, less than 80×10^{-5} inch per 1500 hours.
 - c. Type-304 stainless steel, about 240×10^{-5} inch per 1500 hours.

The corrosion rate was parabolic as was the case without the salt contamination, the rates diminishing considerably after 1000 hours.

5. Scale adherence on Incoloy, Hastelloy-X, Inconel, and Ni-O-Nel was excellent with no observed exfoliations, either under stress or without stress to exposures up to 5195 hours. Salts had no effect on scale adherence of these alloys. Types-304, -347, and -406 stainless steel exhibited duplex scale with some porosity. The Croloy alloys exhibited triplex scales with considerable porosity at metal-oxide interface and throughout the oxide layers. Although scale exfoliation was not serious for the Croloys after 3000-hour exposures, the oxide thickness and porosity indicated future exfoliations would be extensive.
6. Physical metallurgy of the test alloys was studied as a function of exposure time at superheat temperatures. The alloys investigated were Incoloy, Ni-O-Nel, Hastelloy-X, AISI 406, Inconel, and Types-304, -310, and -347 stainless steels. Extensive grain boundary coarsening was observed in Hastelloy-X and Type-310 stainless steel. Some grain boundary coarsening was observed for Ni-O-Nel and Inconel, and very little for Incoloy. Carbide precipitation was observed in all of the alloys. The carbides in Incoloy were studied extensively using electron microscopy and x-ray diffraction techniques.

Many complex carbide morphologies were observed in the Incoloy coupons, but no adverse effects were detected in uniform corrosion resistance of the alloy. The general corrosion resistance of all the high-nickel alloys was unaffected by sensitization. Indications of gamma prime precipitate were seen in the Incoloy matrix. Inconel exhibited extensive carbide precipitation at grain and subgrain boundaries, but no matrix precipitate was observed. Type-310 stainless steel developed a continuous intergranular network at 1160° F which was identified as sigma phase. Hastelloy-X exhibited an extensive broadening of the grain boundaries due to formation and growth of a continuous network of M_6C carbides. The usual globular carbides were also detected along with indications of Laves phase Fe_2Mo . After 4000 hours of exposure,

Ni-O-Nel exhibited indications of a very fine matrix precipitate, whereas AISI 406 was found to be thermally stable after prolonged exposure to superheated steam.

7. The tensile properties of Incoloy, Inconel, Hastelloy-X, Ni-O-Nel, and AISI-406 were studied as a function of exposure to high-temperature steam, for periods up to 5195 hours. In general, changes in tensile properties were related to changes in the microstructure of these alloys. Hastelloy-X exhibited the most drastic change in ductility, due to intergranular and matrix precipitation of second phases. Incoloy exhibited an initial decrease in ductility corresponding to the suspected fine dispersion of matrix precipitates. However, after 2000 to 3000 hours, an improvement in the ductility was noted indicating a coarsening or "overaging" of the precipitate. Ni-O-Nel dropped from 48 to 40 percent elongation in 2000 hours. No matrix precipitate could be identified visually although a grain boundary growth was observed during this exposure period. The thermally stable alloys, Inconel and AISI 406, showed a slight change in tensile properties after prolonged exposure, probably due in Inconel to carbide precipitation at the grain boundaries.
8. Scale measurements after exposures at 1050 to 1160°F up to 5195 hours compare favorably with recently published results in nonnuclear steam environments. The ferritic alloys formed thick, uniform, porous scales with no subscale attack. The austenitic alloys developed thinner scales of varying thickness with many subscale penetrations "anchoring" the scale to the alloy matrix.

Reynolds, M. B., "Strain-Cycle Phenomena in Thin-Wall Tubing," GEAP-4462, January 1964.

SUMMARY

The service life of a nuclear reactor fuel element is related to the ability of its metallic cladding to accommodate severe local plastic strains caused by fuel expansion or external coolant pressure. A technique has been developed for subjecting samples of thin-wall tubing to cyclic plastic strains similar to those experienced in reactor service. By applying gas pressure, the thin tubular specimen is alternately expanded and contracted between rigid concentric mandrels until failure occurs as indicated by gas leakage. Relative cyclic strain range is determined by mandrel dimensions, specimen diameter, and specimen wall thickness. At larger strain ranges, deformation occurs by reverse bending rather than by simple tension and compression. These tests have been made at 1300°F on specimens of Hastelloy-X, Inconel-600, Incoloy-800, and Type-304 stainless steel; the last two in-reactor as well as in the laboratory. Specimen life in cycles to failure, N was found to be related to strain range $\Delta\epsilon_0$ computed from mandrel dimensions by $N^\alpha \Delta\epsilon_0 = C$, in which α varied from 0.6 to 1.35 and C from 0.14 to 3.3 depending on material. The effective strain range was found to be less than $\Delta\epsilon_0$ because of specimen wrinkling. The fatigue life of Incoloy-800 and Type-304 stainless steel was somewhat reduced by testing in fast neutron radiation. Specimen failure was by longitudinal cracking, with little evidence of gross plastic deformation, although there was evidence of deformation within the grains of the alloy in the vicinity of cracks.

Kirby, R. F., MacMillan, D. F., and Punches, J. R., "Fabrication of Fuel Cladding from Incoloy Alloy 800 - An Evaluation of Methods," GEAP-4557, April 1964.

SUMMARY

On the basis of its high-temperature physical and corrosion properties, Incoloy Alloy 800 was selected as a candidate for fuel cladding in nuclear superheat applications. At the time of its selection, there was little information or experience with Incoloy-800 in the production of thin-walled, small-diameter tubing suitable for nuclear fuel cladding. As a result, special purchasing efforts were required for the procurement of initial tubing used in fuel fabrication.

As-received welded and drawn tubing proved to be generally good, but showed some conditions which were undesirable, the major one being lack of complete recrystallization and homogenization of the weld zone. The possible effect of this condition upon the fuel performance was not immediately known; however, subsequent development work indicated that the nonhomogeneity of the weld could adversely affect its mechanical and corrosion properties in relation to the parent metal.

A development program was initiated to determine treatment sequences suitable for the fabrication of welded and drawn tubing with a fully recrystallized and homogenized weld structure. This was accomplished by butt welding lengths of Incoloy strip which were subsequently cold rolled and annealed to simulate tube fabrication steps. Requirements imposed on this work were that all processes developed must be amenable to normal production equipment and procedures used in commercial tube manufacturing. As a result of the experiments undertaken, a sequence of cold drawing and annealing steps was established suitable for recrystallization and homogenization of weld zones in welded-and-drawn tubing.

To obtain complete sequential chemical and metallurgical history of nuclear grade Incoloy-800 tubing, a 3000-pound ingot was purchased to the required chemical specifications and reduced to tubing through both seamless and welded and drawn routes. The management of the sequential metallurgical steps needed for tubing fabrication resulted in establishing a workable process through a series of reliable vendors. By providing these vendors with the appropriate basic technical support through the tubing production stages, a total yield of 85 percent was obtained in producing nuclear-grade tubing from the ingot. Much of the basic information needed to support vendor processing of the Incoloy ingot was obtained from the experiments described in this report. This activity resulted in not only enhancing our understanding of the properties of Incoloy-800, but illustrated that the practical steps needed for good quality tubing were within the capability of the vendors. The results from this program have undoubtedly helped in increasing the number of potential vendors and reducing the price of high-quality tubing needed for fuel element fabrication.

Spalaris, C. N., "Incoloy-800 for Nuclear Fuel Sheaths (A Monograph)," GEAP-4633, July 1964.

SUMMARY

Incoloy-800* was selected in 1961 as fuel cladding material for nuclear superheat reactors. At that time, the alloy was one of a group of possible candidates. Because of its lower neutron cross section

*ASTM designation: Nickel-Iron-Chromium Alloy. In the absence of a compact, convenient, generic designation, "Incoloy-800" is used throughout this report. "Incoloy," "Inconel," and "Ni-O-Nel" are trademarks of the International Nickel Company, Inc.

- relative to other high-nickel alloys - and because of its successful use in petrochemical and household appliance industries. Incoloy-800 was considered first in relation to the other alloys selected for study. Since 1961, fuel elements sheathed in Incoloy-800 performed satisfactorily in a superheat environment for 135 days without failure, whereas fuel elements clad in Type-304 stainless steel failed within 29 days. In other tests, Type-304 stainless steel failed after even shorter periods of time (10 to 15 days). In corrosion loops where superheated steam conditions were simulated, Incoloy-800 performed satisfactorily for 63 days without failure. Type-304 stainless steel failed within 2 to 3 days in similar experimental runs. Under stress conditions in a corrosion loop that included temperatures of 1150 F and initial chloride concentrations of 70 $\mu\text{gm}/\text{in.}^2$ on the specimen surface, Incoloy-800 performed satisfactorily up to 2000 hours of exposure.

Because of the encouraging results obtained to date, Incoloy-800 was chosen as the reference cladding material for fuel cladding applications in superheat reactor cores under study by G. E. - APED. It was decided to group the available data into one report, a monograph, to enable fuel technologists to locate the information needed for fuel design or fuel performance analysis. Although detailed information is included, the primary objective of this report is to summarize properties and behavior of Incoloy-800.

a. Availability

Incoloy-800 is available in all forms of wrought products such as tubing, bar stock, flat bars, sheet, pipe, and other standard products. Ingots of the material have been purchased and reduced to wrought products without difficulty. The alloy is described in ASTM Specifications B-407, B-408, and B-409, designated as "Nickel-Iron-Chromium Alloy."

b. Fabricability

Fabricability of Incoloy-800 is comparable to that of Type-304 stainless steel. Welded and drawn tubing has been readily produced in large quantities, and end plug welds for fuel elements have been made without difficulty. Impurities such as sulfur should be excluded from the material or cover atmosphere when welding.

c. Long-Term Stability at High Temperatures

The alloy is relatively stable at temperatures of interest to superheat fuel designs. Exposures at 1050 and 1150 F for periods of over 10,000 hours resulted in reduced ductility, but values lower than 20 to 25 percent elongation were not observed. The principal metallurgical changes detected by electron microscopy were carbide precipitation at grain and twin boundaries. The observed tensile property changes appeared to be related to the combined effect of a strengthened matrix and formation of a continuous intergranular carbide network.

d. Effects of Neutron Flux Upon Tensile Properties

Neutron exposures to 4×10^{20} nvt ($E_n \geq 1$ MeV) at 700 to 800 F do not affect the room-temperature tensile properties of Incoloy-800. When tested at 1100 F, irradiated tensile specimens showed no change in 0.2 percent offset yield strength but did show a drop in ultimate tensile strength and elongation. Minimum elongation values recorded were 23 percent.

e. Strain Cycle Properties

Fractures after strain cycling (slow frequencies, 30 min/cycle) showed plastic deformations. Irradiation decreased the number of cycles to failure at a given strain range by a factor of approximately one third.

f. Electron and Optical Microscopy

Carbide precipitates in various morphologies, which were observed after prolonged exposures at 1050 to 1150 F, have been identified as Cr_{23}C_6 .

g. Corrosion

Incoloy-800 was shown to remain intact in cyclic exposure with imposed stress in a loop operating with water and steam, in which 20 ppm oxygen, 2 to 3 ppm hydrogen, and 0.5 ppm chloride (iron chloride) were added to simulate reactor superheated steam conditions. Under similar conditions, Types-304, -347, -316, and -304 vacuum-melted grades of stainless steel developed localized cracking within 2 weeks. The tests with Incoloy-800 were carried out for 63 days without a sign of cracking.

Under steady-state corrosion at 1300 F in ex-reactor oxygenated steam, Incoloy-800 showed a corrosion rate which gave a predicted 3-year maximum weight loss of 6520 mg/dm^2 , or 3.3 mils metal loss (95 percent confidence limits). Under similar conditions, nonlinear extrapolation of data showed a metal loss of 1 mil per 3 years.

Coupons under stress and with chloride salts on their surface exposed to 1150°F in steam showed no cracking or localized corrosion attack.

h. Fuel Element Irradiation Experience

Exposures of 135 days at peak heat fluxes of $445,000 \text{ Btu/h-ft}^2$ with annular fuel elements demonstrated that Incoloy-800 was less susceptible to cracking failure than Type-304 stainless steel by at least a factor of 5 in the environment tested. No failures were evident in Incoloy-800 sheaths, whereas Type-304 stainless steel failed intergranularly within 12 to 15 days in a similar reactor environment.

The EVESR (ESADA-Vallecitos Experimental Superheat Reactor) core contains 150 annular geometry fuel elements clad with Incoloy-800. This represents about one-half of the entire reactor core. In addition, four advanced design fuel assemblies with rod clusters will be inserted in the EVESR by September 1964. Information on the performance of this fuel will be used to influence further use or improvements regarding Incoloy-800 as superheat fuel sheath material.

SUMMARY

The microstructures and associated tensile properties of the high-nickel alloys, Hastelloy-X, Inconel-625, Incoloy-800, Incoloy-825, and Inconel-600, have been studied as a function of exposure time at nuclear superheat temperatures, 1050 and 1150°F. Light and electron microscopy studies of the exposed coupons show that each alloy is subject to some degree of thermal instability, with resultant precipitation effects and tensile property changes. In general, the test data show that increasing exposure time at superheat temperatures resulted in a strengthening at room temperature of the solid solution alloys and a concurrent reduction in elongation. Examination of the alloy microstructures did not reveal any known precipitation mechanism to which these property changes could be attributed. However, each of the test alloys; except Inconel-600, has been shown to develop an equilibrium precipitate after extended exposure at slightly higher temperatures. This latent instability at higher temperatures suggests that, at the temperatures of exposure, pre-precipitation clustering or zone formation may occur which could lead to a strengthening of the matrix at room temperature. The reduction in elongation would, therefore, be directly related to the strengthening of the matrix, and the concurrent formation of a brittle precipitate network in the grain boundaries. Examination of the tensile fracture surfaces revealed a fracture of dual nature - partially transgranular (ductile) and partially intergranular (brittle). The dual fracture characteristic suggests that the fracture path followed the embrittled grain boundary only when the boundary was favorably oriented for crack propagation.

1. The room temperature tensile elongation of Hastelloy-X decreased to 12 percent after exposure at 1050 and 1150°F. This reduction in elongation was much more rapid after 1150°F exposure because of the more rapid growth of the brittle intergranular network of M_6C and a greater matrix strengthening effect associated with an unidentified finely-dispersed precipitate. The increase in strength after 1050°F exposure was not as substantial and was not associated with any observed precipitation.
2. Exposure at 1050 and 1150°F caused an increase of 75 percent in the strength of Inconel-625. The corresponding elongation values were reduced, but remained about 20 percent. This change in properties was attributed to the suspected coherent precipitation of a niobium-rich gamma prime. NbC was observed as an intergranular precipitate, but it did not form a continuous network.
3. Incoloy-800 was significantly affected by exposure at 1050 and 1150°F. The alloy readily formed an intergranular network of $M_{23}C_6$ at 1050 and 1150°F. The magnitude of strengthening observed after extended exposure was related to the amounts of aluminum and titanium in the alloy, both of which contribute to the suspected strengthening mechanism. After periods of exposure up to 10,200 hours at 1050°F and 5100 hours at 1150°F, room-temperature elongation values remained above 20 percent, with aluminum and titanium levels present in commercially available material.
4. Ni-O-Nel-825 exhibited significant increases in yield strength and reductions in elongation after exposure at 1050 and 1150°F. Post-exposure examination of the microstructure revealed

a continuous intergranular network of $M_{23}C_6$, but no matrix precipitation to which the property changes could be attributed. Studies of Ni-O-Nel-825 at slightly higher temperatures indicated that the alloy is susceptible to sigma formation. This information suggests that a pre-precipitation form of the sigma may be contributing to the strength increases measured at room temperature. The reduction in elongation was associated with brittle fracture along the carbide network in favorably-oriented grain boundaries.

5. Prolonged exposure at 1050 and 1150°F had relatively minor effects on the tensile properties of Inconel-600. The most noticeable changes were associated with the precipitation of large Cr_7C_3 particles in the grain boundaries and a dispersion of fine precipitates throughout the matrix, suspected to be Cr_3C_2 .

Comprelli, F. A., and Lewis, J. E., "Microstructural Evaluation of Superheat Cladding Materials," GEAP-4751, January 1965.

SUMMARY

Data are presented which describe the thermal response characteristics of selected superheat alloys. The alloys include Incoloy-800, Incoloy-801, Incoloy-804, Incoloy-825, Type-310 stainless steel, Hastelloy-X, Hastelloy-F, Inconel-625, Inconel-718, and a G. E. -Modified Incoloy-800. Each alloy was evaluated by monitoring the room temperature hardness after exposure for various time intervals at 1200 or 1400°F. Light microscopy was used to monitor microstructural changes. Precipitate phases were identified by x-ray diffraction. Relative amounts of elements in each phase were determined by x-ray fluorescence.

Beaudreau, B. C., Busboom, H. J., Comprelli, F. A., and Spalaris, C. N., "Tensile Properties of Fuel Clad Materials Irradiated in Superheat Environments," GEAP-4754, April 1965.

SUMMARY

The effect of irradiation upon the tensile properties of several alloys was studied as a function of accumulated neutron exposures and tensile test temperature. Alloys studied were Incoloy-800, Inconel-600, Hastelloy-X, AISi-406, and several 300-series stainless steels. Tensile tests were conducted at temperatures of 75, 1100, and 1300°F after accumulated neutron exposures of 0.6 to 4×10^{20} nvt ($E_n \geq 1$ MeV) in separate experiments. Both pre-cut coupons and tensile specimens, machined from irradiated fuel sheaths, were tested.

Comprelli, F. A., "High-Temperature Tensile Properties of Selected Alloys after Prolonged Thermal Exposure," GEAP-4794, February 1965.

SUMMARY

Several alloys considered for service at superheated steam conditions were tensile-tested at 75, 1100, and 1300°F. after exposures up to 5000 hours at temperatures ranging from 1000 to 1300°F.

Each alloy exhibited property changes at all test and exposure conditions. Changes in tensile properties due to aging correlated with the thermally-induced microstructural transformations of each alloy. The results can be summarized as follows:

Incoloy-825 exhibited increases in strength at all temperatures, with a maximum increase of approximately 80 percent after exposure at 1000 to 1200°F. Under all conditions, total elongation values remained above 15 percent.

Inconel-718 showed substantial increases, up to 100 percent, in strength at all test temperatures, particularly after exposure at 1000 to 1200°F. Prolonged exposure at 1300°F resulted in overaging and loss of strength at all tensile temperatures. Elongation values dropped as low as 6 percent.

Hastelloy-X showed a 50 percent increase in strength at each tensile temperature after exposure at 1300°F, with relatively minor reductions in elongation.

Hastelloy-F exhibited significant increases in strength (about 200 percent) after prolonged exposure at 1200 and 1300°F. The corresponding elongation values were reduced to approximately 5 percent.

All exposure conditions produced increases in strength of Inconel-625 at each tensile temperature. Exposure at 1200°F resulted in the largest increases, 100 to 200 percent, while the corresponding elongation values were reduced to 10 percent.

Prolonged exposure of René-62, particularly at 1200 and 1300°F, produced increases of 300 to 400 percent in strength both at room and elevated tensile temperatures. The corresponding elongation values dropped to essentially zero.

Increases in strength of 200 to 300 percent were measured for Incoloy-801 at all tensile temperatures after extended exposure. Elongation values were reduced, but remained above 15 percent.

The tensile properties of Incoloy-800 were relatively stable, with the largest strength increases, 50 to 60 percent, measured after exposure at 1000 and 1100°F. Elongation values were very stable, and remained above 20 percent.

Busboom, H. J., and Snyder, M. E., "Fabrication and Testing of Dissimilar Metal Joints for Thermal Spectrum Superheat Reactors," GEAP-4756, December 1964.

SUMMARY (This summary given previously in Task B)

The application of Zircaloy-2 for process tubes in thermal spectrum superheat reactors presents the problem of joining Zircaloy-2 to austenitic stainless steel alloys. A review of literature related to the problem of joining these metals revealed that the most promising bonding method was hot tandem extrusion.

Diffusion bonding experiments were considered, but were not carried out since literature data showed this method to be less favorable for large-diameter tubes (greater than 1.0 inch) because of problems in temperature control and accurate alignment of parts. The large intermetallic layers formed during bonding are unfavorable for ductile joints.

Hot extrusion experiments demonstrated that well-bonded, large-diameter tubular joints (1.75-inch o. d. $\times$ 0.065-inch wall) can be fabricated between Zircaloy-2 and Incoloy-800 or Type-304L stainless steel. It was also demonstrated that long Zircaloy-2 tubes can be extruded with these joints, thus allowing the fabrication of a complete full-size process tube, resulting in a substantial cost reduction.

Ultrasonic testing and metallography revealed that these joints were 100 percent bonded, and that they would have adequate strength and elastic stability. The joints were found to have adequate corrosion resistance in a simulated superheat reactor environment where steam and boiling water temperatures, pressures, and flow conditions were maintained for 3472 hours. Measurements made during the corrosion run indicated that the stagnant steam gap which was maintained between the process tube and liner provided excellent insulation to loss of heat from the superheat to boiling side of the process tube. Thermal cycling between 100 to 600°F did not appear to be detrimental to joint integrity. The elastic mechanical behavior of these joints was similar to that of the base metals. The plastic mechanical properties were found to be limited by the Zircaloy-2 rather than the joint material.

Reproducibility of fabrication quality from joint to joint was found to be excellent.

Busboom, H. J., Francke, P., and Beaudreau, B. C., "Neutron Radiation Damage Studies in High Nickel Alloy," GEAP-4812.

SUMMARY

In these irradiation programs, alloys which have received heavy attention for nuclear superheat cladding were irradiated to fast neutron doses up to 3.5×10^{20} (SC-8) and 2.0×10^{21} nvt (SC-9) (energies above 1 MeV). Irradiation has been carried out at temperatures ranging from 1000 to 1300°F. For both programs, out-of-pile aging studies were conducted to establish the influence of temperature, and thus separate ageing from radiation-induced effects upon the mechanical properties. Only the tensile coupons irradiated at 1300°F were tested. Selected tensile test temperatures ranged from 78 to 1300°F. For better comparison of results with those obtained from earlier experiments, the strain rate was held at 0.02 in./min.

Busboom, H. J., "Fast Neutron Damage Studies in Nickel Alloy," GEAP-4985.

SUMMARY

Recent studies have shown that the elevated temperature ductility of high-nickel alloys is greatly reduced by exposure to both thermal and fast neutron flux. This report describes the effect of a fast neutron flux, without a thermal neutron flux, on the elevated temperature behavior of high-nickel alloys. Tensile-type test specimens of Incoloy-800, Inconel-625, and Hastelloy-X were irradiated in the EBR-II for 8 days at 1000°F to a total integrated exposure of 3.5×10^{20} nvt, >1 MeV. A comparison was made of pre- and post-irradiation tensile properties at 78, 1100, and 1300°F.

Whereas the room temperature ductility of Incoloy-800 was only slightly affected by the fast neutron exposure, the 1100° F post-irradiation ductility was substantially reduced from 33 percent to 22 percent total elongation. This reduction in ductility was more severe as the test temperature was increased, from 63 percent to 13 percent total elongation at 1300° F.

A more severe reduction in 1300° F ductility was found for Inconel-625, from 36 percent to 5.8 percent total elongation, and for Hastelloy-X, from 50 percent to 8.4 percent total elongation. However, a large increase in post-irradiation yield stress was also observed in these two alloys but not for Incoloy-800, indicating that accelerated precipitation had occurred due to the high fast neutron flux of EBR-II, 4×10^{14} nvt, >1.34 MeV. This degree of precipitation has not been observed for similar exposure conditions without a neutron flux. The lower ductility found in Inconel-625 and Hastelloy-X was due to a combined effect of precipitation hardening and fast neutron damage.

An increase in the 1300° F ductility of irradiated Incoloy-800 from 13 percent to 26 percent total elongation was observed by increasing the strain rate from 0.02 min^{-1} to 2.0 min^{-1} .

Metallographic examination of both fractured irradiated and unirradiated specimens revealed that the decrease in elevated temperature ductility was related to a change in fracture mode from ductile transgranular to brittle intergranular fracture. A large amount of precipitation was readily observed in the irradiated Inconel-625 and Hastelloy-X specimens.

These results show that fast neutron exposure alone can greatly reduce the ductility of high nickel alloys, and can also accelerate precipitation in alloys which undergo thermally-induced aging at elevated temperatures. The strain rate dependence on the total elongation indicates that the mode of deformation plays an important role in elevated temperature ductility.

Hill, H. W., Lewis, J. E., and Grossman, L. N., "Sintering Studies and Phase Characterization in the System Eu_2O_3 - UO_2 ," GEAP-5008, January 1966.

SUMMARY

The sintering behavior of Eu_2O_3 - UO_2 compositions containing 0 to 10 weight percent Eu_2O_3 was studied in hydrogen atmospheres at 1550, 1650, and 1750° C. It was found that Eu_2O_3 inhibits densification in the binary oxide system; inhibition is most pronounced at the lower temperatures. At 1750° C, densities above 95 percent of theoretical were attained. X-ray fluorescence techniques were employed for the analysis of compositional uniformity in the sintered specimens. The sintered samples were characterized for phase structure by x-ray diffraction techniques; only the fluorite phase was detected.

Phase equilibria in the Eu_2O_3 - UO_2 system was determined by high temperature thermal analysis and x-ray diffraction on specimens containing between 0 and 56.6 weight percent Eu_2O_3 . A partial phase diagram is given. The lattice parameter of the fluorite phase decreases linearly from 5.4707 Å for pure UO_2 to 5.400 Å for 57.5 mole percent $\text{EuO}_{1.5}$.

5.0 TASK D - EXPERIMENTAL PHYSICS

5.1 THE TASK

The work on this task was concerned with the design, experiments, and analysis of nuclear superheat critical experiments.

5.2 DOCUMENTATION

Part of the work reported in GEAP-3591, "Manufacture of the Adhesive Bonded AEC Superheat Critical Fuel," listed under Task B was done on this task.

Other reports under this task are:

Antunez, H. M., "Conversion Ratio Measurement in Uranium-Fueled, Light-Water-Moderated, Superheater Lattices," GEAP-3863, March 1962.

SUMMARY

The conversion ratio has been measured in uranium-fueled, light-water-moderated, superheater lattices. A new method applicable also to other types of reactors, was developed for use in the measurement. To make sure that the fuel samples used were representative of the fuel, full-size pellets of the irradiated fuel were dissolved and gamma counted. Neither geometrical imperfections nor neutron spectrum alterations affect the measurement. The method also has these advantages: 1.) chemical separations are not required, 2.) irradiation under cadmium is not required, and 3.) irradiations amounting to less than 10^{12} n/cm² are enough. Special care was taken to prevent the occurrence of systematic errors, for which there are several possible sources in similar methods. The precision was improved up to ± 0.6 percent (70 percent confidence intervals). The basic idea of the method is that two separate gamma-counting channels are defined so that the corrected count rate in the first channel is proportional to the number of plutonium atoms produced in an ordinary fuel pellet, and that the corrected count rate in the second channel is proportional to the number of fissions which occurred in the same fuel pellet. The relative counting efficiencies of the two channels are normalized by irradiating samples of uranium of different enrichments in a thermal flux.

Peterson, G. T., and Warzek, F. G., "AEC Superheat Criticals - A comparison of Experiment and Theory of Uniform Lattices," GEAP-3882, January 1962.

SUMMARY

Measurements were conducted with uniformly-spaced arrays of superheat fuel (no controls or water gaps) to obtain a direct measurement of the physical parameters within the fuel element cell. The element pitches of 1.800, 1.900, and 2.000 inches were utilized to cover

a water-to-fuel volume ratio range of 1.47 to 3.67. The actual measurements were performed at the Vallecitos Atomic Laboratory in two phases.

Phase I

1. Critical Size
2. $(\partial\rho/\partial H)$ versus H (ρ = reactivity, H = water height)
3. Void Coefficient
4. Temperature Coefficient
5. Flux Distribution

Phase II

6. Thermal Utilization
7. Conversion Ratio

Phase I of the program covered the period October 1960, through January 1961. Phase II of the program covered the period May 1961, through June 1961. The primary reason for delaying the Phase II measurements was to provide the time necessary to develop a new and more accurate method of conversion ratio measurement, which was then applied to the superheat lattices.

The results of the Phase I measurements were compared to the predictions of an engineering design model that had demonstrated reasonable accuracy in treating BWR and PWR lattices. Relatively large discrepancies in all three parameters (reactivity, void coefficient, and temperature coefficients) exist. This model led to reactivity overpredictions of 2 to 5 percent $\Delta k/k$, and to more negative void and temperature coefficients than were measured.

Possible areas of uncertainty in the engineering design model included:

1. Thermal utilization,
2. Resonance escape, and
3. Neutron leakage.

Of these three, only thermal utilization appeared to offer a consistent explanation of the reactivity and reactivity coefficient discrepancies. Furthermore, it appeared that there was valid uncertainty as to the application of the zero-dimensional thermal spectrum calculation present in the model.

Based on the assumption that the thermal spectrum treatment in the thermal utilization calculation was the prime source of error, a semi-empirical correction, normalized to the reactivity error in one case, was introduced to modify the calculational model. The largest reactivity discrepancy became 0.3 percent $\Delta k/k$ rather than 5 percent, and both the void and temperature coefficient discrepancies were markedly reduced. Part of the reactivity agreement is a result of normalizing the correction to give zero reactivity error in the 1.800-inch pitch flooded case, but it is quite significant that all of the other discrepancies dropped simultaneously and the previous trend toward larger error when the element pitch was increased or the other coolant passage flooded, was removed.

To confirm the corrected thermal utilization as calculated by the modified engineering design model, calculations utilizing the SLO P-1 code were performed. SLO P-1 is a multi-thermal group code which permits scattering from each energy group to every other energy group, and solves the neutron transport equation by the P-1 approximation. Shortly after the SLO P-1 calculations had been completed, the Phase II thermal utilization measurements were made. Excellent agreement was attained between the measured thermal utilization and both the SLO P-1 and modified engineering design model calculations. The engineering design model, however, overpredicted thermal utilization by as much as 4.3 percent $\Delta f/f$, verifying the assumption that thermal utilization was the prime cause of the reactivity discrepancies.

In addition to the check of thermal utilization, resonance capture by U-238 received careful study. Calculations of the resonance escape probability utilizing the Monte Carlo code NYU-REP were performed. NYU-REP accurately treats the spatial capture in each of 55 resolved U-238 resonances. After a correction for the unresolved resonance contribution was made, the resonance escape probabilities as determined by Monte Carlo were compared to the predictions of the engineering design models. The agreement was excellent. The largest discrepancy was only $\pm 0.003 \Delta p/p$ or $\pm 0.03 \Delta (1-p)/(1-p)$, which indicates that resonance escape for U-238 is treated accurately by the engineering model for this fuel geometry.

Although the resonance escape probability for U-238 was not measured directly, the conversion ratio was. Since the resonance escape probability is directly related to the conversion ratio, an indirect measurement of the resonance escape probability was made. Good agreement was attained in all cases, with the largest discrepancy only 3 percent ($\Delta CR/CR$), which would translate into less than 0.5 percent error in the resonance escape probability.

6.0 TASK E - COOLANT CHEMISTRY

6.1 GENERAL

The work on this task was originally divided into two categories: 1. Out-of-Pile and 2. In-Pile. The in-pile work was concerned with the chemistry associated with the operation of the SADE and Expanded Superheat Advanced Experiment (ESADE) loops in the VDWR. This work ended with the shutdown of the VBWR in December 1963.

Out-of-pile work continued throughout the life of the project to evaluate the corrosion characteristics of candidate nuclear superheat materials.

6.2 WORK DURING THE LAST QUARTER

6.2.1 Corrosion with Heat Transfer - G. G. Gaul, W. L. Pearl

1. The planned testing of Inconel-625 under heat transfer with a calculated uniform metal temperature of 1500°F has been completed. The results are tabulated in Table 6-1 and plotted in Figure 6-1.

TABLE 6-1

CORROSION RESULTS
Exposure Temperature 1500°F

<u>Exposure Time (hours)</u>	<u>Weight Change (mg/dm²)</u>	<u>Total Metal Corroded (mg/dm²)</u>	<u>Calculated Metal-to- System Loss (mg/dm²)</u>	<u>Percent Loss</u>
21	0	53	15	28
314	-60	191	97	50.8
1003	-106	274	153	55.8
2688	-228	504	305	60.5

The above data indicate a constant corrosion rate at 1500°F under heat transfer of 95 mg/dm²/mo, with a metal-to-system loss rate of 64 mg/dm²/mo. The corresponding intercept of the linear rate law line to time zero is 145 mg/dm² for the corrosion and 65 mg/dm² for the loss to system.

A correction factor of 0.25 mil must be added to the calculated total corrosion to allow for the damaged metal not removed by descaling. This type of damage is shown on a descaled specimen (Figure 6-2). The degree of damage appears to be the same at 1003 hours as at 2688 hours, and appears to be characteristic of Inconel-625 above 1100°F.

A 2.1-mil depth of corroded metal is calculated for a 3-year life projection of these data. This includes no provision for tolerance limits. The equivalent of 1.2 mils depth of metal can be expected to be lost to the system as oxide during a 3-year exposure of the heat transfer surface

Figure 6-1. Corrosion at Metal Temperature - 1500°F

Descaled

2000×

(reduced to 77 percent for reproduction)

Figure 6-2. Inconel-625 Surface Heat Transfer Exposure for 2688 hours at 1500°F

at 1500°F to superheat steam. The oxide remaining after a 3-year exposure would be equivalent to 0.6 mil of metal. Assuming a scale density of 5.5, this oxide layer would be 1.0 mil thick. It is not believed that this thickness would raise the temperature of the reaction interface (metal-scale) significantly to influence the corrosion rates.

The compositionally disturbed layers developed are shown in Figure 6-3. No apparent decrease of the layer thickness with time, as noted on layers developed up to 1300°F (GEAP-4932), was noted on the material exposed at 1500°F.

2. The Inconel-625 heat transfer corrosion data obtained to date (1500°F data reported in item I. above) are summarized in Table 6-2.

TABLE 6-2

INCONEL-625 CORROSION WITH HEAT TRANSFER

Calculated Metal Temperature F	Exposure Time (hours)	Measured Weight Change (mg/dm ²)	Descaled Weight Loss (mg/dm ²)	Calculated Metal-to-System Loss (mg/dm ²)
800 to 900	652	-7	48	19
900 to 1100	652	-7	50	19
	3295	-20	104	43
1100 to 1300	618	-34	97	51
1280 to 1410	311	-41	135	68
	2052	-163	285	197
	3685	-271	471	325

Although adequate data are not yet available for accurate extrapolation, the above data can be used for establishing an intermittent value for design purposes. The 1280 to 1410°F exposures can be extrapolated for a 3-year exposure to give a 1.6- to 1.8-mil depth of corroded metal. Although these numbers contain no confidence limits, they may be adequately conservative for use as a 1300°F design number. The corresponding metal lost to system would be about 1 mil.

Metallographic examination of the sheath exposed at a metal temperature at 1280 to 1410°F for 3685 hours was completed. The oxide at 1300°F (Figure 6-4a) and at 1400°F (Figure 6-4b) continue to be thin with similar penetrations to those noted on the 1500°F specimens (Figure 6-2). The compositionally disturbed layer at 1300°F (Figure 6-5a) is similar to that seen previously after 2052 hours of exposure (Figure 6-1, GEAP-4932), and did not "disappear" as was noted previously (GEAP-4932). The microstructure at 1400°F (Figure 6-5b) shows a change of structure similar to that noted in shorter times at 1500°F (Figure 6-3).

6.2.2 Microprobe Studies - E. G. Brush

The corrosion of iron and nickel base alloys in high-temperature steam is a partially selective process. As discussed previously (GEAP-4760), this behavior results from the fact that, at least initially, chromium oxidizes at greater rates, while iron and nickel oxidize at lesser rates, than those to be expected

a. 21 hours

b. 314 hours

c. 2688 hours

Etchant: $\text{HCl-H}_2\text{O}_2$

2000×

Figure 6-3. Inconel-625 Microstructure Heat Transfer Exposure at 1500°F

a. 1300°F

b. 1400°F

With Scale

2000×

(reduced to 77 percent for reproduction)

Figure 6-4. Inconel-625 Oxide Heat Transfer Exposure for 3685 hours

a. 1300°F

b. 1400°F

Etchant: $\text{HCl-H}_2\text{O}_2$ With Scale
(reduced to 77 percent for reproduction)

2000×

Figure 6-5. Inconel-625 Microstructure Heat Transfer Exposure for 3685 hours

from the bulk alloy composition. The process is accompanied by a compositionally-changed alloy phase "layer," extending into the alloy from the oxide-alloy interface. It contains less chromium and more iron and nickel than the bulk alloy. The selective corrosion process can significantly affect the worth of a given alloy for superheat cladding in two important areas. First, it can alter the extent of contamination due to oxide-to-system loss over that expected if the composition of the lost oxide were stoichiometric. Such alteration may or may not be in a beneficial direction. Secondly, and probably detrimentally, the alloy phase "layer" may have undesirable mechanical properties. The selective corrosion process is being studied by the electron microprobe. This instrument is capable of providing composition measurements at 1-micron ($1 \mu = 1 \times 10^{-5}$ dm) intervals. The quantitative interpretation of the probe results has been hampered by: 1) the lack of appropriate descriptive models that take account of corrosion-diffusion couples where all alloying elements oxidize to varying degrees, and where the corrosion rate laws that govern such oxidation undergo transformation; and 2) the lack of diffusivity data in ternary systems. A descriptive model has been developed (GEAP-4490 and GEAP-4832) which considers the corrosion of chromium, for instance, as composed of two components: a selective fraction with rate $j_{Cr(s)}$ and a nonselective fraction with rate $j_{Cr(ns)}$. The total amount of chromium that oxidizes is then

$$J_{Cr} = j_{Cr(s)} + j_{Cr(ns)}.$$

The nonselective rate is given by

$$j_{Cr(ns)} = C_{Cr(o)} d\Delta W/dt,$$

where $C_{Cr(o)}$ is the alloy phase chromium concentration at the oxide-alloy interface, and $d\Delta W/dt$ is the alloy corrosion rate. The selective rate $j_{Cr(s)}$ is the rate at which excess chromium crosses the boundary $x = d\Delta W/\rho dt$, Figure 6-6, and oxidizes along with the mass of chromium ($= C_{Cr(o)} d\Delta W/dt$) contained in the oxidizing layer of thickness $d\Delta W/\rho dt$. This excess chromium mass is defined as a continuous diffusion source operative in the plane $x = d\Delta W/\rho dt$. For the values of $d\Delta W/dt$ that exist in practical superheat steam service, this plane is essentially coincident with the plane $x = 0$, the oxide-alloy interface. The diffusion equation which describes the concentration of a diffusing species with respect to the moving oxide-alloy interface is

$$\partial c / \partial t = D \partial^2 C / \partial x^2 + (d\Delta W / \rho dt) \partial C / \partial x.$$

The alloy phase behavior of chromium is then given by

1. During the nonlinear transient ($n \neq 1$):

$$C_{Cr\infty} - C_{Cr} = nk_{Cr(s)(n\ell)} / 2\rho \sqrt{\pi D} \int_0^t t^{n-\frac{3}{2}} \exp \left[- \left(\frac{x}{2\sqrt{Dt}} + \frac{Kn\ell t^{n-\frac{1}{2}}}{2\rho \sqrt{D}} \right)^2 \right] dt,$$

in which

$$d\Delta W_{n\ell} / dt = nK_{n\ell} t^{n-1} \text{ and}$$

$$j_{Cr(s)(n\ell)} = n k_{Cr(s)(n\ell)} t^{n-1}$$

2. During the linear steady state

$$C_{Cr\infty} - C_{Cr} = j_{Cr(s)(\ell)} / 2K_{\ell} \sqrt{\pi D} \int_0^t t^{-\frac{3}{2}} \exp \left[- \left(\frac{x}{2\sqrt{Dt}} + \frac{K_{\ell} t}{2\rho \sqrt{D}} \right)^2 \right] dt,$$

Figure 6-6. Schematic Representation of Partially Selective Oxidation

in which

$$d\Delta W_{\ell} / dt = K_{\ell} = \text{Constant and}$$

$$j_{Cr(s)(\ell)} = \text{Constant.}$$

The solution to the steady-state profile description is

$$C_{Cr\infty} - C_{Cr} = j_{Cr(s)(\ell)} / 2K_{\ell} \left\{ \text{erfc}(u_{\ell}) + \exp\left(-\frac{K_{\ell} x}{\rho D}\right) \text{erfc}(v_{\ell}) \right\},$$

where

$$u_{\ell} = (x/2\sqrt{Dt}) + (K_{\ell} t/2\rho\sqrt{D}) \text{ and}$$

$$v_{\ell} = (x/2\sqrt{Dt}) - (K_{\ell} t/2\rho\sqrt{D}).$$

A solution to the nonlinear transient region can be found if it is assumed

1. that

$$j_{Cr(s)(n\ell)} = \alpha d\Delta W_{n\ell} / dt;$$

i.e., the selective chromium rate is some fraction X (which is probably a fraction of t) of the alloy rate and that

2. An average alloy rate in the nonlinear transient can be described by, Figure 6-7,

$$\text{av}(d\Delta W_{n\ell} / dt) = K_{n\ell} t_c^{n-1}.$$

The nonlinear transient profile description will then be

$$C_{Cr\infty} - C_{Cr} = \bar{\alpha}/4 \left\{ \exp\left(-\frac{K_{n\ell} t_c^{n-1}}{\rho D}\right) \text{erfc}(v_{n\ell}) - \text{erfc}(u_{n\ell}) \right\},$$

where

$$u_{n\ell} = (x/2\sqrt{Dt}) + (K_{n\ell} t_c^{n-1} \sqrt{t}/2\rho\sqrt{D}) \text{ and}$$

$$v_{n\ell} = (x/2\sqrt{Dt}) - (K_{n\ell} t_c^{n-1} \sqrt{t}/2\rho\sqrt{D}),$$

and where

$$\alpha = (1 + C_{Cr\infty}) / 2.$$

Microprobe traces exist for the specimens listed in Table 6-3. An additional problem in their analysis involves their time location with regard to the rate law, nonlinear or linear, that was governing at the time at which they were taken. Earlier discussions had advanced the concept that parabolic (GEAP-4760) or cubic (GEAP-4932) nonlinear transient laws might apply. However, if the weight change data (GEAP-4760) are carefully examined with respect to their magnitude and the magnitude of the linear rate constant, it can be shown that the nonlinear time dependence " n " in $d\Delta W_{n\ell}/dt = nK_{n\ell} t^{n-1}$ can numerically approach values down to 1/10, depending upon how soon the onset of linearity is fixed. In view of the recent discussions of Warzee [J. Electrochem. Soc. 112, 670-674, (1965)], of Caplan and Cohen [J. Electrochem. Soc. 112, 471-477, (1965)], and of the expected diffusivity values (GEAP-4832), it appears as though the time

$$\text{ave } (d\Delta W_{n\ell}/dt) = \frac{1}{t_c} \int_0^{t_c} n K_{n\ell} t^{n-1} dt = K_{n\ell} t_c^{n-1}$$

Figure 6-7. Calculation of Average Corrosion Rate during Nonlinear Transient from $t = 0$ to $t = t_c$

TABLE 6-3

ELECTRON MICROPROBE TRACES TAKEN ON
SPECIMENS EXPOSED TO SUPERHEATED STEAM

Specimen No.	Material	Metal Temperature (°F)	Total Exposure Time (h)
632870	Type-304 stainless steel	1050	5500
61444	Type-304 stainless steel	1300*	2465
621804	Hastelloy-X	1050	1000
632669	Hastelloy-X	1050	5116
622236	Hastelloy-X	1300*	990
621803	Inconel-600	1050	1000
632665	Inconel-600	1050	5116
621921	Inconel-600	1100*	1000
632871	Inconel-600	1100*	1895
621922	Inconel-600	1300*	1000
633568	Inconel-625	1150	977
644481	Inconel-625	1150	2946
650251	Inconel-625	1300*	2105
650250	Inconel-625	1350*	2105
650253	Inconel-625	1400*	2105
650503	Inconel-625	1400*	2052
650502	Inconel-625	1500*	1003
621802	Incoloy-800	1050	1000
632667	Incoloy-800	1050	5116
622077	Incoloy-800	1100*	965
633771	Incoloy-800	1100*	4078
622076	Incoloy-800	1300*	965
633772	Incoloy-800	1300*	4078
644144	Incoloy-800	1400*	995

*Oxide-metal interface temperature of heat transfer exposure.

exponent "n" might well have a value of $n = 1/4$. Taking this into account with the trace data of the specimens in Table 6-3, and knowing the average linear rate law period corrosion rate constants, K_l (fairly accurate, based upon developed procedures), trial and error estimation using:

1. The values of C_{Cr} at $x = 0$ to determine D from the nonlinear transient,

$$C_{Cr\infty} - C_{Cr(0)} = (\bar{\alpha}/2) \operatorname{erf} \left(K_{nl} t_C^{n-\frac{1}{2}} / 2\rho\sqrt{D} \right),$$

where t_C = the duration of the nonlinear transient, and assuming that the traces were taken during the linear rate law period,

2. Either slope measurements at $x = 0$ (where in both 2. and 3. t_ℓ = exposure time during linear rate law period)

where

$$\partial C_{Cr}/\partial x = (j_{Cr(s)(\ell)}/2 K_\ell) \left\{ (2/\sqrt{\pi D t_\ell}) \exp(-K_\ell^2 t_\ell / 4 \rho^2 D) + K_\ell / \rho D \operatorname{erfc}(-K_\ell \sqrt{t_\ell} / 2 \rho \sqrt{D}) \right\},$$

or

3. Selective mass as measured by the area under the chromium profile

$$\Delta W_\ell = j_{Cr(s)(\ell)}/2 K_\ell \left\{ (2\sqrt{D t_\ell}/\pi) \exp(-K_\ell^2 t_\ell / 4 \rho^2 D) + (2\rho D/K_\ell) \operatorname{erf}(K_\ell \sqrt{t_\ell} / 2 \rho \sqrt{D}) - (K_\ell t_\ell / \rho) \operatorname{erfc}(K_\ell \sqrt{t_\ell} / 2 \rho \sqrt{D}) \right\},$$

will help establish the value of D , as well as characterizing the difficulty associated with x-ray source area effects (GEAP-4490). Once D is established, the magnitude of the alloy layer compositional change can be assessed, for it will cease to grow when

$$K_\ell \sqrt{t_e} / 2 \rho \sqrt{D} \cong 1.8,$$

and insertion of this value of t_e and of D in the linear steady-state profile description will establish the distance X_e at which

$$C_{Cr} \cong C_{Cr\infty}.$$

In regards to the evaluation of loss to system, values of D are not so important as the values of " n " and of t_c . For, as can be ascertained from the linear rate law region profile description the sum of $j_{Cr(s)(\ell)} + j_{Cr(ns)(\ell)} = J_{Cr}(\ell)$, will equal the stoichiometric corrosion rate for chromium, $C_{Cr\infty} d\Delta W_\ell / dt$ (or $C_{Cr\infty} K_\ell$). Stoichiometry will also exist for iron and for nickel. Thus, during the linear steady state, the model predicts that all alloying elements will oxidize in direct proportion to their bulk alloy composition. If the nonlinear transient behavior can therefore be characterized, it is an easy task to translate oxide microprobe traces taken during the linear steady state into alloying element loss to system.

6.2.3 General Corrosion - W. L. Pearl

1. A study is being made to look at the effect on the oxide and disturbed metallic surface layer of Incoloy-800 resulting from exposure at 1150°F after pre-exposure at 932°F. The weight gain and weight loss measurements after 3000 hours at 1150°F have been made and are reported in Table 6-4 with the data reported previously.

The improved condition of the pre-exposed specimens has remained, although there are not enough data to establish whether the effect is merely a variation in the initial corrosion (ΔW_0) or the long-term corrosion rate.

TABLE 6-4

**CORROSION OF PRESCALED INCOLOY-800 AFTER EXPOSURE
IN 1150°F SUPERHEATED STEAM**

932°F Pre-Exposure Time (h)	Weight Gain (mg/dm ²)				Metal Loss (mg/dm ²)		
	Pre-Exposure	1036 h	2045 h	3042 h	Pre-Exposure	1036 h	3042 h
0	-	89	103	104	-	255	329
388	18	50	62	63	77	160	195
907	26	61	60	63	103	191	228
1906	35	58	71	72	127	188	237
2911	33	60	64	65	132	204	-

2. Isothermal Incoloy-800 and Inconel-625 specimens, oxidized in superheated steam through different times from 333 hours to 10,000 hours at 1150°F, have been examined for oxide scale formation, mechanism, and structure.

Parts of all specimens were mounted for metallographic preparation and polished, so that of each specimen two representative areas could be photographed at 2000- $\times$ magnification.

a. Incoloy-800

For reasons of comparison, a composite of seven metallographic pictures of the 9898-hour exposure (Figure 6-8) was made, representing a whole length of about 400- μ oxide scale.

- (1) After 333 hours, the oxide scale on Incoloy-800 specimens consists of two different layers:
 - (a) An upper light one which is not completely protective and seems to develop from spreading oxide nuclei.
 - (b) A lower dark one, penetrating the metallic structure along the grain boundaries and in special grains, including metallic particles.
- (2) After more than 1000 hours, all oxide scales have three different layers:
 - (a) An upper light one of homogeneous structure, but of local different thickness.
 - (b) A middle grey one of homogeneous structure and constant thickness.
 - (c) A lower dark one, rich in uncompletely oxidized small metal particles, which are penetrating the structure of the metal phase near grain boundaries and in grains of favored crystalline orientation.

Unetched

2000x

Figure 6-8. Oxide Formed on Incoloy-800 after 9898 hours of Exposure to 1150°F Superheated Steam

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- (3) Starting from the 3000-hour time point, the scale does not lose its protective character, and includes metallic particles or grains of noticeable size only in the lower part of the scale.
- (4) The total oxide scale thickness and the thickness of these three different oxide phases were measured planimetrically. The measurements of two representative pictures of each specimen are shown in Table 6-5.
 - (a) During the first 3000 hours, the scale formation is relatively fast. It slows down to a nearly constant value of scale thickness during the following 7000 hours. The upper layer seems to lose at least the amount of oxide continuously produced by the oxidation reaction, perhaps more than this because of the decreasing thickness. The slow growth of the middle scale can be considered as representative for scale growth without interference under equilibrium conditions between the middle and outer layers.

It is hoped that these studies will increase our capability of extrapolating our data beyond the 10,000-hour test period.

TABLE 6-5
SCALE THICKNESS OF INCOLOY-800 AFTER EXPOSURE
IN 1150°F SUPERHEATED STEAM

Exposure Time (h)	Total Oxide Scale Thickness (μ)	Outer Scale Thickness (μ)	Middle Scale Thickness (μ)	Inner Scale Thickness (μ)
333	2.9 3.3	-----Measurements Not Possible-----		
2090	11.8 11.6	1.7 2.0	4.4 3.8	5.7 5.9
3067	14.0 12.8	2.5 1.7	4.7 5.2	6.8 5.9
4016	13.6 12.3	2.6 1.5	5.3 4.7	5.7 6.1
5036	14.1 13.6	2.0 1.7	5.4 5.1	6.8 6.8
9898	14.0 14.0	1.4 1.5	5.9 5.7	6.6 6.9

b. Inconel-625

The oxide thickness planimeter measurements on the Inconel-625 oxides formed in 1150°F superheated steam are given in Table 6-6. A slowly growing oxide scale is indicated. Despite the gaps and many little included metallic particles in the oxide scale, the oxide maintains its protective character. The oxidation process forms a thin, uniform layer with some suboxidation and bulk-penetration noted. Losses to the system appear to be

gradual, with a resultant slowly growing scale thickness. A high magnification of the thin oxide after 9772 hours of exposure is shown on Figure 6-9.

TABLE 6-6

SCALE AND DISTURBED LAYER THICKNESS OF
INCONEL-625 AFTER EXPOSURE IN 1150°F
SUPERHEATED STEAM

Exposure Time (h)	Descaled Weight Loss (mg/dm ²)	Average Oxide Scale Thickness (μ)	Average Disturbed Layer Thickness (μ)
333	*	1.5/2.2	8.3
1926	130		8.3
2749	*	2.2/2.6/2.3	7.1
2946	98		11.4
4011	118	2.9/3.4	13.0/13.3
5691	105	2.5/2.5	13.8/14.9/11.9/15.9
9772	146	3.5/3.4/4.1	2.5

*Different starting time of exposures. No descaling carried out.

Carefully etched specimens of Inconel-625 were photographed at 2000- $\times$. A planimeter was used to measure the average thickness of the compositionally disturbed metallic layer next to the oxide scale. The results are presented in Table 6-6. As seen optically, the layer thickness goes through a maximum after about 5000 hours of exposure (Figure 6-10a) and is nearly gone by 10,000 hours (Figure 6-10b). A similar phenomenon was noted with the heat transfer specimens in shorter times at higher temperatures. Further work is required to determine if a change in composition is occurring that is not revealed by the etching reaction.

3. The remaining alloys being tested in superheated steam for 10,000 hours were terminated during the period. The 1050°F exposure data available are compared with previous data in Table 6-7, and the 1150°F exposure data in Table 6-8.

Using the data in Table 6-8 the corrosion rates reported in the last Quarterly (GEAP-4932) have been updated. Assuming a linear rate after 1000 hours, the corrosion rates (K_p) and corresponding intercepts (ΔW_0) obtained by the least squares technique from a computer are shown in Table 6-9.

The assumption of a linear rate after 1000 hours appears reasonable for the Hastelloy-F and Incoloy-804 taken at both temperatures of exposure and for the Inconel-718 and René-62 exposed at 1050°F. As noted previously (GEAP-4861), however, the Inconel-718 and René-62 data at 1150°F are somewhat suspect. A sharp increase can be noted in the descaled weight loss data after about 4000 hours (Table 6-9). Similarly a large increase in weight gain prior to descaling is noted.

Unetched

2000×

(reduced to 77 percent for reproduction)

Figure 6-9. Oxide Formed on Inconel-625 after 9772 hours to Exposure to 1150°F Superheated Steam

a. 5691 hours

b. 9772 hours

Etchant: $\text{HCl-H}_2\text{O}_2$

with scale

2000×

(reduced to 77 percent for reproduction)

Figure 6-10. Microstructure of Inconel-625 Exposed to 1150°F Superheated Steam

TABLE 6-7
CORROSION IN 1050°F SUPERHEATED STEAM
(mg/dm²)

Hours	Inconel-718		Hastelloy-F		René-62	
	Weight Gain	Descaled Weight Loss (mg/dm ²)	Weight Gain	Descaled Weight Loss (mg/dm ²)	Weight Gain	Descaled Weight Loss (mg/dm ²)
1680	7	38	10	57	15	68
2749	8	44	16	60	16	67
3797	4	43	12	58	18	75
4801	4	52	12	60	20	96
5761	10	58	19	65	27	103
9415	6	65	19	72	35	152

TABLE 6-8
CORROSION IN 1150°F SUPERHEATED STEAM

Hours	Inconel-718		Hastelloy-F		René-62		Incoloy-804	
	Weight Gain	Descaled Weight Loss (mg/dm ²)	Weight Gain	Descaled Weight Loss (mg/dm ²)	Weight Gain	Descaled Weight Loss (mg/dm ²)	Weight Gain	Descaled Weight Loss (mg/dm ²)
332	7	32	10	50	12	46		
1680	7	40	12	46	16	77	-5	84
2749	4	45	16	46	17	74	-10	96
3797	6	56	20	50	17	91	-12	116
4801	0	80	14	58	26	108	-15	147
5761	3	113	18	64	33	160	-17°	-
7803	23	-	20	-	66	-	-21	-
8800	-	-	-	-	71	224	-	-
9415	36	219	22	74			-25	213

TABLE 6-9
CALCULATED CORROSION RATES 1050 and 1150°F

Material	Corrosion Rate, K _l (mg/dm ² - mo)		Intercept, ΔW ₀ (mg/dm ²)	
	1050°F	1150°F	1050°F	1150°F
Inconel-718	2.6 ±1.0	15.3 ±2.9	33 ±7	-3 ±20
Hastelloy-F	1.4 ±1.3	2.2 ±1.2	53 ±10	43 ±8
René-62	9.1 ±2.0	15.4 ±3.2	36 ±14	24 ±22
Incoloy-804	-	12.3 ±2.8	-	54 ±20

The oxides of the Inconel-718 and René-62 exposed at 1150°F were examined. The change in appearance of the oxides with time is shown in Figures 6-11 and 6-12, respectively. Planimetric measurements are listed in Table 6-10.

TABLE 6-10
SCALE THICKNESS OF INCONEL-718 AND RENÉ-62
AFTER EXPOSURE IN 1150°F SUPERHEATED STEAM

Exposure Time (h)	Average Oxide Scale Thickness (μ)	
	Inconel-718	René-62
333	-	1.4/1.6
1680	0.9	1.8/1.8
2749	1.3/1.7	1.3/3.4
3797	1.5/2.0	2.7/3.1
4801	2.0/3.7	2.5/3.2
8800	-	5.1/7.8
9415	6.3/7.0	-

The Inconel-718 and René-62 oxidize in 1150°F superheated steam in a normal manner for the first 3000 hours. Beyond that time the oxidation rate apparently increases by some mechanism that cannot be clearly established with the evidence available to date. However, the corrosion could be consistent with theories previously set forth (GEAP-4760) if it were assumed that the barrier film damage resulting in transformation to linear rate law control starts other than at time zero.

6.3 DOCUMENTATION OF OUT-OF-PILE STUDIES

Reports issued to cover the work of the out-of-pile studies under this task are:

Fitzsimmons, M. D., Pearl, W. L., and Siegler, M., "A Simulated Superheat Reactor Corrosion Facility," GEAP-3778, August 1961.

SUMMARY

A versatile superheated steam facility has been developed for out-of-pile corrosion testing of materials that are being considered for application in a superheat reactor (SHR) system utilizing steam from a boiling water reactor (BWR). The following capabilities have been achieved in this facility built as part of the USAEC Superheat Program:

1. Dynamic fuel-cladding-material tests operated with a heat flux of 170,000 Btu/h-ft² for testing tubular cladding under representative heat transfer conditions.

a. 4801 hours

b. 9415 hours

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2000×

(reduced to 77 percent for reproduction)

Figure 6-11. Inconel-718 Oxide after Exposure to 1150°F Superheated Steam

a. 4801 hours

b. 8800 hours

Unetched

1500×

(reduced to 77 percent for reproduction)

Figure 6-12. René-62 Oxide after Exposure to 1150° F Superheated Steam

2. Control of oxygen and hydrogen content in the superheated steam to simulate the gas conditions from radiolytic water decomposition found in a BWR.
3. Attainment of metal temperatures of 1300°F while superheating steam to 1050°F.
4. A SHR process tube test under dynamic conditions simulating a BWR steam water environment on the outside of the tube and superheated steam at any desired temperature below 1050°F on the inside of the tube.

A previously existing simulated BWR corrosion facility was utilized to supply representative BWR steam.

Of special interest is the internal heater developed for the heat transfer test section capable of producing a high thermal flux through prototype cladding materials. The heater is electrically heated, but electrically insulated from the cladding under test. An electrolytic decomposer was designed and installed to dissociate water continuously to oxygen and hydrogen to balance the recombination experienced when the superheated steam contacted the sheaths at metal temperatures up to 1300°F.

The inlet steam to the superheat fuel cladding test sections enters at 546°F and 1000 lb/in.² with an inlet velocity of approximately 100 ft/sec. The steam exits from the third test section at 900 lb/in.² and 1050°F with a velocity of 200 ft/sec. The large pressure drop is caused by the narrow annulus used to obtain the high steam velocities. Coupons of various materials can be tested isothermally at 20 to 30 ft/sec velocities in a 1050°F coupon section.

Gaul, G. G., and Pearl, W. L., "Corrosion of Type 304 Stainless Steel in Simulated Superheat Reactor Environments," GEAP-3779, October 1961.

SUMMARY

Type-304 stainless steel cladding material has been corrosion tested under heat transfer conditions at metal temperatures up to 1300°F in specially constructed out-of-pile superheat facilities. The hydrogen and oxygen contents of the steam have been controlled to simulate those found in boiling-water-reactor-type systems.

Good corrosion resistance and low metal release to system up to metal temperatures of 1100°F were experienced with an expected pattern of an initially high corrosion rate that decreased to a lower constant rate with time up to 4500 hours. A high-alloy layer developed adjacent to the scale in the 1100 to 1300°F metal temperature range on the heat transfer specimens. The layer continued to grow with time, but had little effect on the corrosion rate within the 2500 hours of testing. Deposited carryover products were analyzed. Intergranular attack was associated with scale fluxing in the higher temperature areas.

Gaul, G. G., Pearl, W. L., and Siegler, M., "Stress Corrosion of Type 304 Stainless Steel in Simulated Superheat Reactor Environments," GEAP-4025, February 1962.

SUMMARY

A fuel jacket failure that occurred in May 1961 in the Type-304 stainless steel-clad fuel element exposed in the Vallecitos Boiling Water Reactor (VBWR) superheated steam loop (SADE) was attributed to chloride stress corrosion cracking. To better understand the failure, a test program was carried out to try to reproduce the rapid stress corrosion attack in the simulated superheat reactor environment of the CL-1 superheat facility.

The methods of corrosion testing under heat transfer conditions were modified:

1. To apply a longitudinal stress on the test sheaths to produce a 0.1 percent elongation in 1000 hours;
2. To increase the chloride content of the moisture carryover with the steam by increasing the chloride in the recirculating water to 1.5 ppm;
3. To expose the solids deposits to various metal temperatures.

After 1000 hours of exposure, no significant attack was noted on the test sheaths.

The test procedures were further altered to simulate the significant amount of SADE fuel element exposure to saturated steam at varying temperatures with little to no superheat being generated. A 776-hour total exposure was carried out with the test conditions cycled several times in the following sequence:

1. Inlet - 546°F saturated steam, outlet - 1050°F superheated steam;
2. Inlet - 350°F saturated steam, outlet - 360°F superheated steam;
3. Inlet - 546°F saturated steam, outlet - 550°F superheated steam;

The entrance heater (calculated metal temperature during normal operation 800 to 900°F) was covered with numerous fine cracks that did not penetrate completely through the 0.030-inch sheath. The cracks were predominantly transgranular.

Discontinuities were located by ultrasonic techniques in the middle (metal temperature ~1000°F) and exit (metal temperature ~1200°F) sheaths. The discontinuities were confirmed by metallography as intergranular penetrations of 0.006- to 0.010-inch depth, respectively. Both types of attack had been found on the SADE fuel element cladding failure.

Chloride salts of chromium, copper, iron, and nickel have been found by x-ray diffraction in the deposit taken from the test sheaths. Some laboratory tests with unsensitized and sensitized Type-304 stainless steel indicated that water solutions of copper or iron chloride salts chemically attacked the sensitized material intergranularly, independent of stress.

The test results indicated that chemical and chloride stress corrosion attack can act either singularly or in combination to produce the type failures experienced in the SADE and CL-1 tests.

The presence of stress had little apparent effect on the uniform corrosion rate of the test sheaths, except when the stresses were high enough to cause creep. The creep resulted in scale spalling with some accelerated corrosion in the areas of scale cracking.

Pearl, W. L., and Gaul, G. G., "General and Stress Corrosion of High Nickel Alloys in Simulated Superheat Reactor Environment," GEAP-4165A, March 1963.

SUMMARY

The results of investigation on out-of-pile general and stress corrosion studies carried out to date on Type-304 stainless steel, Inconel-600, Incoloy 800, and Hastelloy-X are presented. The performance of Inconel, Incoloy, and Hastelloy-X was acceptable under conditions that caused failures with previously tested Type-304 stainless steel. The cycle test produced a type failure that can occur in a superheat reactor system, but has not reproduced the most general type failure experienced in the fuel cladding exposures in the SADE facility. Under certain conditions that may exist in a superheat reactor system, oxidizing chlorides (e.g., CuCl_2 and FeCl_3) attack the materials tested to varying degrees with or without the presence of stress.

Pearl, W. L., Gaul, G. G., and Wozadlo, G. P., "General and Localized Corrosion Studies of Type 300 Series Austenitic Stainless Steels in Simulated Superheat Reactor Environment," GEAP-4310, July 1963.

SUMMARY

The tests reported herein include general and stress corrosion studies carried out to date on several 300 series stainless steels in specially constructed out-of-pile superheat facilities.

Heat transfer tests of the stainless steel claddings were run at metal temperatures up to 1300°F and heat flux of 175,000 Btu/h-ft² to produce superheated steam at 1050°F. The hydrogen and oxygen contents of the steam were controlled to simulate that found in steam generated from boiling water reactor (BWR) type systems. The uniform descaled weight losses and metal-to-system losses of the heat transfer tests of Types-310, -316, and -347 stainless steel are summarized and compared to those of Type-304 stainless steel reported previously. Only the Type-316 stainless steel was found to be less corrosion resistant.

A materials screening test utilizing the superheat facilities was developed previously with 1.5 ppm chloride added as sodium chloride to the recirculating water. The test permitted the heater sheaths to be exposed through several cycles to saturated steam (with its accompanying moisture carryover) and superheated steam. Failure of Type-304 stainless steel was obtained in periods of less than 2 weeks with the failures predominantly transgranular in nature. Type-347 and vacuum melted Type-304 stainless steels failed in this NaCl-cycle test, while vacuum melted Type-310 stainless steel was acceptable.

An improved chloride cycle test utilizing 0.5 ppm chloride added as ferric chloride to the recirculating water has been developed. An intergranular type failure was obtained similar to that experienced in the superheat fuel cladding failures in the superheat in-pile loops in the VBWR. Sensitized Types-304 and -316 stainless steel have failed in this test.

During the development and carrying out of the cycle runs, the superheat facilities were exposed to a myriad of conditions within the extremes of the test parameters involved. As such, many pieces of equipment became useful test specimens in developing an understanding of the types of failures experienced in reactor systems over long periods of time. Intergranular chemical attack was experienced essentially independent of stress, but the attack was generally distributed. In the presence of high stress the intergranular attack was more localized advancing normal to the stress. It is hypothesized that a definite interplay exists between chemical attack and stress, and that the application of the stress increases the rate of the intergranular attack preferentially in a direction perpendicular to the stress.

Pearl, W. L., Gaul, G. G., and Wozadlo, G. P., "Localized Corrosion of Stainless Steels and High-Nickel Alloys in Simulated Superheat Reactor Environment,"
GEAP-4450, February 1964.

ABSTRACT

A program was instituted to study and reproduce the in-reactor intergranular failures of Type-304 stainless steel fuel cladding found in superheated steam. The program was directed toward finding ways to eliminate the cause of failure or to use improved alloys that would be less susceptible to failure.

A materials screening test was developed in the out-of-pile superheat facilities with 1.5 ppm chloride added as sodium chloride to the recirculating water in the presence of typical boiling-water reactor quantities of oxygen and hydrogen. During the test, the heater sheaths were exposed through several cycles to saturated steam (with its accompanying moisture carryover) and superheated steam. Failure of Type-304 stainless steel was obtained in periods of less than 2 weeks; the failures were predominantly transgranular. Type-347 and vacuum-melted Type-304 stainless steels failed in this NaCl-cycle test, while Inconel-600, Incoloy-800, Hastelloy-X, Type-406 stainless steel, and vacuum-melted Type-310 stainless steel were acceptable.

An improved chloride cycle test with 0.5 ppm chloride added as ferric chloride to the recirculating water was developed. An intergranular failure was obtained similar to that experienced in the superheat fuel cladding failures in the superheat in-pile loops in the VBWR. Sensitized Types-304 and -316 stainless steel failed intergranularly in this test. Inconel-600, Incoloy-800, and vacuum-melted Type-310 stainless steel did not fail when exposed to the test for much longer time periods.

Brush, E. G., "Electron Beam Microprobe Studies of the Oxidation Behavior of the Iron-Chromium-Nickel System in High-Temperature Steam I. Trace Interpretation Techniques," GEAP-4490, March 1964.

SUMMARY

A technique was developed for the interpretation of microprobe data on the oxidation of Fe-Cr-Ni alloys. Type-304 stainless steel, Incoloy-800, Inconel-600, and Hastelloy-X exposed to steam at 1050°F and 1300°F, were used in the development. X-ray source areas, standard oxide-phase morphological models, and mathematical models were studied.

Pearl, W. L., Brush, E. G., Gaul, G. G., and Wozadlo, G. P., "General Corrosion of Incoloy-800 in Simulated Superheat Reactor Environment," GEAP-4495, March 1964.

SUMMARY

Incoloy-800 fuel cladding material has been corrosion tested under heat transfer conditions at metal temperatures up to 1410°F in specially-designed out-of-pile superheat facilities. The hydrogen and oxygen contents of the steam were controlled to simulate those found in boiling water reactor systems.

The corrosion data from the 4000-hour heat transfer tests indicated good corrosion resistance up to at least 1300°F metal temperature. An analytical model was developed by which the data may be treated to yield corrosion behavior at a given specific temperature. A compositionally changed layer developed at the metal-oxide interface. The changed layer depth appeared to be a function of time and temperature of exposure.

The descaled weight loss data for the sheaths operated at a metal temperature of 1100 to 1300°F indicate that greater than 80 percent of the oxide corrosion product adhered during the first 1000-hour exposure, but only about 50 percent of the total oxidation product remained after 4000 hours.

The uniform corrosion experienced by the Incoloy-800 when exposed isothermally to 1050 and 1150°F for 10,000 hours indicates an initially high corrosion rate that decreases to a lower constant rate within the first 1000 hours. An insignificant amount of the oxide was lost to the system.

Pearl, W. L., Brush, E. G., Gaul, G. G., and Wozadlo, G. P., "General Corrosion of Materials for Nuclear Superheat Applications," GEAP-4760, March 1965.

ABSTRACT

The general corrosion properties of several commercially available materials have been investigated in this AEC sponsored program for application as fuel cladding in superheat reactor systems. The materials have been corrosion tested under heat-transfer conditions at metal temperatures to 1410°F and isothermally at 1050 and 1150°F. The oxygen and hydrogen content of the steam has been controlled to simulate those found in boiling water reactor systems.

Each datum point obtained under heat-transfer conditions represented an integrated corrosion response over a temperature gradient along the length of the electrically-heated test specimen. An analytical model is developed by which these data may be treated to yield corrosion behavior at a given, specific temperature. The model is used to extrapolate the corrosion behavior of Incoloy-800* as fuel cladding. For a 3-year exposure at a metal temperature of 1300°F, Incoloy-800 was calculated to have a maximum corrosion of 0.0022 inch.

The heat transfer corrosion data available for Incoloy-800, Incoloy-825, * Inconel-600, * Hastelloy-X** and several Type-300 stainless steels are updated and summarized.

Results are reported for a 10,000-hour study of several nickel and stainless steel alloys that were exposed isothermally to 1050 and 1150°F superheated steam. Hastelloy-X, Incoloy-825, Inconel-625, * and Incoloy-800 had low initial and long-time linear corrosion rates, and formed good protective and tenacious oxides up to 1150°F. Type-406 stainless steel had a high initial but low long-time linear corrosion rate, and formed a protective and tenacious oxide up to 1150°F. Inconel-600 had adequate corrosion resistance to 1050°F but formed a nonprotective oxide film with a significant portion lost to the system at 1150°F. Type-304 stainless steel had a significant corrosion rate at 1050 and 1150°F, but formed a relatively tenacious oxide at both temperatures that eventually reached a limiting thickness with subsequent spalling.

6.4 DOCUMENTATION OF THE IN-PILE STUDIES

The chemistry studies associated with the SADE and ESADE loops are covered in the following topical reports:

Epstein, L. F., and Evans, T. F., "Deposition of Matter from a Flowing Stream - Part I: General Relations and Equations," GEAP-4140, December 1962.

ABSTRACT

The contamination of system components by radioactive species carried in the coolant of a nuclear reactor can cause problems in accessibility and maintenance. To examine this question, a physical model has been formulated in which the deposition process is determined by mass transfer considerations and an adsorption-desorption mechanism. The bulk concentration of the contaminant in the fluid and the surface concentration on the walls are obtained as the solutions to two simultaneous partial differential equations, in distance and time; and these solutions come out in terms of an integral involving Bessel functions of a purely imaginary argument.

Evans, T. F., "Deposition of Matter From a Flowing Stream Part II: Application to a Radioactive Superheated Steam System," GEAP-4141, December 1962.

*Registered Trademark of the International Nickel Co., Inc.

** Registered Trademark of the Haynes Stellite Co., A Division of Union Carbide Corp.

SUMMARY

In the absence of fission products released from the superheat fuel, activated corrosion products were the principal radionuclides deposited on surfaces exposed to nuclearly superheated steam. Co-60, Co-58, and Cr-51, which are the principal long-lived isotopes commonly present in boiling water reactors using stainless steel- or zirconium-clad fuel, were the principal corrosion products found.

In addition, Ag-110m was a major surface contaminant when silver-plated flange seals were present in the experimental loop. In the presence of fission products released from the fuel, radioiodines were the principal surface contaminants during the first few weeks after reactor shutdown. Following radioactive decay of the iodines, corrosion products again become predominant. Various other fission products were found in much smaller amounts.

Significant differences were found in the amounts of radioactive deposit on various surfaces. Among those tested, leaded nickel silver was particularly receptive to the accumulation of radioactive deposits, but differences were found even among varieties of stainless steel. Under the test conditions (370°), radioiodines evaporated from steel surfaces at rates equivalent to half lives of 6 to 21 hours, although at much lower rates, radio-cobalts also appeared to be removed from surface deposits at rates sufficiently high to be of importance in limiting long-term surface contamination in nuclear superheat systems. During the earlier periods of the SADE tests, surface contamination was due largely to reactor water leakage and carryover rather than to neutron-activated corrosion products from components unique to the superheat system itself. In the more recent tests, surface contamination was low while stainless steel-clad fuel was being tested, but appeared to be somewhat greater in the presence of Inconel-clad fuel.

The deposition processes are considered to include transfer of particles (which may be of molecular size) across a fluid boundary layer to the wall, adsorption on the wall, and subsequent desorption or removal by some other process. The deposition of particles in the micron range is characterized by a sticking coefficient at the wall in addition to a mass transfer constant for transport across the fluid boundary layer. Values for the rate constants calculated from the data and from literature sources were in qualitative agreement, but apply to only a limited range of experimental conditions. These values provide a basis for further experimental investigations and for preliminary estimates of radiation shielding requirements, but further experiments covering a wider range of conditions should be performed to broaden the scope of the data.

Skarpelos, J., "Chemistry of ESADE-VBWR Nuclear Superheat Loop," GEAP-4813,
February 1964.

SUMMARY

Radiochemistry measurements made during the ESH-1 irradiation gave no indication of defective fuel. Results after "sipping" the fuel bundle at the completion of the irradiation indicated that a fuel element might have been defective; however, the signal was sufficiently weak that if a defect were present it was very small.

Radiochemistry measurements made during the ESH-1.1 irradiation and results of sipping the fuel bundle after the irradiation did not indicate that the fuel was defective.

The chemistry of the steam coolant during ESADE operation was significantly better than that provided during SADE operation; however, measurable chloride concentrations in reactor water were periodically observed during the ESADE irradiations.

Sampling facilities provided with the loop did not supply adequate samples for all the chemistry information that was desired. The ESADE experience was used to design improved sampling facilities for the ESADE* Vallecitos Experiment Superheat Reactor (EVESR).

*Empire States Atomic Development Associates

7.0 TASK F - HEAT TRANSFER TESTS - W. A. Sutherland and C. W. Miller

7.1 WORK OF THE LAST QUARTER

Work performed in the reporting period has been directed towards evaluating and reporting the heat transfer and friction data obtained for turbulence promoters in parallel rod arrays. The boundary layer penetration has been investigated by varying the relative roughness parameter and the Reynolds number. Results obtained with a relative promoter spacing $s/h = 10$ are shown in Figures 7-1 and 7-2.

The investigation has been extended to evaluate the effect relative promoter spacing will have on heat transfer performance. Two sets of rods have been prepared with 0.010-inch-high turbulence promoters, one with a promoter spacing $s/h = 7.15$ and the other with a promoter spacing $s/h = 14.82$. This will complement the data obtained previously at a promoter spacing $s/h = 10.0$. The testing has been completed for these spacings in a parallel rod array with rod spacings $p/d = 1.15$ and 1.25 .

7.2 DOCUMENTATION

The following topical reports were issued covering work of the Heat Transfer Task:

Sawochka, S. G., Polomik, E. E., and Levy, S., "Heat Transfer Coefficients with Annular Flow During 'Once-Through' Boiling Water to 100 Percent Quality at 800, 1100, and 1400 psi," GEAP-3703, May 1961.

SUMMARY

"Once-Through" boiling of water to 100 percent steam quality was successfully accomplished in a pre-boiler followed by 0.120- and 0.060-inch annular flow test sections, with electrically heated tubes of 0.625-inch o.d. and 40-inch length. Sufficient data was obtained for preliminary design of once-through boilers in the regions of departure from nucleate boiling (DNB); transition to film boiling; and film boiling at system pressures of 800, 1100, and 1400 psi. Mass velocity was varied from 0.75 to 1.9×10^6 lb/h-ft² and heat flux from 190,000 to 700,000 Btu/h-ft². Steam qualities in the annular heater were in the range of 15 to 100 percent.

Results of DNB, transition and film boiling tests can be summarized as follows:

1. At medium and high steam qualities, operation could be sustained beyond so-called "burnout" design lines based on DNB data, because film boiling coefficients were high enough to maintain the heated wall temperature at acceptable values. For the flow conditions tested, high performance heat flux values of 300 to 400,000 Btu/h-ft² can be used with heater temperatures below 1000°F.
2. At low enthalpies, heat flux at DNB decreases sharply with increase of steam quality; reaches a minimum in the region of 50 to 60 percent steam quality at all three pressures investigated; and finally, increases in the higher steam quality regions.

3. The transition zone is characterized by sharply declining heat transfer coefficients and temperature oscillations. Reduced temperature oscillations are associated with increased pressure and points of reduced heat flux at DNB.
4. Film boiling heat transfer coefficients ranged from 800 to 2000 Btu/h-ft²-°F. The film boiling coefficient increases with flow rate. It approaches the Colburn equation value at 100 percent steam, decreases slightly as the steam quality is reduced, passes through a minimum, and increases as the water weight fraction exceeds 0.45 to 0.65.
5. Correlation of the film boiling data was achieved within ±20 percent using a Colburn-type equation modified by steam quality and steam void groupings.

Sutherland, W. A., "Heat Transfer to Superheated Steam," GEAP-4258, May 1963.

SUMMARY

The physical property variation of superheated steam differs sufficiently from most other gases to warrant experimental investigation of heat transfer performance. Results are reported here of measurements made in a uniformly heated circular duct with steam at 1000 psi. The data agree very well with the expression used for design purposes, which is based on information in the literature for heating of other gases as well as steam.

$$Nu_b = 0.021 Re_b^{0.8} Pr_b^{0.6} \left(\frac{T_b}{T_w} \right)^{0.575} \quad (1)$$

This work was a continuation of that performed under Task (Heat Transfer) of the Nuclear Superheat Project, AEC Contract AT(04-3)189, Project Agreement 13.

Sutherland, W. A., and Miller, C. W., "Heat Transfer to Superheated Steam - II. Improved Performance with Turbulence Promoters," GEAP-4794, November 1964.

SUMMARY

Heat transfer and flow friction measurements have been made with superheated steam in a circular tube to investigate the improved performance to be gained with boundary layer turbulence promoters. Data are presented for hydrodynamically and thermally fully established conditions for one type of discrete roughness configuration. The influence on the heat transfer coefficient of fluid property variation and of Prandtl number are found to be similar for both smooth and rough walls, and the data compare well with predictions for boundary layer turbulence promoters based on measurements with air in an annular geometry.

Figure 7-1. Friction Factor at Various Roughnesses

Figure 7-2. Nusselt Number at Various Roughnesses

Sutherland, W. A., "Improved Heat Transfer Performance With Boundary Layer Turbulence Promoters." GEAP-5115.

SUMMARY

Boundary layer turbulence promoters are considered as a means for improving heat transfer performance in parallel rod arrays. Discrete two-dimensional roughness elements, in the form of small fins transverse to the flow, are investigated experimentally to establish the effect of promoter height and promoter spacing. A general correlation is presented which can be used to optimize the heat transfer performance in parallel rod arrays.

8.0 TASK G - MECHANICAL DEVELOPMENT

8.1 THE TASK

The work of this task was concerned with the development of mechanical components associated with a nuclear superheat reactor. In particular, the major effort was on the development of high performance steam-water separators and dryers. This was terminated in 1963.

8.2 DOCUMENTATION

The following topical reports were issued to cover the work done under this task.

"Nuclear Superheat Project, Interim Report on Steam Dryer Development,"
GEAP-3563, November 1964.

SUMMARY

The description of impingement-type wire mesh dryers of configurations suitable for nuclear superheat fuel designs is given. Test conditions and results are presented for dryer tests in steam-water mixtures at 1000 psig.

It was established that entering steam-water at superficial velocities greater than 5.5 fps must be relatively dry (1 percent moisture or less) to obtain an exit moisture content of 0.1 percent.

Ongman, H. D., and Steamer, A. G., "Interim Report Results of Air-Water and Steam-Water Tests on Primary Steam Separators," GEAP-3564, October 1960.

SUMMARY

Three types of primary steam-water separators were tested. These units utilize centrifugal force to aid in the separation. Testing was carried out using air-water mixtures, and one of the three was tested with steam-water mixtures at 1000 psig.

The separator types are identified as the Archimedes, Internal Tangential Nozzle Upflow, and Radial.

Comparative data was obtained against the Combustion Engineering Turbo Separator, and the first two types were found to be of lower volumetric capacity. The Radial separator could not be directly compared, and was found promising enough to justify steam-water testing.

Descriptions and drawings of all separators tested are presented along with all data obtained.

Riesland, J. I., "Results of Air-Water and Steam-Water Tests on Radial Vane Steam Separator Models," GEAP-3787, August 1961.

SUMMARY

Three radial vane models were constructed for tests. Sufficient test work has been performed to indicate that radial-vane-type separators can meet the design requirements for boiling water superheat reactor application. Exit moisture less than 6 percent by weight was obtained with 4-inch-radius vanes with inlet qualities ranging from 6 to 12 percent for 1000-psig saturated steam-water flow rates up to 9500 ft³/h per inch of vane spacing. Carryunder measurements were less than 0.2 percent on occasions, and indicate that this limitation can be met or very nearly approached in reactor operation of radial vane separators.

Comparison of air-water with steam-water data indicates promise of predicting steam-water capacity from air-water test results. However, more development work is required before this can be assured.

Improvements in development models and additional tests in both steam-water and air-water are required before design parameters of radial-vane-type separators for boiling water superheat reactor application can be reliably established. Further correlation between air-water and steam-water data should be obtained, improvements of nozzle designs are desired, plenum and downcomer designs should be developed, deflectors and other devices at the steam exit of the vanes need to be developed, and assurance of extrapolating data to larger sizes should be established.

Robbins, C. H., "Analyses of the Radial Separator," GEAP-4012, February 1962.

SUMMARY

The analyses presented in this report are both concerned with flow on the vane. The first is concerned with the spread of flow on the vanes due to centrifugal force, which necessitates making the vane height greater than the nozzle. The second analysis deals with the separation of the vapor from the liquid.

The following conclusions are indicated:

1. Spread on the vane due to centrifugal force is not severe, and amounts to about 4 inches above and below the nozzle for a 4-inch radius vane with an 18-inch long, 1/4-inch-wide nozzle. Nozzle velocity does not affect the spread due to centrifugal force.
2. Further experimental work on drag coefficients and bubble sizes is necessary to prove or disprove the equations derived for bubble separation. The conclusions from the analysis are now strongly dependent on many assumptions.

3. The analysis of bubble separation shows that nozzle velocity can be increased without exceeding the limits for separation if:
 - a. The vane radius is increased,
 - b. The film thickness is decreased, and
 - c. The available arc length is increased.
4. The analysis tends to indicate that complete separation on the vane was to be expected in the range of air-water tests conducted. Lack of knowledge of drag coefficients in 545°F water makes a similar calculation difficult for the steam-water tests. The analysis also strongly suggests that separation on the vane is not likely to be a major limitation or problem for the radial separator.

Moen, R. H., "Nuclear Superheat Project Internal Steam Separation Development of Radial Vane Steam Separators," GEAP-4272, May 1963.

SUMMARY

A program to develop primary steam separators for the boiling water section of a nuclear superheat reactor has been worked on since 1959. A full-circle, radial vane separator has been developed as the primary separator. Steam-water tests of this model have demonstrated that it has a vane capacity in excess of that required for the 300-MW_e separate superheat reactor and for the 300-MW mixed spectrum superheat reactor. It is proposed that the vane capacity requirement of the 600-MW_e separate superheat reactor may be attained by increasing the nozzle length.

A model of the flow on the wall of the separator has been proposed and partly checked. A method of designing a reactor separator has been proposed, as well as a future development program to obtain the required information for the design method.

The objectives of the development program have been attained on a model separator. The proposed program should permit attainment of the objectives on a reactor separator.

9.0 TASK H - ESADE

9.1 THE TASK

The nuclear superheat test loops utilized during the P.A. 13 superheat development program were included as Task H of this program. The original SADE loop was replaced with the larger ESADE loop during the program, with the final SADE loop operation completed in April 1962. Operation of the ESADE loop continued until December 1963, when the VBWR, in which the ESADE loop was installed, was shut down. Following this, a final fuel bundle, ESH-3A, was designed and built to adopt the ESADE fuel program to the EVESR reactor.

9.2 DOCUMENTATION

Topical reports covering the work of this task are:

Janssen, E., "Superheat Process Tube Heat Transfer Tests Nuclear Superheat Project," GEAP-3319, January 1960.

SUMMARY

It is important to know the cladding temperature as a function of the rate at which heat is removed from the SADE process tube, under conditions of zero flow. If the outside cladding temperature is known, the inside cladding temperature may be calculated. The object of these tests is to determine the outside cladding temperature.

Tests were conducted using a simulated process tube. The fuel element was simulated by a cartridge-type electric heater, the superheated steam by nitrogen and helium.

The test results show that:

1. Heat transfer by natural convection inside the process tube is negligible; i.e., the heat transfer through the gas (superheated steam) is by conduction and radiation.
2. The coefficient of emissivity ϵ (Type-304 stainless steel tube and liner, and Type-321 stainless steel cladding) is 0.495.

These results were checked by using them to predict the maximum heater surface temperature under test conditions, and comparing them with the actual measured temperatures. The agreement between calculated and measured temperatures is good. The results provide a reliable basis for predicting cladding temperatures in the actual process tube.

Spalaris, C. N., Boyle, R. F., Evans, T. F., and Esch, E. L., "Design, Fabrication, and Irradiation of Superheat Fuel Element SH-4B in VBWR," GEAP-3796, September 1961.

SUMMARY (This summary given previously in Task B)

Irradiation testing of SH-4B, a fuel element designed and fabricated for use in superheat steam applications, yielded valuable information necessary for improving fuel design. The reactor exposure in the VBWR superheated steam loop was terminated when high activity release was detected emanating from the fuel compartment.

Examination of the fuel element, as well as operational data, revealed evidence to show that the jacket failure was due to stress corrosion cracking. Although all conditions necessary to promote stress corrosion cracking were present during irradiation, only the high chloride concentration in the reactor water and water seepage in the loop were unusually high or abnormal and not anticipated in planning the experiment. The test revealed limiting operating conditions with regard to coolant chemistry, and provided data such as UO_2 -jacket interactions, dimensional cladding stability, inadequacy of spacer design and method of attachment, stability of UO_2 under severe conditions, heat transfer, and mechanical behavior in a reactor superheated steam environment.

The fuel element tested was 0.028 inch thick, Type-304 stainless-steel-clad, with an annular geometry cylindrical configuration. The coolant flows downward on the outside, reverses itself at the bottom, and then enters the second (inner annulus) pass to exit on top. The UO_2 was in the form of sintered pellets, 94 percent minimum theoretical density. The total test period in reactor was 617 hours, with a maximum cladding temperature of 1150°F and maximum steam temperature of 900°F maintained for 80 hours.

Upon post-irradiation examination, the fuel element was found to have performed satisfactorily from the dimensional point of view, but two jacket defects penetrating through the 28-mil outer cladding were found. In addition to these defects, other cracks were found, both in the outside and inside jackets. Cracks were predominantly intergranular in nature, except where sensitization of Type-304 stainless steel was less prevalent and some transgranular attack and cracks were present.

The uranium dioxide near the defects was oxidized, and the interior of the cladding had a corrosion deposit diminishing with distance away from the cladding defect. These observations, together with the activity release during reactor operation, point to the fact that element SH-4B operated in the loop with a jacket defect for about 5 to 7 days.

Based on data now available, it may be concluded that abnormally high water leakage in the loop and the concurrent high chloride concentration in the reactor water (1 to 3 ppm maximum for 8 hours) were responsible for the 10,000 parts of chloride per 10^6 parts of iron ions found in the surface deposit. The long-term, wet-dry mechanism for chloride build up on heat-producing surfaces cannot be precluded as another possible source of high chloride surface concentration.

Information relative to the UO_2 -jacket interaction indicated that no axial expansion of UO_2 occurs relative to the fuel jacket during irradiation. Instead, slight contraction of the fuel column was detected. This information is invaluable in designing fuel elements with thin cladding.

Spalaris, C. N., Raymont, W. R., Lyons, M. F., Evans, T. F., Boyle, R. F.,
Murdock, T. B., and Weiss, N. D., "Fuel Element Experiments in SADE-VBWR
Nuclear Superheat Loop," GEAP-4240, May 1963.

SUMMARY (This summary given previously in Task B)

Fuel element experiments were conducted at G.E. - APED from July 1961, to January 1962, with the following objectives:

1. Test improved design and fabrication process,
2. Obtain additional data on failures brought about by irradiation, and
3. Obtain additional information on operational characteristics of the loop.

The experimental results presented in this report confirm earlier data obtained during the SH-4B fuel element test.* Leakage through the flange separating the inlet steam flow from the boiling water moderator resulted in high deposit accumulation on fuel superheat surfaces. Eliminating the flange leakage by seal welding resulted in less crud accumulation. The moisture present in the inlet steam was found to be sufficient to cause some crud accumulation which was associated with the area of the fuel element SH-5A found to be extensively cracked.

All cracks found by metallographic examination were intergranular, with some cracks extending into grain twins. Secondary cracks, adjacent to or away from the main cracks, were found to emanate from the steam side. Grains were dislodged from the cladding matrix exhibiting "holes" with rounded edges; an indication that the corrodant steam and/or impurities played a leading role in dissolving grain boundaries or adjacent grain matrix.

Cracks were highly localized and branched. Areas distant (0.1 inch) from the main cracks were found to be intact and ductile. Element SH-4C grain deformation was found at the edges of the crack, showing that mechanical deformation (stresses) played a leading role in crack propagation.

Crud analysis showed the presence of chloride ions in relatively high concentrations. Quantitative analysis for chloride ions was difficult because of size of crud sample; therefore, a material balance between crud deposit, coolant flow, and moisture carry-over was not possible.

Dimensional measurements showed 0.028-inch-wall fuel cladding to collapse uniformly around the UO_2 at the high heat flux region. Inside diameters showed no significant changes in the 0.028-inch-thick cladding fuel. The 0.016-inch cladding fuel showed uniform collapse both on the inside and outside diameters. The plenum design support was successful. This design was based upon the information obtained from SH-4B regarding UO_2 -cladding relative movement. The "gum drop" in SH-4C confirmed earlier findings that the UO_2 charge did not move toward the plenum, but rather contracted slightly. The over-all fuel length (end plug to end plug) did not change for SH-5A, SH-4, or SH-4B. Measurements for SH-4C show growth, but in view of results to the contrary for elements SH-5A, SH-4, SH-4B, and NUSU, it is suspected that pre- or post-irradiation measurements of SH-4C are in error.

*Spalaris, C. N., Boyle, R. F., Evans, T. F., and Esch, E. L., "Design, Fabrication, and Irradiation of Superheat Fuel Element SH-4B in VBWR," GEAP-3796, September 1961.

Shields. C. M.. "Description of ESADE Loop," GEAP-4617, August 1964.

SUMMARY

The primary purpose of the Expanded Superheat Advanced Demonstration Experiment (ESADE) loop was to provide a significant (ninefold) increase in the fuel irradiation facility over the older SADE loop.

The ESADE loop replaced four Vallecitos Boiling Water Reactor (VBWR) boiling water fuel elements so that, from a nuclear point of view, the loop is a steam-cooled, water-moderated sub-core of the VBWR. Steam is taken from inside the pressure vessel, it flows back into the core, and is superheated by the ESADE fuel elements. The superheated steam is taken outside the pressure vessel, condensed, and returned to the VBWR system. Extensive instrumentation is used to measure the flow rate and temperature of the steam.

The first fuel irradiated was the ESH-1 assembly, consisting of nine two-pass annular fuel elements. The second assembly was ESH-1.1 which had five annular elements from ESH-1, plus three single-rod elements and one three-rod assembly. The third fuel assembly, ESH-4, had nine multirod assemblies, designed and fabricated by the Nuclear Power Department of the Allis-Chalmers Manufacturing Company.

Spalaris, C. N.. "Incoloy-800 for Nuclear Fuel Sheaths (A Monograph)," GEAP-4633, July 1964.

SUMMARY (This summary given previously in Task C)

Incoloy-800* was selected in 1961 as fuel cladding material for nuclear superheat reactors. At that time, the alloy was one of a group of possible candidates. Because of its lower neutron cross-section - relative to other high-nickel alloys - and because of its successful use in petro-chemical and household appliance industries, Incoloy-800 was considered first in relation to the other alloys selected for study. Since 1961, fuel elements sheathed in Incoloy-800 performed satisfactorily in a superheat environment for 135 days without failure, whereas fuel elements clad in Type-304 stainless steel failed within 29 days. In other tests, Type-304 stainless steel failed after even shorter periods of time (10 to 15 days). In corrosion loops where superheated steam conditions were simulated, Incoloy-800 performed satisfactorily for 63 days without failure. Type-304 stainless steel failed within 2 to 3 days in similar experimental runs. Under stress conditions in a corrosion loop that included temperatures of 1150°F and initial chloride concentrations of 70 $\mu\text{gm/in.}^2$ on the specimen surface, Incoloy-800 performed satisfactorily up to 2000 hours of exposure.

Because of the encouraging results obtained to date, Incoloy-800 was chosen as the reference cladding material for fuel cladding applications in superheat reactor cores under study by G.E.-APED. It was decided to group the available data into one report, a monograph, to enable fuel

*ASTM designation: Nickel-Iron-Chromium Alloy. In the absence of a compact, convenient, generic designation, "Incoloy-800" is used throughout this report. "Incoloy," "Inconel," and "Ni-O-Nel" are trademarks of the International Nickel Company, Inc.

technologists to locate the information needed for fuel design or fuel performance analysis. Although detailed information is included, the primary objective of this report is to summarize properties and behavior of Incoloy-800.

a. Availability

Incoloy-800 is available in all forms of wrought products such as tubing, bar stock, flat bars, sheet, pipe, and other standard products. Ingots of the material have been purchased and reduced to wrought products without difficulty. The alloy is described in ASTM Specifications B-407, B-408, and B-409, designated as "Nickel-Iron-Chromium Alloy."

b. Fabricability

Fabricability of Incoloy-800 is comparable to that of Type-304 stainless steel. Welded and drawn tubing has been readily produced in large quantities, and end plug welds for fuel elements have been made without difficulty. Impurities such as sulfur should be excluded from the material or cover atmosphere when welding.

c. Long-Term Stability at High Temperatures

The alloy is relatively stable at temperatures of interest to superheat fuel designs. Exposures at 1050 and 1150°F for periods of over 10,000 hours resulted in reduced ductility, but values lower than 20 to 25 percent elongation were not observed. The principal metallurgical changes detected by electron microscopy were carbide precipitation at grain and twin boundaries. The observed tensile property changes appeared to be related to the combined effect of a strengthened matrix and formation of a continuous intergranular carbide network.

d. Effects of Neutron Flux Upon Tensile Properties

Neutron exposures to 4×10^{20} nvt ($E_n \geq 1$ MeV) at 700 to 800°F do not affect the room-temperature tensile properties of Incoloy-800. When tested at 1100°F, irradiated tensile specimens showed no change in 0.2 percent offset yield strength but did show a drop in ultimate tensile strength and elongation. Minimum elongation values recorded were 23 percent.

e. Strain Cycle Properties

Fractures after strain cycling (slow frequencies, 30 min/cycle) showed plastic deformations. Irradiation decreased the number of cycles to failure at a given strain range by a factor of approximately one third.

f. Electron and Optical Microscopy

Carbide precipitates in various morphologies, which were observed after prolonged exposures at 1050 to 1150 F, have been identified as Cr_{23}C_6 .

g. Corrosion

Incoloy-800 was shown to remain intact in cyclic exposure with imposed stress in a loop operating with water and steam, in which 20 ppm oxygen, 2 to 3 ppm hydrogen, and 0.5 ppm chloride (iron chloride) were added to simulate reactor superheated steam conditions. Under similar conditions, Types-304, -347, -316, and -304 vacuum-melted grades of stainless steel developed localized cracking within 2 weeks. The tests with Incoloy-800 were carried out for 63 days without a sign of cracking.

Under steady-state corrosion at 1300°F in ex-reactor oxygenated steam, Incoloy-800 showed a corrosion rate which gave a predicted 3-year maximum weight loss of 6520 mg/dm², or 3.3 mils metal loss (95 percent confidence limits). Under similar conditions, nonlinear extrapolation of data showed a metal loss of 1 mill per 3 years.

Coupons under stress and with chloride salts on their surface exposed to 1150°F in steam showed no cracking or localized corrosion attack.

h. Fuel Element Irradiation Experience

Exposures of 135 days at peak heat fluxes of 445,000 Btu/h-ft² with annular fuel elements demonstrated that Incoloy-800 was less susceptible to cracking failure than Type-304 stainless steel by at least a factor of 5 in the environment tested. No failures were evident in Incoloy-800 sheaths, whereas Type-304 stainless steel failed intergranularly within 12 to 15 days in a similar reactor environment.

The EVESR (ESADA-Vallecitos Experimental Superheat Reactor) core contains 150 annular geometry fuel elements clad with Incoloy-800. This represents about one-half of the entire reactor core. In addition, four advanced design fuel assemblies with rod clusters will be inserted in the EVESR by September 1964. Information on the performance of this fuel will be used to influence further use or improvements regarding Incoloy-800 as superheat fuel sheath material.

Hazel, V. E., Boyle, R. F., Busboom, H. J., Murdock, T. B., Skarpelos, J. M., and Spalaris, C. N., "Fuel Irradiations in the ESADE - VBWR Nuclear Superheat Loop," GEAP-4775, March 1965.

SUMMARY

A multichannel loop in the VBWR permitted the simultaneous testing of nine fuel elements, under superheat conditions, with steam supplied from the boiling portion of the reactor core. Selection of Incoloy-800, Inconel-600, Type-310 VM stainless steel, and Type-304 VM along with Type-304 stainless steel for fuel cladding materials was due to prior experience in the program, where Type-304 stainless steel was subjected to stress-accelerated localized intergranular corrosion attack.

The first fuel bundle, ESH-1, contained nine annular fuel elements which were irradiated for 29 days. The second fuel bundle, ESH-1.1, contained five annular elements from the ESH-1 test, one three-rod cluster, and three single-rod-type fuel elements.

The operational mode of both assemblies was similar to the previous experiments, where the chloride level in the water was maintained at 0.02 ppm, with occasional short time (2 to 4 hours) peaks of 0.1 to 0.5 ppm. Maximum temperatures of the fuel cladding were in the range of 1000 to 1300°F, with steam temperatures to 850°F.

One Type-304 stainless steel element from the ESH-1 test contained through-cladding failures in the outer cladding. Although fission product release gave evidence that these cracks completely penetrated the cladding during the post-irradiation examination, these cracks were initiated and penetrated well into the cladding during the irradiation test.

No cladding failures were experienced in the ESH-1.1 test, as confirmed by the absence of fission product release during and after operation, and by post-irradiation visual observations, ultrasonic testing, and metallography.

Rod-type elements were spaced with: 1.) helical wires attached only at the end plugs, and 2.) brazed spacers; both of which performed satisfactorily. No signs of fretting or wear were detected on the outer surface of the elements. All elements decreased in over-all length from 0.010 to 0.150 inch, for a total fuel element length of 40 inches. Diametral collapse was non-uniform, with formation of longitudinal wrinkles in the annular Inconel fuel elements, which had original diametral gaps of 0.004 to 0.007 inch. Maximum collapse strain for the high power annular fuel elements was 0.007 inch/inch.

Metallography at the peak heat flux regions showed adherent uniform corrosion films approximately 5×10^{-5} inch thick. No accelerated intergranular corrosion attack was found in the alloys tested, although substantial degrees of sensitization were present as expected from the cladding operating temperature. The superior corrosion resistance of Inconel and Incoloy was compared with previous experiments where, under similar conditions, Type-304 stainless steel failed by localized corrosion within 29 days of reactor exposure. Except for carbide precipitation, microstructures were unchanged from preirradiation samples. Minor changes in microhardness of sheaths confirmed the annealing-out of most fast neutron damage. Significant changes in the hardness of brazed material in the BONUS Type-348 stainless steel cladding rod spacers show that further use of the brazing technique for this application must be done with caution. Fission gas release was less than 1 percent of theoretical values, consistent with the prevalent UO_2 maximum temperature of 2800°F.

These results show the relative superiority of Incoloy and Inconel performance in superheat conditions, over that of Type-304 stainless steel. Types-304 VM, -310 VM, and -348 stainless steel performed satisfactorily. However, these materials operated at lower stresses and heat ratings than for the annular fuel. This experiment also suggests that at low stresses, even Type-304 stainless steel will last longer in the superheat reactor environment, illustrating the importance of stress upon the mechanism of failure experienced with annular, highly-stressed Type-304 stainless steel cladding fuel elements.

In addition to these topical reports, several supplemental sections were included in various Quarterly Reports to cover some aspects of the work under this task. These supplemental sections are:

Pennington, R. T., "Nuclear Superheat Project. First Quarterly Report,"
GEAP-3290, Section II - A Detailed Description of the SADE Loop.

SUMMARY

The loop is designed to test prototype fuel elements for superheat reactors under environmental conditions. The initial design provides for nine elements in parallel flow channels.

The test facility consists of three fundamental parts:

1. Equipment within the reactor vessel,
2. Main loop equipment outside the reactor vessel, and
3. Gas collection and filtering system.

The primary loop coolant is steam which is collected in the upper region of the reactor vessel adjacent to the access port. The steam flow is down around the outside of the nine parallel tubular fuel elements and up through the center of each element. The superheated steam passes to the external piping through a flanged connection at the access port. The steam is then directed by piping outside the reactor through two parallel flow control valves, a condenser, and finally to the reactor condensate system.

A detailed description of the facility and a discussion of proposed system operation are given. Provisions for controlling release of radioactive gas are described. Heat transfer and physics characteristics are discussed, along with the possible safety hazards associated with the loop operation and the safety features included to mitigate the severity and probability of accidents.

Pennington, R. T., "Nuclear Superheat Project, Eighth Quarterly Progress Report,
April - June 1961," GEAP-3785, July 1961, Section II - Superheat Advance Demonstration
Loop - SADE.

SUMMARY

The SADE loop provides a fully-instrumented, flexible test facility. The fueled section and connecting piping is inserted into the VBWR through one of the pressure vessel refueling ports. Saturated steam from the dome of the VBWR pressure enters at the top of the fuel element and makes a single pass down the outside and a single pass up through the center of the fuel element before leaving the vessel through the refueling port connection. The test facility is provided with air cooling as backup cooling on startup.

The SADE design condition is for 825°F superheated steam at a thermal power of approximately 50 kW.

A complete description of the loop, its instrumentation, physics analysis, and early operating experience are presented.

Pennington, R. T., "Nuclear Superheat Project. Fourteenth Quarterly Report, October - December 1962," GEAP-4153. January 1963. Appendix A, Irradiation of NUSU, A Boiling Water-Superheat Steam Fuel Element by C. N. Spalaris, R. F. Boyle, F. A. Comprelli, T. F. Evans, and T. B. Murdock.

SUMMARY

A combination boiling-superheating fuel element was irradiated in the SADE-VBWR. The annular geometry and loop hardware permitted cooling by boiling water on the outside diameter, whereas the inside surface was cooled with steam. The outside diameter was clad with Type-347 stainless steel and the inside diameter was clad with Inconel-X alloy.

The fuel element was irradiated for 34 days (at power) for an accumulated average exposure of 722 MWD/T.* The total number of cycles during the entire exposure was 21 from VBWR power levels of 15 MW_t or greater. The maximum heat flux was 236,000 Btu/h-ft², with an exit steam temperature of 770°F (maximum of 780°F for 48 hours) and a corresponding maximum steamside cladding temperature of 1175°F. The corresponding waterside heat flux was 390,000 Btu/h-ft².

During the exposure, sampling of the loop coolant flow indicated fission product release from the fuel with a maximum rate of release of 46.1 μ Ci/sec. Water entered the loop from a leaky flange at rates of 0.5 to 20 lb/h.

Although the loop radiochemistry measurements indicated a leak through the NUSU cladding or the end plugs, post-irradiation examination failed to show evidence of a cladding defect. It is possible that a small leak, existing during reactor exposure, closed up upon removal of the fuel from the loop.

Detailed metallographic examinations showed no evidence of cladding defect or localized corrosion attack. Slight corrosion film buildup was uniform in thickness and free of the intergranular corrosion attack observed in fuel elements clad with Type-304 stainless steel irradiated previously in the loop. Chloride deposits were found on the cladding surfaces, undoubtedly due to water leakage through the flange and due to cycling temperature and pressure conditions of the loop throughout the VBWR exposure.

*From cesium measurements.

Fiock, W. L., "Nuclear Superheat Project, Twenty-Second Quarterly Report, November 1964 - January 1965," GEAP-4790, February 1965, Appendix A - The ESH-4 Irradiation Program, T. B. Murdock and H. J. Busboom.

SUMMARY

The ESH-4 fuel bundle with nine fuel elements fabricated by the Allis Chalmers Company was irradiated in the ESADE loop during a 17-day period. The basic ESADE fuel bundle, originally designed to accommodate nine separate two-pass annular superheat fuel elements, was revised to handle nine single-pass fuel elements comprising seven fuel rods in a cluster.

Reference (1)* (Page 1) stated the purpose of the ESH-4 test "as the first phase of a comprehensive irradiation study of various fuel materials and concepts in superheat steam."

The ESH-4 fuel bundle was irradiated in the ESADE loop of the VBWR from November 23, 1963, through December 9, 1963. The irradiation test was terminated on December 9, when the VBWR was shut down permanently to deactivate this facility. No unusual problems or difficulties were experienced with this fuel bundle, and there were no indications of failures of any of the fuel elements in this fuel bundle during the period of operation.

RML post-irradiation examinations were confined to nondestructive operations because of the short period of irradiation of this fuel bundle. No cladding failures or severe corrosion attacks were observed.

The fuel elements included cladding of Incoloy, Hastelloy-X, and Type-316 stainless steel of varying thicknesses. Ferrule and special wire spacers were used. Fuel burnups of 130 to 210 MWD/T uranium were achieved.

The fuel bundle was in the reactor for a total of 634 hours. The reactor was critical for 326.5 hours during this period, and at significant power (5 MW reactor power or higher for a total of 320 hours, with 297.4 hours of this time at 15 MW or higher.

The reactor water chloride level was 0.05 ppm or less throughout the period of irradiation.

* 1. Bezella, W., "Thermal Analysis of ESH-4 Fuel Element Assemblies," Allis-Chalmers Manufacturing Company.

10.0 TASK J - MSSR STUDY

10.1 THE TASK

This work was to study the design of a small prototype reactor to demonstrate the important features of a Mixed Spectrum Superheater. Of primary concern in this study were the characteristics of the core during flooding and unflooding of the superheater, the Doppler effect in the fast superheating section, and the control and coupling between the superheating and boiling sections of the core. This work was terminated in 1962.

10.2 DOCUMENTATION

Reynolds, A. B., "Flood Safety of the Mixed Spectrum Superheater,"
GEAP-3737, May 1961.

SUMMARY

Calculations indicate that the flooding effect on reactivity in the fast oxide superheater section of the Mixed Spectrum Superheater (MSS) can be rendered small by the addition of epithermal poisons. The reactivity effect of flooding the superheater was calculated as a function of the concentration of the epithermal poison europium in the superheater and as a function of the superheater core size. For the Pu-239-fueled 3.5-foot cubical superheater, 0.23 volume percent europium oxide must be added to the superheater to produce zero reactivity change between the unflooded and completely flooded superheater. For the 3.5-, 2.0-, 1.5-, and 1.25-foot U-235-fueled superheaters, the corresponding volume percentages of europium oxide which must be added are 0.06, 0.8, 2.7, and 4.4 percent, respectively.

The effects of partially flooding the superheater by the uniform addition of water to the coolant volume and the effect of loss of steam pressure were calculated for a 3.5-foot plutonium core and a 1.5-foot U-235 core. In both cases it was possible to choose a europium concentration which would limit the maximum reactivity addition from complete loss of pressure to complete flooding to less than 1 dollar.

Reactivity effects during the flooding process in which water floods the superheater exit steam pipe and rises up into the fast core were estimated for a 3.5-foot plutonium core and a 1.5-foot U-235 core. The reactivity throughout this flooding process is negative whenever the reactivity of the completely flooded superheater is negative.

A high degree of flexibility through the use of various combinations of epithermal and thermal poisons is available to obtain a given reactivity response during a flooding or unflooding accident. Flexibility through use of various geometrical arrangements of the poison has not been studied. Further study is needed to specify the required reactivity response for the various possible mechanisms by which flooding or unflooding can occur.

Calculations were made using nineteen group diffusion theory. Cross sections above 9100 eV were obtained from Argonne National Laboratory fast reactor cross-section compilations. In the resonance region, cross sections for U-238, europium, and the fissile materials were obtained from the calculated Doppler broadened energy dependent resonance integrals. Statistical variation of both the unknown fission widths and the unknown neutron widths were taken into account for the fissile materials.

Brynsvold, G. V., Hikido, K., Reynolds, A. B., and Riley, D. R.,
"Conceptual Design Study for a 75 MWe Mixed Spectrum Superheating
Reactor Power Plant," GEAP-4016, February 1962.

SUMMARY

1. Description of Plant and NSSS

The Mixed Spectrum Superheating Reactor (MSSR) is a nuclear reactor contained within a single pressure vessel in which both boiling of water and superheating of steam take place. The reactor core consists of four annular regions, which, going from inside to outside, are: an unmoderated region (fast core), an unmoderated buffer region (fast buffer), a water-moderated buffer region (thermal buffer), and a water-moderated boiling region (thermal core). The boiling region is very similar to that in a conventional boiling water reactor core. Both moderated regions are cooled by water; the unmoderated regions are cooled by steam.

The neutron spectrum varies from a thermal spectrum in the boiling part of the core to a fast spectrum in the superheating region. Thus, the designation of MSSR is applied to the present concept.

Water is boiled in the outer boiling core region. The steam-water mixture passes up through the chimneys above the fuel elements to the free surface where the steam is separated. The steam passes through the dryers and then travels down through an offset duct to the unmoderated superheating core. Water surrounding the offset duct reduces radiation streaming to the vessel head to minimize activation of the head. After passing through the superheating core, the steam leaves the pressure vessel through the pipe that penetrates the vessel bottom head. Beneath the superheating core within the fast core shroud is a series of shielding plugs which reduces radiation steaming to the vessel bottom.

The boiler is cooled by natural circulation. The water separated from the steam flows downward outside the chimneys where it mixes with 300°F feedwater. The subcooled water then flows downward between the boiler shroud and the pressure vessel to the bottom of the vessel where it turns and flows upward through the boiling core. Nozzles have been provided in the present vessel for later conversion to forced circulation if higher power operation is desired.

The reactor is pressurized to 1015 psia. Superheated steam is supplied to the turbine at 800 psia and 950°F. Feedwater is returned to the vessel at about 300°F. A reactor thermal power of 206 MW, of which 23.5 percent is supplied by the superheating core, is required to produce 75 MW of gross electrical power. The gross cycle efficiency is estimated to be 36.4 percent.

The boiler power is controlled manually by bottom-mounted poison elements. The superheating core power and the thermal buffer are controlled by top-mounted control drives. In addition, coupling control poison blades which are bottom-mounted are provided between the fast and thermal cores. System steam pressure is regulated automatically by the turbine control and bypass valves. The superheater steam outlet temperature is indicated with thermocouples and controlled manually by manipulating the control elements in the fast core. Load rejection is absorbed by the use of full-capacity bypass valves to the condenser.

2. Reactor Building

The reactor building contains the Nuclear Steam Supply System. This includes fuel transfer equipment, fuel storage room for new fuel, pool storage space for spent fuel, fuel pool cooling and cleanup system, emergency condenser, emergency steam dump tank, shutdown heat exchanger system, emergency liquid poison injection system, and reactor service cooling water system. A 60-ton-capacity polar crane is provided for reactor servicing, fuel handling, and general maintenance. The reactor and its radioactive components are contained within a vapor-tight, steel enclosure. The building is designed to withstand the effects of a maximum credible accident which would yield an estimated peak pressure of 30 psig. It is felt that future versions of the MSSR will be able to take advantage of pressure suppression as a means of containment. In the present plant, however, where the primary purpose is to demonstrate feasibility of the concept as well as to establish plant characteristics, it is felt that dry containment is more appropriate. Two air locks, one for personnel and the other for equipment are provided in the building.

Inside the reactor building, much of the structures are reinforced concrete, which also serves as shielding where required. Space is provided within the building for the addition of recirculation pumps and other equipment necessary to double the capacity of the system.

3. Balance of Plant

Primary emphasis in this study has been placed on the Nuclear Steam Supply System. The design of the Balance of Plant does not differ greatly from conventional plant practice. Only sufficient information is generated to provide a basis for the cost estimate.

4. Plot Plan

The conception of the general layout is based upon the following main principles:

- a. Interlocation of buildings according to function.
- b. Accessibility of reactor building, turbine hall, and loading dock by road and railway.
- c. Possibility of exercising access control to controlled areas, i. e. , to reactor building and part of the turbine hall.
- d. Future expansion to double the generating capacity of the plant.

To provide the shortest possible length of steam and feedwater connections, the reactor building is placed adjacent to the turbine hall but at the opposite end of the building from the turbine. In this position, it is out of line from the rotating part of the turbine, thus minimizing the possibility of damaging the reactor building by a turbine accident. It will also be between the two turbines, if a second unit should be added later.

Heating and ventilation facilities are placed in the end of the turbine building nearest the reactor building. The stack is next to the reactor building near the ventilation room. The rad-waste facility is located next to the river in the vicinity of the reactor building and turbine hall.

Office building and control room are located adjacent to the turbine hall near the reactor building. The present plot plan also shows an access control building connected with the office building. This building and the control room may be combined with the office building, since the space shown is probably more than adequate. A warehouse and shop are also provided.

11.0 TASK L - MIXED SPECTRUM SUPERHEAT CRITICAL EXPERIMENT

11.1 THE TASK

This task was to conduct experiments in the MSCA intended to verify the calculational techniques and nuclear constants utilized to predict the physics characteristics of a mixed spectrum superheating reactor.

11.2 DOCUMENTATION

"Summary Safeguards Report for the Critical Experiment Facility - Vallecitos Atomic Laboratory," GEAP-4054, July 1962.

SUMMARY

This report was prepared to meet the following objectives:

1. To provide a document containing technical specifications and supporting data for the Critical Experiment Facility (CEF) modified for operation with two Critical Assemblies, the Thermal Critical Assembly, and the Mixed Spectrum Critical Assembly.
2. To provide sufficient information to justify operation of the facility with the latitude required to accomplish the experimental and developmental work for which the facility is constructed.
3. To discuss procedural and instrumental control to be used to insure safe operation of the CEF.
4. To discuss the changes in the CEF neutron instrumentation and safety system resulting from the modification of the facility to include the Mixed Spectrum Critical Assembly.
5. To evaluate the safety of the facility by investigating potential hazards in the operation of the CEF and analyzing a postulated maximum credible accident.

Data and discussion are presented to meet these objectives.

Reynolds, A. B., "Physics Design of the Mixed Spectrum Critical Assembly,"
GEAP-4320, August 1963.

SUMMARY

The Mixed Spectrum Superheater Reactor (MSSR) is an integral superheater reactor in which boiling occurs in an annular Boiling Water Reactor section, and steam is superheated in an unmoderated fast section in the center. A Mixed Spectrum Critical Assembly (MSCA) to be operated at the Vallecitos Atomic Laboratory has been designed to mock up a 75- to 150-MW_e prototype MSSR.

The principal experimental measurements aimed at proving the feasibility of the MSSR concept include power distribution, Doppler effect, flooding effects, distribution of reactivity, control rod worths, and the effect of the control system on the power distribution.

Physics Design

The physics design criteria include a power split between the fast and thermal sections with ~ 0.3 of the power in the fast section, a power distribution such that no large power spikes exist at the boundaries of the different zones, and prompt negative temperature coefficients in both the fast and thermal sections.

To obtain the required power distribution, an unmoderated fast-buffer region and a water-moderated thermal-buffer region must be provided between the fast and thermal cores. The calculated critical enrichment of the Inconel-clad UO_2 fuel in the fast core is 19.9 percent, which results in a U-235 inventory of 350 kg. The fast core enrichment for an alternate fast core design containing 0.016 volume fraction polyethylene was also calculated to be 19.9 percent. The enrichment of the steel-clad UO_2 fuel with aluminum with a water-to-fuel ratio of 2.35. The thermal buffer fuel contains natural uranium; the thermal core fuel enrichment is 1.8 percent U-235.

The fast core is controlled by four safety rods and two shim control rods. All rods consist of movable fuel assemblies; in addition, the four safety rods have boron-carbide followers. The calculated reactivity worth of each safety rod is 1.6 percent $\Delta k/k$; the worth of each shim control rod is 0.4 percent. The combined worth of the four safety rods is 3.6 percent. To provide an additional ultimate shutdown mechanism, 16 of the 68 fast core fuel assemblies are supported by U-235 fuses which will give way during an excursion, allowing the fused assemblies to fall out of the core. The total worth of the 16 fused assemblies is 3.7 percent. The values reported here for the control and fused assemblies were calculated with the thermal core present; the effect of the thermal core is to reduce the worth of these assemblies below their worth in an all-fast system.

The thermal core will first be loaded to criticality with only the fast buffer fuel in the fast section. The fast core fuel will then be loaded gradually, proceeding in steps from a thermal reactor to the MSCA design loading of 68 fast core assemblies. Design calculations have been made for the initial loading without the fast core present, and for intermediate loadings containing 44, 56, and 64 assemblies.

Safety Considerations

There are two important prompt negative-reactivity temperature effects in the fast section of the MSCA, the Doppler effect and the axial thermal expansion effect, which would limit an excursion in the MSCA. The maximum tolerable excursion for the fast section of the MSCA was postulated to be the condition at which the hottest fuel reaches 2000°F. This temperature is based on the estimate that the Inconel cladding, which also reaches the fuel temperature since no coolant is present, will not fail below 2000°F.

The calculated reactivity due to the Doppler effect for the excursion described in the previous paragraph is $\Delta k/k = -0.26$ percent. The Doppler effect in the fast section of the MSCA is appreciable due to the degraded neutron spectrum in the UO_2 fuel (about 80 percent in the contribution

to the Doppler effect occurs below 10 keV). Inconel is being used as the structural material in the fast section instead of steel to enhance the Doppler effect due to its higher scattering cross section. An alternate fast-core design contains 0.016 volume fraction polyethylene to simulate a larger amount of moderator such as BeO which could be added to the MSSR prototype, to further degrade the neutron spectrum and enhance the Doppler effect. The effect of the added polyethylene is to double the MSCA Doppler effect. Three methods of measuring the Doppler effect are being analyzed, including heating of the entire fast section, irradiating U-238 foils at different temperatures, and pile oscillator techniques.

The second prompt reactivity effect in an excursion of the fast section is the axial expansion of the fuel. The calculated reactivity due to this expansion during an excursion to a maximum fuel temperature of 2000°F is $\Delta k/k = -0.13$ percent.

The thermal core lattice is typical of low-enriched aluminum-clad BWR lattices with a water-to-fuel ratio of 2.35. Both the Doppler coefficient and the void coefficient are negative for this type of lattice. The Doppler coefficient is about $-2.5 \times 10^{-5} \Delta k/k^\circ\text{C}$ at room temperature, where k refers to the thermal core (not to the criticality factor of the coupled system). The void coefficient is $-2 \times 10^{-3} \Delta k/k$ per percent voids for the thermal-core criticality factor.

Flooding the fast section in which no epithermal poisons (such as hafnium) are present would introduce a large positive reactivity; hence, the fast section has been designed so that accidental flooding is incredible. Partial and complete flooding of some of the fuel assemblies will be carried out experimentally with and without hafnium present. The calculated reactivity effect of completely flooding the central four assemblies with no hafnium present is +2.9 percent $\Delta k/k$. By adding 0.03 volume fraction of hafnium to the fast core (and increasing the U-235 enrichment from 19.9 percent to 21.8 percent, essentially no change in reactivity should occur on flooding the central four assemblies.

Petersen, G. T., Warzek, F. G., Johnston, H. F., Jaworowski, T. R., Hallam, J. W., and Meyer, P., "Mixed Spectrum Superheat Experiments - Thermal Core Measurements," GEAP-4620, June 1964.

SUMMARY

The MSCA was loaded to initial criticality without the fast core present on December 28, 1963. This process consisted of inserting 588 1.3 percent enriched fast buffer fuel rods, 160 natural enriched thermal buffer fuel bundles, and 402 thermal core fuel bundles. In general the agreement is good. The "just critical" loading at 20°C was determined to be 396 bundles compared to the calculated value of 385 bundles. This agreement is considered excellent for the annular reactor since it represents only 0.5 percent in multiplication. The worth of fuel added to the periphery agrees within the experimental error to the predicted worth. The experimental value for (β/ℓ) is only slightly lower than the predicted value. A small decrease in the calculated (β/ℓ) would be anticipated if detailed spatial effects were taken into account in the calculation. The temperature effect is in excellent agreement. Since a detailed safety sheet worth calculation was not performed, the discrepancy between the calculated and measured worths is not significant. Finally the radial power peaking, based on bundle averages was determined to be about 5 percent $\Delta p/p$ greater than the predicted value. No predictions were made for the local power peaking within a fuel bundle.

Wall, I. B., Warzek, F., "Mixed Spectrum Superheat Critical Experiments," GEAP-4680, October 1965.

ABSTRACT

An experimental program has been conducted in the Mixed Spectrum Critical Assembly to measure integral and differential parameters in the central fast spectrum zone, and to establish the reactor physics characteristics of the coupled fast thermal reactor, thereby demonstrating concept feasibility. The Mixed Spectrum Superheater Reactor concept consists essentially of an annular boiling water reactor coupled to a central fast spectrum superheater. The following measurements were performed in the fast spectrum: the neutron spectrum between 12.5 and 115 keV using a proton recoil chamber; the neutron spectrum between 0.14 and 2950 eV using six resonance detectors; the Doppler effect by the activation and reactivity methods; reactivity worths of small samples using an oscillator; and fission ratios of U-233, U-234, U-235, U-236 and Np-237 using small solid-state diodes. Since all these measurements were performed at essentially the same location, the integral reaction and reactivity data are consistent with the spectrum determination, an important fact in the evaluation of fast reactor cross-section data. Extensive comparative calculations have been performed. Further experiments demonstrating concept feasibility, included superheater flooding and fuel compaction effects, power distribution control at superheater-boiler interface, and power split control. In all aspects, the measured effects were significantly within the limits established by the earlier feasibility evaluation.