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TITLE

DESIGN OF PRODUCTION TEST IP-247-A-8-FP, IRRADIATION OF 1.47% ENRICHED SELF-SUPPORTED I & E FUEL ELEMENT IN RIBLESS PROCESS TUBES

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W. H. Hodgson

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DESIGN OF PRODUCTION TEST IP-247-A-8 FP, IRRADIATION OF 1.47% ENRICHED SELF-SUPPORTED I & E FUEL ELEMENTS IN RIBLESS PROCESS TUBES

INTRODUCTION

To evaluate the self-supported fuel element concept, tests are underway to determine the performance of collapsible bridge-rail supported fuel elements in ribless process tubes under present reactor conditions at B Reactor. It appears expedient, however, to extend this evaluation to future operating conditions in order to establish the relative feasibility of conversion to self-supported fuel elements in ribless tubes in all present reactors. Utilization of 1.47% U-235 enrichment will provide fuel element powers comparable to those attainable under proposed future conditions. Since I & E fuel elements of this enrichment have previously attained exposures in excess of 2000 MWD/T at specific powers averaging 75 KW/ft in C Reactor[1], this test will specifically evaluate the feasibility of the self-supported fuel element concept. The purpose of this report is to present the design of a test to fabricate and evaluate self-supported fuel elements under conditions of comparable severity to those expected for future loadings of this geometry.

OBJECTIVE

The objective of this test is to evaluate the resistance of projection fuel elements to jacket corrosion at high specific power.

TEST SUMMARY

Four columns of 1.47 per cent U-235 self-supported I & E elements in the ribless tube demonstration facility and four columns of 1.47 per cent U-235 I & E fuel elements in ribbed tubes will be irradiated in B Reactor to a goal sufficient to demonstrate significant improvement of performance of self-supported fuel elements over standard I & E fuel elements, or until two ruptures occur in each type.

BASIS

Preliminary testing[2,3] of both solid and OIIN I & E self-supported fuel elements in the ribless tube facility in 105-B has indicated that the concept of a self-supported fuel element in a ribless process tube is feasible under current operating conditions. Specifically, it has shown that the bridge-rail supports are adequate, do not severely obstruct flow channels, and do not of themselves create an area of rapid corrosion. Nine columns of solid self-supported and sixteen of I & E self-supported fuel elements have been discharged to date without evidence of excessive localized rapid corrosion. In summary, nothing has been found as yet to indicate that the self-supported fuel element will not prevent misalignment and associated jacket corrosion ruptures.

1. HW-43078 C, "Production Test IP-1-A-73 MT, Evaluation of I & E Slugs Operating at High Specific Powers", F. W. Van Wormer, 9-13-56, Secret.

2. HW-50991 A, "Production Test IP-84-A, Evaluation of Slugs Having Projection for Use in Ribless Tubes", R. E. Hall, 3-12-58, Secret.

3. HW-50991 B, "Supplement A to Production Test IP-84-A, Evaluation of Slugs Having Projections For Use in Ribless Tubes", R. E. Hall, 9-29-58, Secret.

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Prior to removal of the ribless process tubes from B Reactor, it is desirable to evaluate self-supported I & E fuel elements under future reactor operating conditions. It is necessary that this pilot test loading be accomplished on as rapid a schedule as possible in order that this program may be adequately integrated with the zirconium tube replacement schedule for which tubes are currently being procured.

The method chosen to pilot these conditions involves the use of enriched (1.47% U-235) fuel elements to attain high specific power, oversize fuel elements to attain high annular coolant temperatures with respect to the interior coolant channel and short charging to control the maximum corrosion rate.

FABRICATION OF TEST MATERIAL

1. Components for Self-supported Elements

a) Cans and spires - Can and spire components shall be X-8001 alloy aluminum of the CIIN (1.460" x 0.375") dimension reduced in length to fit a 7 inch length core. Canned fuel dimensions shall be 1.460" x 0.375" x 7.640". Designation of components to be cut down D-2.

b) Cores - Cores will be seven-inch length, enriched (1.47% U-235) uranium I & E (1.370" OD x 0.481 ID) and shall meet all quality requirements for cores used for IP-1-A-73 MT[4], as well as current fuel standards.

c) Supports - The bridge-rail supports shall be the same type used previously[3] except that the height shall be reduced to retain the effective rib outside diameter of 1.600". Eight ribs shall be attached, four to each end at 90° spacing, using spot welding techniques.

2. Components for Comparison Standards

a) Cans and spires - Can and spire components shall be X-8001 alloy aluminum of the OIIN (1.445" x 0.310") dimensions reduced in length to fit a 7 inch length can. Canned fuel dimensions shall be 1.445" x 0.310" x 7.640". Designation of components to be cut down O-2.

b) Cores - Cores will be seven-inch length, enriched (1.47% U-235) uranium I & E (1.356" OD x 0.416" ID) and shall meet all quality requirements for cores used for IP-1-A-73-MT, as well as current fuel standards.

3. Assembly

All fuel elements shall be fabricated by the L Process (Lead Dip Process for Heat Treated Uranium, HW-47029).

4. HW-43078 A, "Design of Production Test IP-1-A-73-MT, Evaluation of I & E Slugs Operating at High Specific Powers", W. C. Riley and F. W. Van Wormer, 5-16-56, Secret.

[redacted]

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4. Quality of Material

All standard quality control measures will be applied in fabrication of fuel elements.

5. Quantity of Material

A total of 104 acceptable self-supported fuel elements and 104 acceptable standard control fuel elements is required for this test.

TESTING

Pre-Irradiation Testing and Measurements

1. Measurement of length, ID and OD 1/2 inch from each end and in the center, and warp is required for each test fuel element. Weight of all test fuel elements is required.

2. Identity and position of measured fuel elements will be recorded by a series and position number stamped on the base of each element.

IRRADIATION TEST

Four columns of 20 pieces each of self-supported fuel elements will be charged in the ribless process channels and four columns of 20 pieces each of comparison controls will be charged in order in ribbed process channels in B Reactor. Exposure goal will be based on the demonstration of a significant improvement of performance over standard elements or until one rupture is sustained, at which time the test will be discharged.

POST IRRADIATION EXAMINATION

All fuel elements irradiated as part of this test shall be visually examined and weaseled; and shall have weight losses, diameter growth and warp recorded. Ribs shall then be removed from a few selected elements for non-destructive can-wall thickness determination. Selected elements may be examined destructively in the radiometallurgical laboratory if desired.

TEST AUDIT

Notebook HW-49756 is to be used to record all data pertaining to this test. Any unusual incidents, whether or not they seem important at the time, are to be recorded in the notebook. The Appendix to this report contains a list of items to be used as a guide for preparation of the notebook.

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TEST LIMITATIONS OR HAZARDS

Precautions for handling and storage of the unirradiated enriched uranium for this test are outlined in specification NS4.0 in HW-47013[5].

R. E. Hall
Reactor Fuels
Process and Reactor Development
Irradiation Processing Department

W. H. Hodgson
Process Engineering
Engineering Section
Fuels Preparation Department

5. HW-47013, "Nuclear Safety Specifications, Fuel Element Manufacturing Process", 12-10-56, Confidential.

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APPENDIX

DATA TO BE RECORDED IN PRODUCTION TEST NOTEBOOK

I. Fuel Element Core

A. Lot Number

B. Heat Treatment

1. Technique
2. Form of material when treated
3. Attendant conditions

C. Finished Dimensions (Pre-assembly)

D. Properties

1. Orientation data
2. Tensile data
3. Other significant properties

E. Pre-assembly treatment (including inspection)

II. Can Lot Number

III. Assembly Process

A. Process Used
B. Brief Explanation of Process
C. Attendant Conditions

IV. Post Assembly Treatment

A. Protective Coating
B. Autoclaving
C. Other Treatment
D. Inspection

V. Pre-Irradiation Examination

A. Dimensions
B. Warp

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VI. Irradiation

A. Coolant Temperature

1. Inlet
2. Outlet

B. Panellit and Crossheader Pressures
C. Number of Shutdowns (Controlled and Scram).
D. Other

VII. Post Irradiation Examination

A. Visual
B. Warp (And Other Profilometer data)
C. Fracture Data
D. Other

VIII. Log. Running account with date recording the progress of the PT, unusual incidents (such as a slight change in process during assembly), and other deviations whether or not they seem pertinent at the time of occurence.

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