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PRODUCTION TEST PT-IP-810
INSTALLATION OF FLAT ALUMINUM HORIZONTAL
CONTROL ROD, D REACTOR

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PRODUCTION TEST PT-IP-810
INSTALLATION OF FLAT ALUMINUM HORIZONTAL CONTROL ROD
D REACTOR

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By DK Hanson, 5-19-94

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PRODUCTION TEST PT-IP-810
INSTALLATION OF FLAT ALUMINUM HORIZONTAL CONTROL ROD
D REACTOR

INTRODUCTION

The two small reactors (B and D) currently utilize a flat aluminum horizontal control rod consisting of three cooling tubes surrounded by boron carbide-aluminum neutron absorber rings and an outer casing of aluminum. Two adverse and complementary characteristics of these rods have served to limit the functional reliability of the rods under current and anticipated reactor operating conditions. The construction of the rod presents a relatively high resistance to transfer of heat from the sheath to the cooling water and the neutron absorber rings show evidence of significant dimensional increase with high accumulated exposure to neutrons.^{1,2,3} With poor heat transfer and current reactor operating levels, the rod sheath temperature may exceed that for which aluminum is suited and the material loses its strength by annealing. Swelling of the neutron absorber rings further aggravates the overheating by increasing the cooling tube to ring gap. The net result is a tendency for hot spots to occur in the rod sheath and in the more advanced cases, these develop into blisters which cause the rod to bind in the channel and/or the step plug.

The above problem is, of course, most prevalent among those rods which reside in the reactor during operation, but it also affects the inner end of the remaining rods since this portion is exposed to the same high temperature and high neutron flux environment.

A new aluminum horizontal control rod (HCR) has been designed to avoid the above problem as a part of Facilities Engineering Section's program to provide means of alleviating horizontal control rod problems at all of the reactors.⁴

OBJECTIVE

The purpose of this Production Test is to authorize installation of a newly designed flat aluminum HCR prototype (see Figure 1) in HCR channel No. 2 at D Reactor for test. The objectives of this test are:

1. Measure rod sheath temperature under normal in-reactor service conditions with the reactor operating at full power level.
2. Establish minimum cooling water flow requirements consistent with cooling water outlet temperature limits.
3. Measure effective control strength of the rod by means of an in-reactor calibration.
4. Confirm that the mechanical operation of the rod under reactor service conditions is satisfactory.

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BASIS AND JUSTIFICATION

Softening and bulging of the existing HCR's in the two smaller reactors by the mechanism previously described is not only a problem to routine functional reliability of the rods but it opens the potential for mechanical failure of the sheath in attempting to move a stuck rod and the resultant costly repair work. Five rods have bulged to the extent that they had to be replaced. In addition, at least 26 more rods show some degree of overheating and accompanying material deterioration.⁴ These figures are current to the first of FY 1966 and are totals for B, D, F, DR and H Reactors, all of which have the same type of HCR. This represents about 55% of the total number of rods at these reactors.

A new design for an aluminum HCR which optimizes the rod cooling and which utilizes vibration packed boron carbide powder as a neutron absorber has been developed (see Figure 1). An analysis of the thermal characteristics of this design⁵ indicates that with a maximum graphite temperature of 890° C and a power level of 2500 Mw, the maximum sheath temperature would be 200° F which is well within the long term service capabilities of 6062 aluminum of which the rod is constructed. Use of vibration packed boron carbide powder should minimize the effect of dimensional change in the boron upon irradiation. The existing C and K rods also use a vibration packed boron carbide powder neutron absorber, and there has been no evidence of this creating a problem.

The calculated control strength of the new rod is about 90% of that of the existing rod. On this basis, a preliminary measure of rod control strength is not planned. It is, rather, planned to calibrate the rod in-reactor to confirm control strength.

Two prototypes of the new aluminum HCR in a half rod configuration are being procured. One will be used in laboratory tests⁶ to measure pressure drop versus flow, dimensional stability of the sheath versus flow (and the attendant internal pressure) and sheath strain versus flow. The other rod, intended for testing in-reactor, will be equipped with two thermocouples to measure rod surface temperature. These thermocouples are sheathed in Inconel 600 tubing 0.024" OD. They will be mounted flush with the surface of the rod in grooves cut into the rod and secured in place. Thus, the thermocouples will not interfere with the function of the gas seal. Inconel 600 is prone to develop long lived induced radioactivity; however, considering the very small size of the thermocouple sheaths, this will not constitute a problem.

Channel No. 2 at D Reactor has been chosen as the test location for the following reasons:

1. The rod in this location is a half rod which is normally about 90% to 95% inserted during reactor operation; therefore, temperature measurements should be quite representative of the maximum to be expected.

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2. No. 2 location should represent the most stringent functional environment for a B-D type reactor and hence, a maximum test for the rod.
3. D Reactor is more convenient than B Reactor to the 189-D laboratory where the rod will be prepared for test.

The information derived from this test will be pertinent to solution of HCR problems at B and C Reactors as well as D because this rod design has direct application potential at both of these other reactors. It is not desirable to perform the test at C because new channel liner plates and a new type of step plug would be required to install the rod. The new rod can be installed as a direct replacement for the existing rod at B or D Reactors.

PROCEDURE

The new aluminum HCR prototype is to be tested in HCR channel No. 2 at D Reactor. Since the rod is a direct replacement for the existing rod, no channel or step plug modifications are required.

The procedure is presented in three parts: a) Installation, b) Cooling Water System Modifications, and c) Test Procedure.

A. Installation Procedure

1. Dismantle the No. 2 HCR assembly and remove the existing control rod.
2. Borescope the entire channel including the step plug to determine channel condition. If the step plug bushings are damaged or if the dimensions are not in accord with those authorized in Design Change No. 734, the step plug bushing insert assembly is to be replaced with a new assembly.

Note: Design Change No. 734 authorizes a bushing aperture dimension 1.639 inches high on all but the first (entrance) bushing. The height of the hole in the first bushing is 1.550 inches. The hole through the shield or spacer pieces is 1.793 inches high.

3. Perform a full length elevation traverse of the HCR channel.
4. Install the new control rod and connect to the drive section.

Note: Extreme care must be exercised in handling the new control rod to avoid damaging the thermocouples attached to it.

5. With the new control rod in place, uncoil the thermocouple lead wires attached to the outer end of the rod. String these along the drive

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section and secure in place in a manner which will avoid their being damaged by the drive pinion and/or the rod guides. These leads will be utilized to read rod sheath temperature during reactor operation.

6. Restore the HCR assembly to operating condition and operate the rod in and out to assure proper movement.

B. Cooling Water System Modifications

Analysis of the thermal characteristics of this rod⁵ indicates that a flow rate of 20 to 22 gpm will be needed to maintain the coolant outlet temperature within the 60° C limit specified in the process standards. This is a calculated value which should provide ample cooling water. One of the purposes of this test is to confirm the minimum cooling water flow rate consistent with outlet temperature limits. Variation in flow rates within the range being considered here (12 gpm to 22 gpm) will have little effect on the rod sheath temperature.

Modifications to the HCR cooling water system are outlined below. Refer to the schematic of the system shown in Figure 2.

1. Remove the existing orifice plate in the No. 2 rod cooling water outlet line. Replace with the new 0.445 inch diameter orifice provided (see Figure 3). The new orifice is designed to provide a back pressure at the higher flow similar to that provided by the existing orifice at current flow for convenience in control room monitoring of rod water flow.
2. Install sensing lines from both sides of the cooling water outlet orifice for No. 2 rod and run them to a point where a portable differential pressure gauge can be attached to them and monitored during reactor operation. The upstream line should be "tee'd" into the existing line at a point near the orifice flange. There is an unused flange tap on the downstream side which can be used directly.
3. Connect a 50 psi range differential pressure gauge to the above sensing lines. This gauge will be used to provide an exact measure of cooling water flow to No. 2 rod. Figure 4 shows the calibration curve for the new orifice plate as determined by laboratory test.
4. Increase the HCR cooling water supply pressure to 110 psig by adjusting the pressure regulating valve in the outer rod room. The pressure relief valves are presently set for 135 psig so the increased supply pressure will not require resetting these valves. The current supply pressure is 75 psig. The increased pressure should produce a flow of 20 to 22 gpm to the new rod.
5. Adjust the globe valve on the upstream side of the hose reel of all

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HCR's except No. 2 to provide normal flow through the rod as indicated by the control room pressure gauges.

6. New flow calibration data for No. 2 HCR pressure gauge in the control room will be provided. Annunciator trip setting on No. 2 HCR gauge will be adjusted as discussed under "Process Standards" section of this document.

C. Test Procedure

The following procedure is designed to accomplish the objectives of this Production Test.

1. Objective: Measure rod sheath temperature.

Method: When the rod has attained a reasonably stable position with the reactor at full power level, measure the temperature indicated by the two chromel-alumel thermocouples installed on the rod. In the first startup utilizing the new rod, the sheath temperature should be checked at about half and three-fourths of full reactor power level to assure that the temperatures are within limits for the material (300° F).

2. Objective: Establish minimum cooling water requirements.

Method: Initial operation of the rod will be with 20 to 22 gpm cooling water flow. Rod water outlet temperature must be carefully monitored during reactor startup to assure that the flow is adequate to maintain the temperature within the 60° C limit. Once the adequacy of the initial flow quantity has been established, the flow will be reduced in steps to determine minimum flow required to maintain the cooling water outlet temperature within the 60° C limit with the rod in normal operating position and the reactor at normal power level. Flow decrements will be determined by the Design Engineer, based on flow-temperature data from the initial operating period. Flow quantity at each step shall be determined by measuring the pressure drop across the outlet orifice for No. 2 HCR.

3. Objective: Confirm nuclear control strength of the rod.

Method: Perform an in-pile calibration of the new rod and compare with similar data for the existing style rod in the same location.

4. Objective: Confirm mechanical operation.

Method: Observe freedom of rod movement on installation and during subsequent reactor operation.

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HAZARDS

The possibility must be recognized that cooling water flow provided will be inadequate to maintain outlet temperature within limits with the rod inserted well into the operating reactor. Although a thorough analysis of the rod heat transfer characteristics has been made⁵, by necessity this was made on a parametric basis which resulted in a range of flow requirements. Since internal heat generated from neutron capture is a significant portion of the heat load, the situation does not lend itself to definition by laboratory testing. The cooling water flow provided should be ample; however, if a problem does materialize, the rod must be removed from service. This would undoubtedly require a minimum reactor shutdown. Reactor safety, however, would not be compromised.

PROCESS STANDARDS

Process Standard A-060, HW-46000-D, Horizontal Rod Cooling Water, is affected by this Production Test. Deviations specified herein apply only to the prototype rod being tested in No. 2 HCR position. Limits for all other HCR's shall remain as stated in the standard.

A. Part 60.01-C, Inlet Rod Water Pressure Limit

Although plans call for increasing the basic pressure to 110 psig, this will not constitute a deviation from this standard. The standard specifies a maximum pressure of 150 psig. The test rod has been hydrostatically tested at 100 psig. With a 110 psig supply pressure, the maximum pressure the rod would experience is 60 psig. Blockage of the cooling water outlet orifice which could subject the rod to higher pressures would be cause for immediate action on the basis of reduced cooling water flow.

B. Part 60.01-D, Rod Water Pressure Annunciator Trip Setting

Trip settings for all but the test rod will remain unchanged. For the test rod, low trip setting will be equivalent to 15 gpm initially. The appropriate control room gauge setting will be determined as part of the flow calibration during installation of the test rod. As the test progresses, however, it may be necessary to adjust this to permit the required flow adjustments.

Note: The orifice location is 23 feet, 6 inches above the control room gauge for an equivalent of 10.2 psi in water elevation head.

High trip setting will remain unchanged from that specified in the standard.

With the exception of that specified above, this Production Test does not violate the intent of any process standard. For assessment of nuclear safety, the control rod may not be considered to be a part of the nuclear control system until it has been calibrated.

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PROCESS EQUIPMENT STANDARDS AND EQUIPMENT MAINTENANCE STANDARDS

Process Equipment Standards A-030 and D-010, HW-41000, and Equipment Maintenance Standard No. 119 are affected by this Production Test. Deviations from these standards will affect only the prototypic rod to be tested in No. 2 HCR position.

A. Standard A-030 and EMS 119

Part 30.02, Outlet Orifice Requirements

The outlet orifice size for No. 2 rod only will be changed from the specified 0.344 inch to 0.445 inch to accommodate the increased flow to No. 2 rod. The necessary calibration data for the new orifice are included as part of this Production Test.

B. Standard D-010

Part 10.01, Horizontal Rod Maintenance

This Production Test Authorization permits departure from the subject standard to install a horizontal rod in position No. 2-D Reactor which is not in accordance with drawing H-1-26928.

With the exception of the items specified above, this Production Test does not violate the intent of any Process Equipment Standards or Equipment Maintenance Standards.

COSTS

Material costs for the proposed work will be borne by Reactor Design, Facilities Engineering Section. Labor costs will be borne by D Reactor Operation.

It is estimated that two shifts of non-chargeable reactor outage time will be required to complete the work.

ISSUE AND EXPIRATION DATES

Issue date for this Production Test is October 22, 1965.

Expiration date for this Production Test is July 1, 1966.

If the test results are satisfactory and the rod is functioning properly at the end of this test, present plans are that the thermocouple leads will be cut at the connector block and the rod will remain in the reactor as a routine operating HCR.

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RESPONSIBILITIES

D Processing Operation

Schedule the work for an appropriate reactor outage and maintain reactor safety during the course of the work.

Approval: J. T. Baker
J. T. Baker, Jr., Manager
D Processing Operation

D Maintenance Operation

Perform the installation work outlined and provide craft support for subsequent test adjustments in cooling water flow.

Approval: R. R. Meyers
R. R. Meyers, Manager
D Maintenance Operation

Process Physics, Process Technology Sub-Section, Research and Engineering

Perform an in-pile calibration of the test rod and make appropriate recommendations relative to properly accounting for it in nuclear safety analyses.

Approval: G. F. Bailey
G. F. Bailey, Supervisor
Process Physics Unit

Reactor Design, Facilities Engineering Section

Prepare the Production Test Authorization. Provide the rod prototype and all accessory equipment required for the installation. Arrange for cooling water flow tests. Provide technical support for the rod installation, evaluate data and issue a summary report.

Approval: R. S. Peterson
R. S. Peterson, Supervisor
Reactor Design Unit

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APPROVALS

H. W. Reacock
H. W. Reacock, Manager
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For W. T. Jessen
R. T. Jessen, Manager
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J. G. Baker, Jr.
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R. S. Bell
R. S. Bell, Manager
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S. M. Graves
S. M. Graves, Manager
Manufacturing Engineering Sub-Section

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REFERENCES

1. HW-84116, Blisters in BDF-Type Control Rods, An Interim Report, G. W. Riedeman, 9/16/64 (Confidential).
2. HW-84116-A, Blisters in EDF-Type Control Rods, Second Interim Report, G. W. Riedeman, 5/25/65 (Confidential).
3. HW-84116-C, Blisters in BDF-Type Control Rods, Final Summary Report, G. W. Riedeman, 8/30/65 (Secret).
4. RL-REA-580, Remedial Program, Horizontal Control Rod and Vertical Safety Rod Problems, B, C, D, and K Reactors, P. H. Hutton, 2/23/65 (Secret).
5. HW-82373, B, D, F, DR, H Reactor New Aluminum HCR Concept, Temperature Study, J. D. Agar, 7/28/64 (Secret).
6. RL-REA-2484, Design Test No. 1302, Mechanical Tests, Flat Aluminum Horizontal Control Rod, P. H. Hutton, 8/23/65 (Confidential).

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HCR COOLING WATER SYSTEM
B&D REACTORS
(REF. DWG'S. HW-71350 & HW-73635)
FIGURE 2

PHH
10/8/65

ORIFICE PLATE
HCR COOLING WATER OUTLET
B & D REACTORS

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FOR USE WITH ALUMINUM TEST HCR (H-1-14483) ONLY

FIGURE 3

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IPD CODE DEVIATION RECORD NO. 6
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HORIZONTAL ROD ASSEMBLIES

APPROVALS

Manager, Development Engineering and Process Design Sub-Section 10-27-65
Date

Manager, Facilities Engineering Section 10/29/65
Date

Manager, Research and Engineering Section 11-2-65
Date

Manager, Manufacturing Section 11-2-65
Date

Manager, Operations 11-2-65
Date

Date Issued 11/2/65

IPD CODE DEVIATION RECORD NO. 6
ASA B31.1, PRESSURE PIPING CODE

HORIZONTAL ROD ASSEMBLIES

Areas

100-B, 100-D, and 100-K.

Buildings

105-B, 105-C, 105-D, 105-KE, and 105-KW.

Equipment

The horizontal control rod assemblies presently installed and the horizontal control rod design of PT-IP-810.

Applicable Code

ASA B31.1, Code for Pressure Piping - Section I, Power Piping Systems.

Description of Deviation

The use of the aluminum materials as specified on the referenced drawings for pressure sustaining components of the horizontal rod assemblies is not covered under the power piping section of ASA B31.1

Reference Drawings and Specifications

H-1-10104 -- Horizontal Rod Tip Section Assembly and Details, 105-B, D.

P-9020 -- Horizontal Rod Tip Section Assembly and Details, 105-C.

H-1-22069 -- Horizontal Rod Assembly

H-1-14483 -- New Aluminum Horizontal Rod

HWS-7287 -- Procurement Specification for Aluminum Horizontal Control Rod
Assembly

Reason for Deviation

Aluminum alloys have low modulus of elasticity, good corrosion resistance, high

thermal conductivity, and low nuclear cross-section, making them suitable for the reactor service conditions that the horizontal rod tip sections must withstand. Aluminum alloys are better suited for the reactor service conditions than the materials included in Section I, Power Piping Systems, ASA B31.1, Code for Pressure Piping.

This code deviation is being issued to conform with the requirements of HAP0 OPG 4.2 and the IPD Pressure Systems Guide.

Discussion

Aluminum alloys have given many years of satisfactory service as structural and pressure containing components of the horizontal control rod assemblies in the Hanford reactors. The aluminum components are limited to the tip section of the horizontal rod assemblies and are shielded from operating personnel when exposed to operating conditions that result in the aluminum and coolant exceeding room temperature.

Hydrostatic and dynamic tests are performed on the horizontal rod assemblies to assure that they perform in accordance with their intended design. The coolant supply and effluent systems are also instrumented to monitor individual rod coolant inlet pressure and outlet temperature with control room annunciation to assure abnormal conditions do not exist.

Although the specific aluminum materials are not covered in Section I, Power Piping Systems, ASA B31.1, Code for Pressure Piping, aluminum materials of similar alloys are permissible under the ASME Pressure Vessel Code, Section VIII (1965).

Certification

I certify that the above deviation maintains the intent of the applicable code.

Reactor Design
Development Engineering and Process Design
Facilities Engineering Section

Oct. 27, 1965
Date

DATE

FILMED

9/16/94

END