Rec. 12/1/86
EJB

Westinghouse Electric Corporation

Power Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

November 25, 1986
NS-NRC-86-3181

Mr. Earl J. Brown
Office for AEOD
U. S. Nuclear Regulatory Commission
Washington D.C. 20555

50-266/301

Subject: **Request for Information on Westinghouse Rod Cluster Control Assembly Performance**

Dear Mr. Brown:

In response to your request to Wisconsin Electric Power Company for information related to the Point Beach LER on RCCA cladding wear, enclosed is documentation summarizing Westinghouse RCCA performance. The following non-proprietary documents attached for your information are representative of technical information on RCCA wear submitted to utilities:

1) Summary of visual inspections of the RCCAs at the Point Beach Nuclear Plant (Unit 1, Cycle 11/12, Unit 2, Cycle 9/10).

2) Westinghouse customer information letter on RCCA wear.

3) Extracts from a brochure containing a summary of Westinghouse current design experience regarding RCCA wear.

In the past, the vehicle used to transmit information on RCCA performance has been the topical WCAP-8183 "Operational Experience with Westinghouse Cores". This is updated annually and issued for input to the NRC Annual Fuel Performance Report.

Very truly yours,

for E. P. Rahe, Jr., Manager
Nuclear Safety Department

/jwh

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Attachment (1)

Summary of Visual Inspections of RCCAs
at the Point Beach Nuclear Plant

During the Cycle 9-10 refueling shutdown at Point Beach Unit 2 in June, 1983, site personnel observed what appeared to be cladding wear in excess of the current Westinghouse criteria on some Rod Cluster Control Assemblies (RCCAs). The RCCAs in question were of the spider mounted design with 16 rodlets per RCCA and are compatible with the 14x14 fuel design used at Point Beach. The absorber material is Silver - Indium - Cadmium. All of the Unit 2 RCCAs were examined with a periscope, and based on the observed wear, Westinghouse recommended replacement of one RCCA and axial repositioning of the normal parked position of the remaining RCCAs by 2-3 steps in order to minimize additional wear. A further recommendation was to inspect the Unit 1 RCCAs for wear indications during the Cycle 11/12 refueling shutdown (Unit 1 had approximately 2 more cycles of operation than Unit 2). This inspection was recently completed and resulted in a Westinghouse recommendation to replace 21 of the 33 RCCAs with an inferred calculated wear depth in excess of the current wear depth criteria.

WEAR OBSERVATIONS

The rodlets were photographed using a 35mm camera coupled to a periscope. An analytical model was developed which was used to infer the depth of wear indications from the observed wear scar widths as measured from the photographs. Quantitative measurements of wear depth have not been performed to date.

Two types of wear patterns were observed and are characterized as follows: (1) Axially continuous wear scar on the rodlet of varying depth probably caused by RCCA axial stepping motion and scrams; (2) Approximately one-inch long score marks at elevations corresponding to the guide plate surfaces.

The wear indications are predominantly on the portion of the cladding surface facing the center or hub of the RCCA. The attached sketch shows wear patterns which are typical of the worst observed wear. Note that, by design, some wear is expected due to contact between the RCCA rodlets and the guide surfaces.

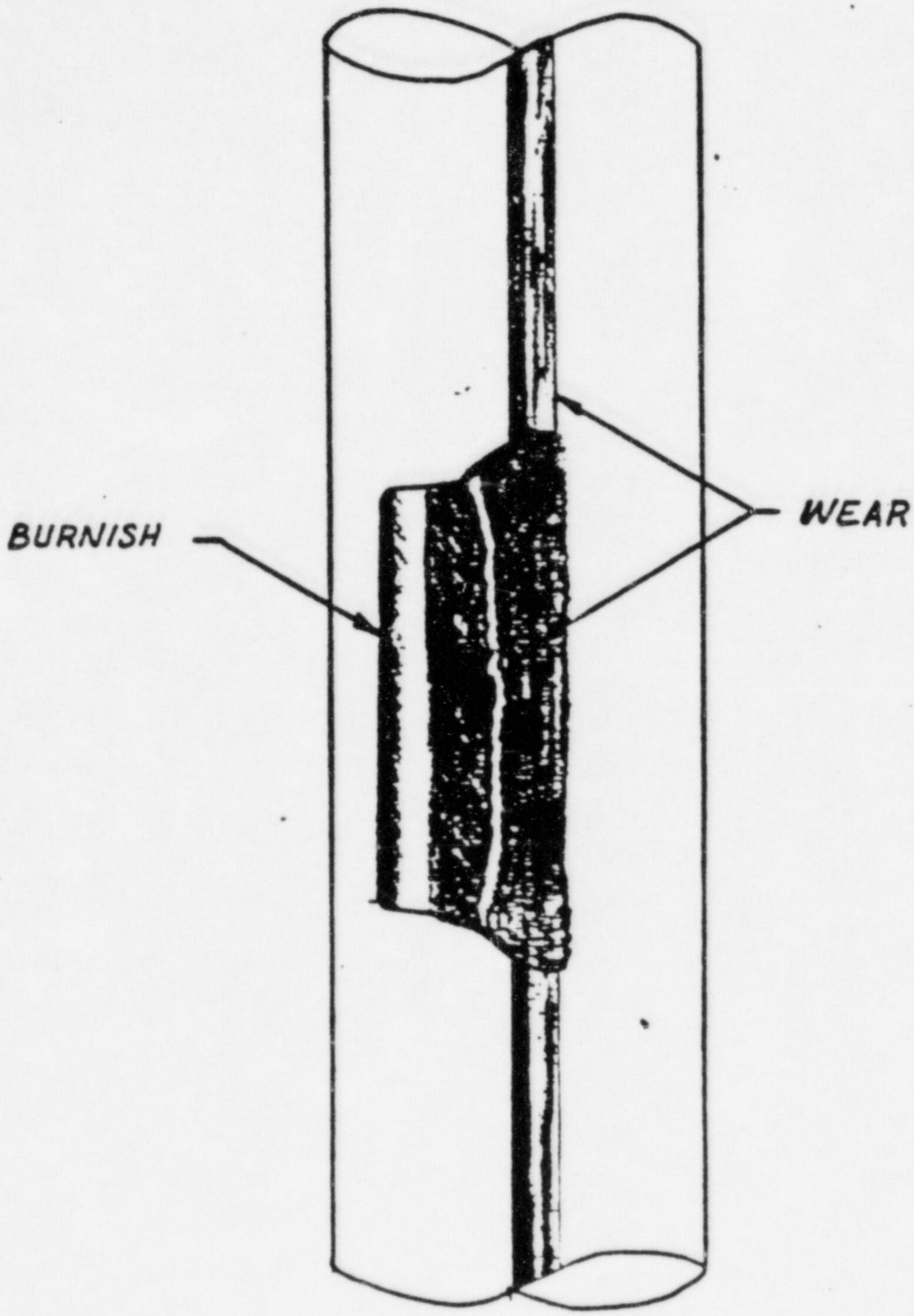
RODLET WEAR CRITERIA

The current limiting wear depth criterion is related to absorber rodlet clad collapse. The allowable wear depth was determined based on autoclave tests (at temperature and pressure) wherein sealed, hollow tubing samples with simulated wear scars were pressurized until collapse occurred. This test of a hollow tube did not take into account the support provided by the absorber material, and therefore is conservative. Westinghouse recommended that the Point Beach RCCAs be replaced based on this criterion and on plant specific operating history.

A preliminary evaluation by Westinghouse indicated that an unreviewed safety question does not exist.

SKETCH SHEET
FORM 28577
MJC 1-30-84

WESTINGHOUSE ELECTRIC CORPORATION



ATTACHMENT 2

NFD CUSTOMER INFORMATION LETTER ON RCCA WEAR

Subject: NFD Customer Information Letter
RCCA Wear

Reference: March, 1984 letter

TO: Customer

Dear Sir:

The referenced letter issued to you earlier this year provided a summary of visual inspections of the Rod Cluster Control Assemblies (RCCAs) at the Point Beach Nuclear Plant and recommendations for future actions. Those inspections identified the existence of wear marks on the cladding of the RCCA absorber rodlets. Some of the wear scars were estimated to exceed the existing Westinghouse maximum wear depth limit. As a result the affected RCCAs were replaced based on our recommendation. The Point Beach RCCAs had operated for 13 years from Cycle 1 initiation with approximately 80% availability.

New developments related to RCCA rodlet cladding wear have been completed. These include a revision to the wear depth limit and a new method of obtaining wear depths from photographs. I have attached an update to inform you of the status of on-going efforts by Westinghouse in the area of RCCA wear. I would be happy to answer any questions you may have on this subject.

Signed
Fuel Projects Engineer

RCCA WEAR UPDATE

Introduction

In our last information letter on Silver-Indium-Cadmium Rod Cluster Control Assembly (RCCA) wear, we brought to your attention in-reactor RCCA wear experience. The following summarizes that letter. The RCCAs at Point Beach Unit 2 (End of Cycle 9-June, 1983) and Unit 1 (end of Cycle 11-December, 1983) showed significant levels of wear. Photographs of the wear scars were taken and a method was developed to estimate wear depth. Some of the RCCAs had estimated wear depths exceeding the then applicable wear depth limit and were therefore replaced. Two types of wear scars were seen : (1) Stepping Wear - an axially continuous scar on the rodlet caused by RCCA axial stepping motion and scrams, and (2) Fretting Wear - an approximately one-inch long score mark at elevations corresponding to guide cards (see Figure 1). Our letter on this subject also described the basis of the Westinghouse wear depth limit and presented preliminary recommendations for future plant specific action. Note that it was concluded that no unreviewed safety question existed at the Point Beach units.

During the inspections it was observed that both types of wear occurred at the interface between the RCCA and the upper internals guide tube guide plates. Based on this study as well as previous studies, no unusual or significant wear has occurred at the interface between the RCCA rodlet tip and the fuel assembly guide thimble tube. Therefore, the observed RCCA wear has no adverse impact on the operation or safe performance of Westinghouse fuel assemblies.

The following sections provide updated information on further inspections, new developments, and plans for hot cell work on the Point Beach RCCAs.

RCCA INSPECTIONS PERFORMED

In addition to the two Point Beach (2 loop, 14x14 fuel) inspections described in our previous letter, Westinghouse has visually inspected RCCAs at three other reactors, two 2 loop 14x14 and one 4-loop 15x15 plant. As a result of these inspections we have concluded that the types of wear observed at other reactors are similar to those observed at Point Beach. The wear did not appear as severe as that at Point Beach; however, the RCCAs were several years younger.

NEW DEVELOPMENTS

Since the last information letter, Westinghouse has completed a more exhaustive analysis in three areas described below: (1) the wear depth limit; (2) the interpretation of photographs to estimate wear depths; and (3) axial repositioning of RCCAs. The following briefly describes these new approaches.

First, a new approach has been taken to establish a Westinghouse RCCA cladding wear depth limit which is used to meet the existing FSAR criteria. The criteria related to control rod clad wear are that the absorber material shall be isolated from the coolant and that the RCCA must be capable of moving freely into the fuel assembly. The impact of absorber clad wear on these criteria for the Ag-In-Cd RCCA was previously considered by Westinghouse by basing the allowable wear on prevention of clad collapse. For conservatism no credit was taken for the support provided by the absorber material. Two dimensional elastic-plastic finite element analyses subsequent to the original Point Beach evaluation have demonstrated that the absorber material can prevent the clad from collapsing in the observed wear geometry. Where the absorber material backs the cladding, the allowable wear depth has been increased by several mils (4-6 mils depending on the RCCA design) while still meeting the existing FSAR criteria.

The second area of improvement is the wear scar photograph interpretation method. Based on test data and a thorough review of the wear phenomenon a new method of estimating wear depth from wear scar width has been developed. The data are derived from out-of-pile tests performed in reactor grade water at reactor operating temperature and pressure and include measurements of wear scar width and maximum wear depth. Results of the application of the new method suggest that our original method of interpreting photographs may yield over estimates of wear depths by several mils at most elevations above the continuous guide area. Because the geometry of the aforementioned tests is not identical to the reactor geometry, an effort is also being made under EPRI sponsorship to obtain hot cell measurements of worn RCCAs to improve our understanding of the RCCA wear phenomenon.

The third area which has been addressed is axial repositioning of RCCAs. If a significant level of wear is observed at your reactor, a redefinition of the axial parking elevation of RCCAs by 2-3 steps can extend the service life of RCCAs for at least one additional cycle. The benefit is derived from spreading the fretting wear over more clad surface, thus reducing the maximum wear depth achieved. An evaluation shows that the RCCAs can be inserted several steps deeper than the normal parking position without adversely impacting operation or safety of the plant. It should be noted that a Technical Specification change may be required. If a change to parking elevations becomes necessary, Technical and licensing support can be obtained through your NFD Fuel Projects Engineer.

With the increase in the wear depth limit and the application of the new photograph interpretation method, the rejected RCCAs at Point Beach Unit 1 appear to be acceptable for continued use. Those RCCAs had operated for 13 years with about 80% availability before the examination.

SKETCH SHEET

MJC 1-30-84

WESTINGHOUSE ELECTRIC CORPORATION

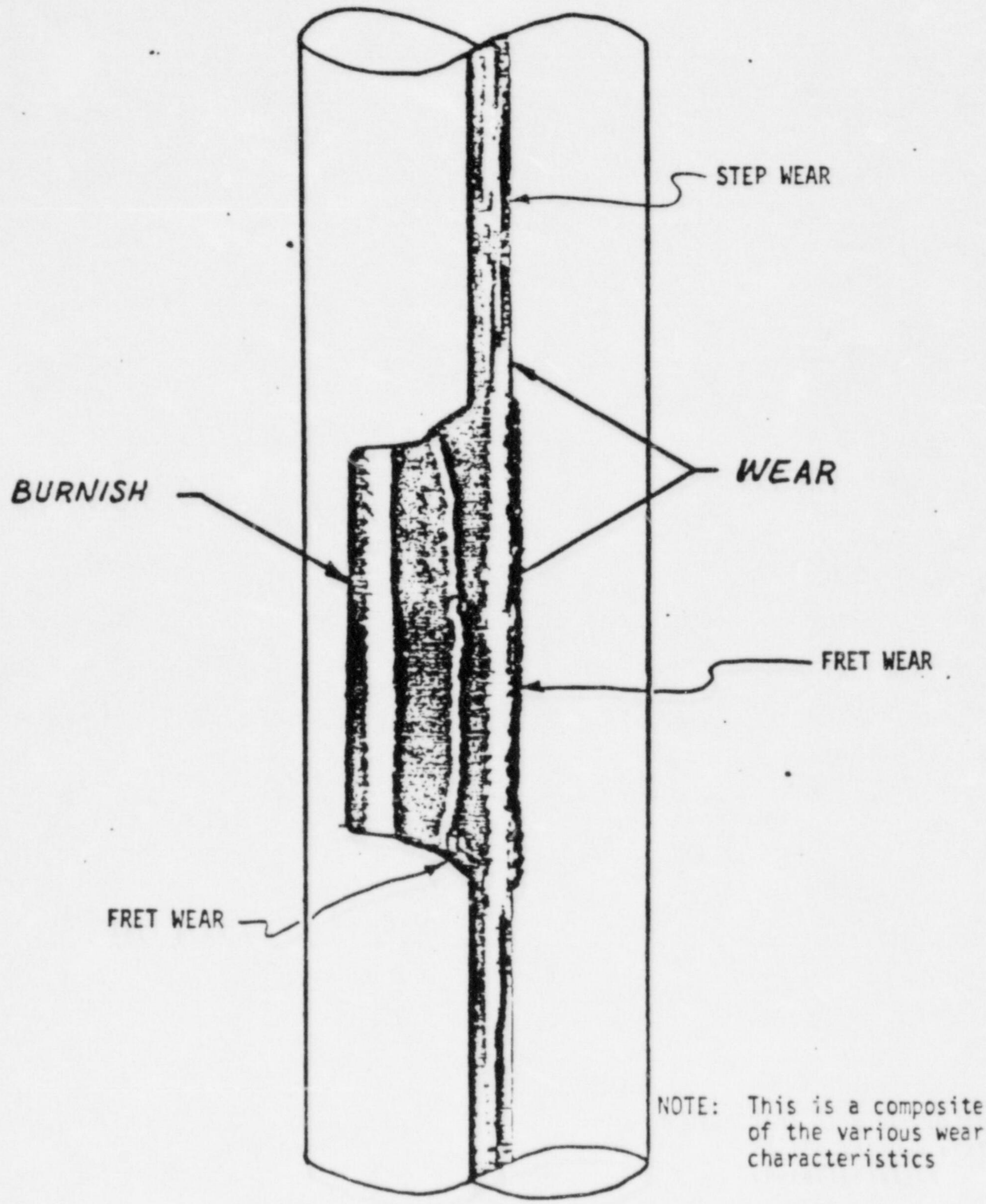
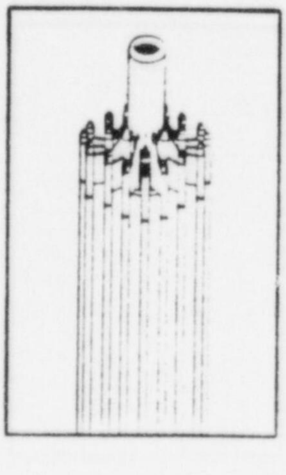


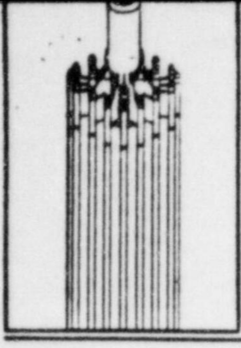
FIGURE 1 - SKETCH OF VARIOUS WEAR CHARACTERISTICS

ATTACHMENT 3

Westinghouse Rod Cluster Control Assembly Performance Review

May 1986

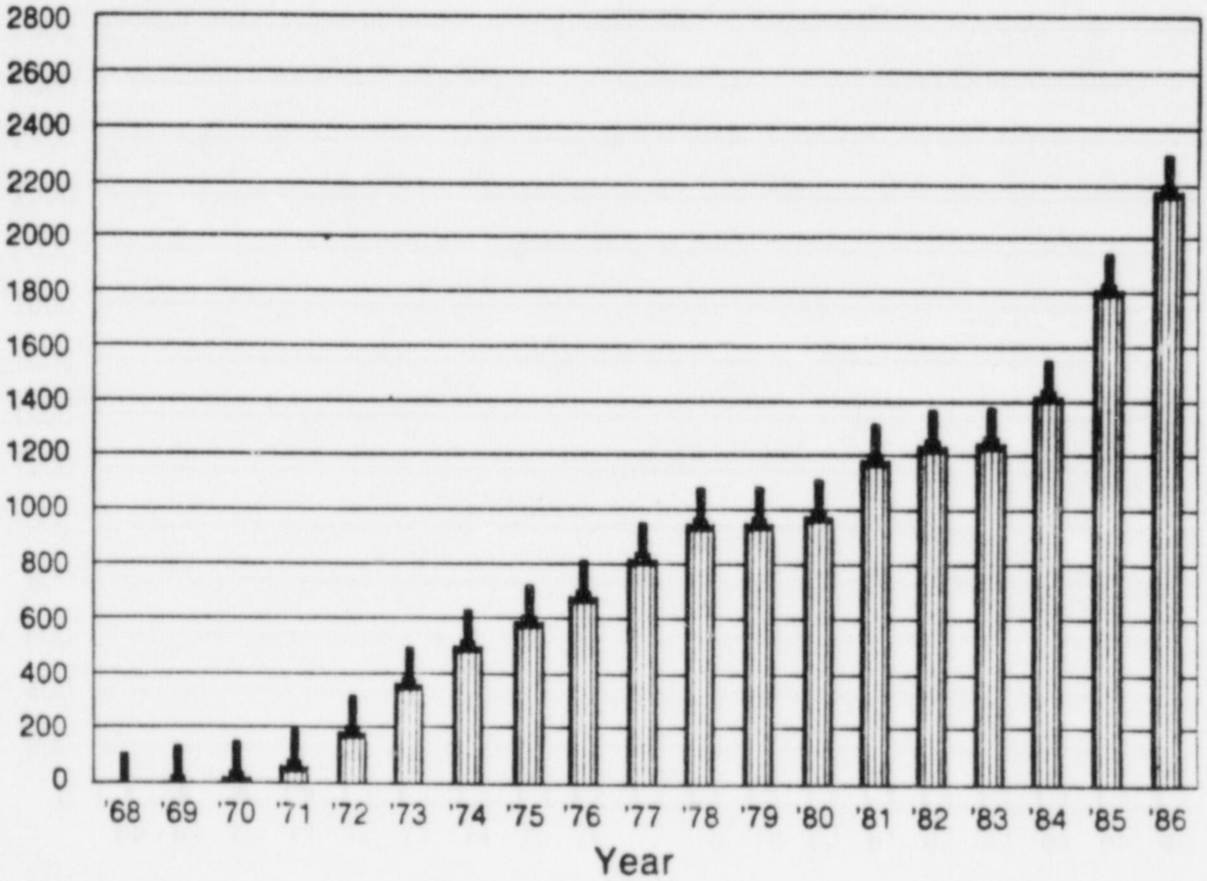


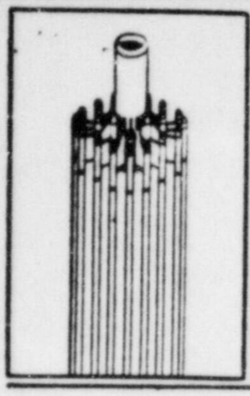


Introduction

As nuclear reactor systems log service beyond a decade, components not subject to periodic maintenance or replacement may be expected to show indications of aging. An ever-increasing number of Westinghouse-supplied PWRs have now been in operation for such extended periods, and some plant operators have observed indications of reduced component integrity. Recently, Rod Cluster Control Assemblies (RCCAs) have shown signs that some wear of rodlet cladding has occurred. In order to obtain a more comprehensive evaluation of these observations, detailed examinations of RCCA rodlets having seen extended service were performed by Westinghouse at its Research Laboratory in Churchill, Pa., under a contract with the Electric Power Research Institute. This report provides a summary of the laboratory examinations and an evaluation of the information obtained.

Cumulative Westinghouse RCCAs in Domestic Service



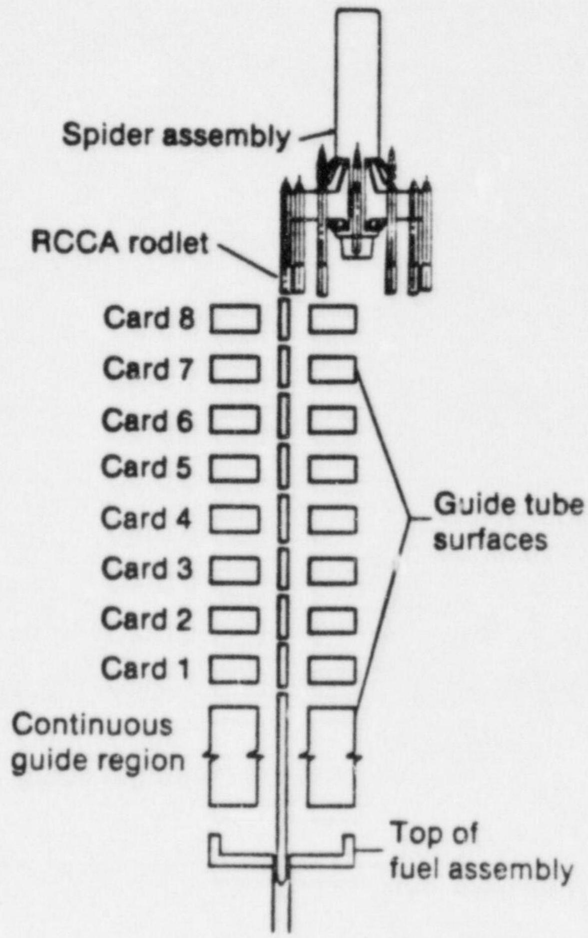


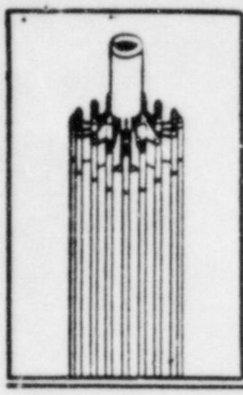
Background

As active components in reactors, control rods are expected to have a lifetime limit in all light water reactors. It is important to monitor and evaluate the operational experience of control rods in order to assess component performance and provide an updated basis for predicting service lifetime. Wisconsin Electric Power Company's Point Beach Nuclear Units 1 and 2 are among the leaders in accumulated service life for Westinghouse-supplied PWRs. After years of successful operation, Wisconsin Electric Power replaced the RCCAs in Point Beach and examinations were performed on the RCCAs taken out of service. Westinghouse, along with Wisconsin Electric Power and EPRI, evaluated selected RCCAs from Point Beach Unit 1 in order to determine a basis for predicting operational lifetime reflecting experience in representative plant operation. A campaign initiated in October 1983 to investigate wear phenomena provided visual and photographic information on 32 RCCAs from Point Beach Unit 1.

One RCCA from Point Beach Unit 1 was selected for detailed examination and testing. Three rodlets were selected from this RCCA for sectioning and detailed testing in the Westinghouse R&D Hot Cell Facility in Churchill, Pa. Ten segments of rodlets were taken for testing. Selection of the specific RCCA and rodlets was based on predetermined criteria including RCCA service history, visual observations of wear, and cracks in rodlet cladding.

RCCA/Upper Internals Layout





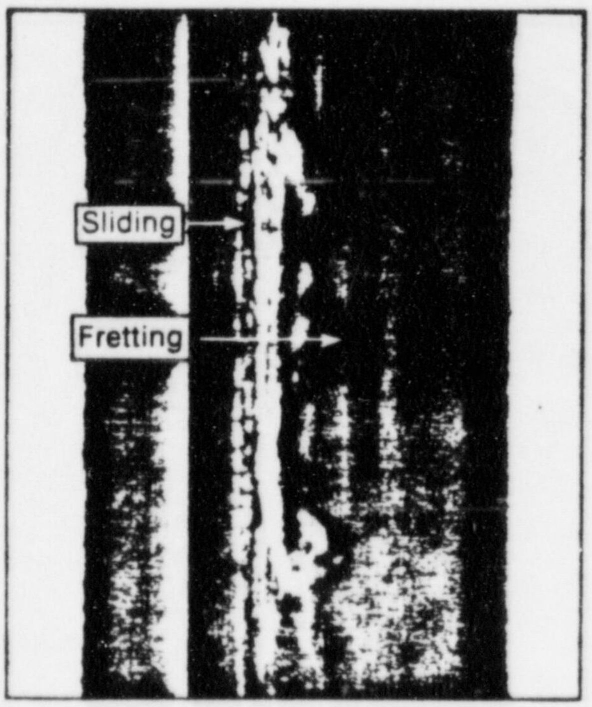
Description of Indications Examined

RCCA Performance

Three potential mechanical performance effects on RCCA cladding were studied.

1. Fretting Wear

Wear spots at specific locations along the length of RCCA rodlets were noted. These locations corresponded to locations of the control rod guide cards in the upper reactor internals when the RCCA would be fully withdrawn from the core (i.e. the parked position). There are eight guide cards in each control rod guide tube, which function to provide lateral support for individual RCCA rodlets when the RCCA is withdrawn. There is also a longer continuous guide directly above the upper core plate.

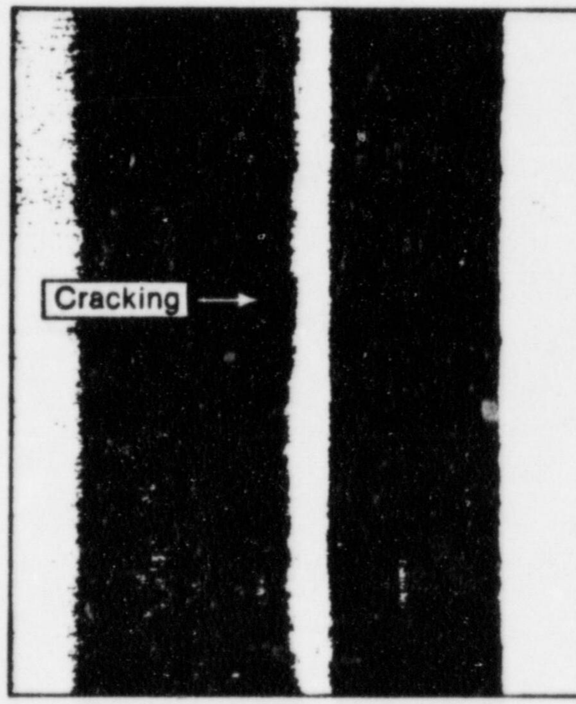


2. Sliding Wear

Wear indications appearing as scratches oriented axially on the surface of the rodlets were noted. These scratches correspond to interaction between RCCA rodlets and the continuous guide, and are the result of control rod withdrawal and insertion movements.

3. Cracking

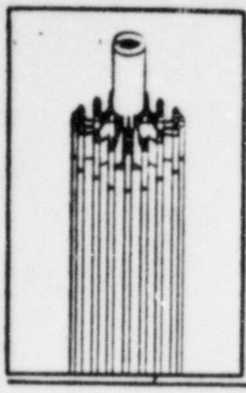
Short hairline cracks at the lower extremity of the cladding were observed on some rodlets. The cracks extended axially for approximately four inches, and penetrated through the cladding.



Service History

The indications were observed after approximately eleven years of "in plant operation". Measurable wear is consistent with expected effects following service of such extended duration.

Wear was originally expected to be life limiting for Westinghouse RCCAs, and a life of approximately fifteen years had been predicted based on sliding wear in conjunction with anticipated plant load change operations.



Laboratory and Engineering Analysis

A. Fretting Wear

Fretting wear is a result of vibrational contact between RCCA rodlets and the guide cards which provide lateral support for the rodlets when RCCAs are withdrawn from the core. Vibration is hydraulically induced by flow of the reactor coolant, and is therefore a continuous process when the reactor coolant pumps are in operation. In general, wear scars resulting from fretting were found to be approximately one-half inch in length and included approximately one-third of the circumference of the rodlet. Depths of clad penetration observed varied in approximate correspondence with the amount of time the RCCAs were in the parked position and location. The degree of wear varied at different guide card elevations. The worst measured depth of clad penetration from fretting wear was approximately 65% of nominal cladding thickness. Since there is a large margin in load-bearing capability for RCCA cladding, the immediate concern if breach were to occur would be deposition of radioactive isotopes into the reactor coolant.

Based on the history of this particular RCCA, a wear rate of approximately 10% of cladding thickness per 10,000 hours reactor operation was calculated.

B. Sliding Wear

The longitudinal scratches observed were determined to be the result of interaction between rodlets and the continuous guide located near the upper core plate. The wear is caused by sliding movement and has two components. One component is attributed to RCCA trips and the other is due to stepping of the RCCA during load change maneuvers.

Wear due to RCCA tripping occurs along the length of the cladding and, for the rodlets examined, averaged less than ten percent of the cladding thickness. Wear due to stepping is most pronounced at the lowest portion of the control rodlet from sliding in the continuous guide region. The combined tripping and stepping wear at the worst location measured less than 20% of the cladding thickness. RCCA contact with the continuous guide causes most of the sliding wear, and sliding wear between the RCCA and guide cards is minimal.

C. Cracking

Several RCCA rodlets experienced hairline cracks in the cladding at the lower extremity of the rodlet. The cracks were typically about four inches long and axially oriented. No circumferential component of cracking was found. The lower extremity of the rodlet typically experiences the highest fluence.

Destructive examination of cracked rodlets was undertaken to identify the probable cause. Both metallurgical analysis and dimensional examinations of the absorber and cladding were performed.

The examinations showed evidence of irradiation assisted intergranular cracking and stresses due to interaction between absorber and cladding. The cladding material was type 304 stainless steel and the absorber was composed of silver (Ag), indium (In), and cadmium (Cd) in proportions of 80, 15, and 5 percent, respectively. Negligible deterioration of the absorber occurred due to the presence of the crack.

Scanning Electron Microscope (SEM) results confirmed the intergranular nature of the cracks. Cladding tensile tests were also performed to correlate mechanical properties with the SEM data. These tests provided evidence of a reduction in cladding ductility consistent with high neutron fluence near the tips.

The tensile tests were performed in an inert environment so as to avoid possible effects due to corrosive agents. Fracture surfaces of both the tensile test samples and the cracked cladding were intergranular and were consistent with other irradiation-induced embrittlement and stress corrosion cracking experience. Based on a number of EPRI programs, such fracture characteristics are considered to be associated with migration of silicon and phosphorus impurities.

Corrosion tests were also performed to determine the extent of intergranular attack. The results provided evidence of irradiation-enhanced segregation of impurities at grain boundaries of the stainless steel cladding.

Dimensional and density measurements confirmed that irradiation-induced swelling of the absorber was the principal cause of tensile stress in the cladding, which resulted in cracking after substantial irradiation.