ATTACHMENT 1

SERIAL: NLS-84-219

BRUNSWICK STEAM ELECTRIC PLANT

PROPOSED TECHNICAL SPECIFICATION PAGE - UNIT 1

(CP&L SERIAL NO: 84TSB23)

(070MAT/cfr)

SUMMARY LIST OF REVISIONS
BRUNSWICK UNIT 1

Page	Comment
5-4	"143 inches" changed to "approximately 143 inches" "or hafnium absorber rods" added

(070MAT/cfr)

5.3 REACTOR CORE

FUEL ASSEMBLIES (Continued)

shall have a maximum enrichment of 2.35 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 2.85 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 137 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing approximately 143 inches of boron carbide, B_4C, powder or hafnium absorber rods surrounded by a cruciform-shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The nuclear boiler and reactor recirculation system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

b. For a pressure of 1250 psig, and

c. For a temperature of $575^{\circ}F$.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 18,670 cubic feet.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown in Figure 5.1.1-1.

ATTACHMENT 2

SERIAL: NLS-84-219

BRUNSWICK STEAM ELECTRIC PLANT

PROPOSED TECHNICAL SPECIFICATION PAGES - UNIT 2

(CP&L SERIAL NO. 84TSB23)

SUMMARY LIST OF REVISIONS
BRUNSWICK UNIT 2

Page	Comment
5-4	"143 inches" changed to "approximately 143 inches" "or hafnium absorber rods" added

(070MAT/cfr)

5.3 REACTOR CORE

FUEL ASSEMBLIES (Continued)

The initial core loading shall have a maximum enrichment of 2.47 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.80 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 137 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing approximately 143 inches of boron carbide, B_4C, powder or hafnium absorber rods surrounded by a cruciform-shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The nuclear boiler and reactor recirculation system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

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5.5.1 The meteorological tower shall be located as shown in Figure 5.1.1-1.

ATTACHMENT 3

SERIAL: NLS-84-219

BRUNSWICK STEAM ELECTRIC PLANT

GENERAL ELECTRIC DOCUMENT NEDO-22290

(CP&L SERIAL NO. 84TSB23)

(070MAT/cfr)

FOR INFORMATION ONLY

NEDO-22290-A
DRF L12-00581
Class I
September 1983

SAFETY EVALUATION OF THE GENERAL ELECTRIC HYBRID I CONTROL ROD ASSEMBLY

Approved: R. J. Brandon, Manager
Nuclear Services Engineering
Operation

Approved: R. L. Gridley, Manager 11/2/83
Fuel and Services Licensing

NUCLEAR POWER SYSTEMS DIVISION • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

GENERAL ELECTRIC

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG [illegible] 1983

Mr. J. F. Klapproth, Senior Engineer
Nuclear Safety and Licensing Operation
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Mr. Klapproth:

Subject: Acceptance for Referencing of Licensing Topical Report NEDE-22290, "Safety Evaluation of the General Electric Hybrid I Control Rod Assembly"

We have completed our review of the subject topical report dated December 1982 and submitted by General Electric Company (GE) letter MFN 096-83. We find this report is acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that GE publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions should incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, GE and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Cecil O. Thomas

Cecil O. Thomas, Chief
Standardization & Special
Projects Branch
Division of Licensing

Enclosure: As stated

Topical Report Evaluation

Report Number:	NEDE-22290
Report Title:	Safety Evaluation of the General Electric Hybrid I Control Rod Assembly
Report Date:	December 1982
Originating Organization:	General Electric
Reviewed By:	Core Performance Branch, Division of Systems Integration

1. SUMMARY OF TOPICAL REPORT

This topical report was originally submitted (Ref. 1) by Philadelphia Electric Company (PECo), licensee for Peach Bottom Atomic Power Station Unit No. 3 (PB-3), in support of a request for a Technical Specification change that would permit operation of Peach Bottom 3 with up to six Type II General Electric (GE) hafnium Hybrid I Control Rods (HICRs). The report actually describes not just one but two types of HICR assembly configurations: Type 1, a production type, and Type II, a surveillance type. As later stated in letters (Refs. 2 and 3) from General Electric and discussed in a March 23, 1983 meeting (Refs. 4 and 5) between representatives of GE, PECo and the NRC staff, PECo's request was for a plant-specific approval of the Type II HICR, while GE wanted a generic approval of the Type I HICR for use in any BWR/2-4 D-lattice plant.

As stated in NEDE-22290, HICRs are meant to be standard replacement control rod assemblies for General Electric BWR/2-4 D-lattice operating reactors. The HICRs are supposed to increase control rod assembly life and eliminate cracking of absorber tubes containing boron carbide (B_4C). The major design changes that are intended to ensure that those objectives are met are (1) the use of an improved B_4C absorber rod tube material to eliminate stress corrosion cracking during the lifetime of the assembly and (2) replacement of some B_4C absorber rods with solid hafnium absorber rods. In addition, there are other material and dimensional changes including a reduction in sheath wall thickness and changes in the pin and roller materials.

2. REGULATORY EVALUATION

Our review of the Type II (surveillance) HICRs for Peach Bottom 3 Reload 5 culminated in approval (Ref. 6) of proposed Technical Specification changes allowing up to six Type II HICRs for Cycle 6 operation. In our review of the surveillance HICRs for the Peach Bottom 3 reload, the key issues involved the potential effects of changes in component materials and dimensions. Thus, we addressed proposed changes in pin and roller materials and control rod tubing material as well as the introduction of hafnium absorber rods in place of B_4C. This latter change resulted in increases in absorber material weight, which had to be offset by a reduction in blade sheath thickness; it also raised some questions about (a) the thermal expansion of Hf relative to the absorber cladding material and (b) the corrosion resistance of unclad Hf. For each of these technical issues we concluded (Ref. 6) that the information supplied in the NEDE-22290 topical report, supplemented by additional information that was obtained in the March meeting with representatives of GE and PECo (Refs. 4 and 5), and complemented by the HICR surveillance program provided reasonable assurance that the six Type II Hybrid Control Rods were acceptable from a mechanical, nuclear and thermal-hydraulic design standpoint.

With regard to the generic review of the Type I production HICRs, the primary technical issues are essentially the same as for the surveillance HICRs except as affected by the use of certain different materials for the two types of blades. For instance, in the Type I production blades, all the hafnium rods will be unclad. Because we had for the surveillance blades already completed (Ref. 6) an extensive review that had encompassed the technical issues common to both types (production and surveillance) blades, our generic review of the production blades was abbreviated and restricted to a few special topics, including the corrosion of hafnium in a BWR core environment and surveillance. For a discussion of our safety evaluation of the features common to both types of blades, as well as those features unique to the surveillance HICRs, the reader is directed to Reference 6. The relevant portion of that document is included as Appendix A to this SER. The following discussion is limited to the few special concerns for Type I production blades.

In the section of NEDE-22290 that deals with the chemical, physical, and mechanical properties of hafnium, there is a short paragraph (Section 4.1.1) that addresses hafnium corrosion in high temperature water. It is concluded that acceptable corrosion resistance in high temperature water and steam exists for hafnium in BWR control blade applications. That conclusion is based primarily on open literature data and experience to date on an experimental, bare, hafnium control rod used in Peach Bottom Unit 2 for one cycle of operation. Because the amount of information available on the corrosion of hafnium in a BWR environment was sparse (there is singificantly more information available on PWR use of hafnium, due to naval reactor applications), hafnium corrosion was a major topic of discussion in the March 1983 meeting with GE and PECo. GE presented (Refs. 4 and 5) considerably more data that showed that hafnium has better corrosion resistance than Zircaloy-2 in high temperature water. Much of the hafnium corrosion data that has been obtained so far, however, has been on high-purity crystal bar hafnium, not on commercial grade material. Furthermore, corrosion in BWR flowing coolant and steam, and in an irradiation field, cannot be duplicated in out-of-pile tests. We believe, therefore, that the ongoing and planned test and surveillance programs, which include the Peach Bottom 2 unclad hafnium test rods and the unclad rods in the Peach Bottom 3 Type II surveillance HICRs, will provide important confirmatory evidence of the satisfactory corrosion resistance of hafnium in GE control blades. We expect to be informed of the results of these and other test and surveillance programs as those results relate to the potential performance of the production version HICRs.

Because of the importance of the test and surveillance programs with respect to providing both lead exposure information and confirmation of predicted performance, we asked for more detail regarding schedules and types of examinations to be performed. Further information was provided in Reference 7. A list of the HICR irradiation programs is provided in Table I, and the examination schedule runs from 1982 through 1990. As shown, the various features of the HICRs, including the new pin and roller materials, the new B_4C absorber rod cladding material, and both the clad and unclad hafnium absorber rods, are being tested at several BWRs. Interim and final results from those tests will be obtained over the next few years.

Two key programs involve the unclad test rods in Peach Bottom 2 and the production rods in the first two (yet unnamed) non-control cell commercial BWRs. The Peach Bottom 2 lead test rods will provide confirmatory information on the performance of unclad hafnium absorber rods. This is a relatively near-term program that will be completed about 1985. A more long-term program will be carried out in the first two non-control cell plants to use Type I production HICRs. In the non-control cell plants, which are for the most part BWR/2s and 3s, and some 4s, the burnup of the control material is relatively slow. Therefore, data can only be obtained after exposures greater than 6 years, beyond the time at which the more short-term test programs in Peach Bottom 2, Quad Cities 1, etc., will be completed. The long term surveillance program will thus provide information on environmental effects for extended residence times. Since all of the other surveillance and test programs will provide data on the HICRs for up to 6 years of operation, the first visual (e.g., boroscope or T.V. camera) examinations will commence on the 6th year of operation of the first two non-control cell plants with Type I HICRs. Subsequent examinations will occur every other cycle thereafter (Ref. 7). We conclude that this is an acceptable surveillance program that, coupled with the other test programs listed in Table I, meets the intent of Standard Review Plan Section 4.2.

3. SUMMARY

We have completed our generic review of the General Electric topical report, NEDE-22290, that describes the GE hybrid control rod (HICR) designs. Because we recently completed review of the Type II (surveillance) HICRs intended for insertion in Peach Bottom Unit 3, we were able to perform an abbreviated review of NEDE-22290. We thus concentrated our generic review of the Type I (production) HICRs on certain technical issues especially important to the use of Type I blades, such as corrosion of unclad hafnium absorber rods and surveillance. Based on our evaluation of the information provided in (a) NEDE-22290, (b) a meeting with GE representatives, and (c) responses to NRC staff questions, we conclude that there is reasonable assurance that the substitution of Type I HICRs for other approved GE control blades will not result in unacceptable hazards to the public and should, in fact, result in improved control blade performance and a positive contribution to reactor safety. Therefore, NEDE-22290, as amended to incorporate this safety evaluation, is approved as a referential document for the GE Type I HICR.

TABLE 1

SUMMARY OF HYBRID IRRADIATION PROGRAMS

PLANT	TYPE	INSERTION	QTY	EXAM
MONTICELLO	PINS & ROLLERS	2/80	2	VISUAL & DESTRUCTIVE
PEACH BOTTOM 2	UNCLAD HF TEST RODS	3/80	2	VISUAL & DESTRUCTIVE
MILLSTONE	PINS & ROLLERS		2	VISUAL & DESTRUCTIVE
QUAD CITIES 1	HIGH PURITY TYPE 304 PRODUCTION RODS		20	RADIOGR[illegible] DESTRUCTIVE
PEACH BOTTOM 3	HYBRID SURVEILLANCE RODS (HP 304, BARE HF) (HP ALLOY 600, ZR-2 CLAD HF)	4/83	8	VISUAL & DESTRUCTIVE
ADDITIONAL PLANT	HYBRID PRODUCTION (HP 304, BARE HAFNIUM)	1984	1	VISUAL
ADDITIONAL PLANT	HYBRID PRODUCTION (HP 304, BARE HAFNIUM)	1984	1	VISUAL

<u>References</u>:

1. E. J. Bradley (PECo), letter to H. R. Denton (NRC), December 30, 1982.

2. A. N. Tschaeche (GE), letter to M. Tokar (NRC), "Revision of Certain Pages of NEDE-22290," April 13, 1983.

3. J. F. Klapproth (GE), letter to Cecil Thomas (NRC), Subject: "Request for Generic Approval of NEDE-22290," May 20, 1983.

4. A. N. Tschaeche (GE), letter to M. Tokar, (NRC), with GE proprietary document titled "Hybrid Control Rod Licensing Presentation to the Nuclear Regulatory Commission, March 23, 1983," March 24, 1983.

5. Gerald E. Gears (NRC), memorandum titled "Summary of the March 23, 1983 meeting with PECo Pertaining to GE Hybrid Control Rods and Unit 3 Reload," April 22, 1983.

6. L. S. Rubenstein (NRC), memorandum for G. C. Lainas, "Peach Bottom 3 Reload 5 Cycle 6 (TACS 49306)," April 18, 1983.

7. J. F. Klapproth (GE), letter to M. Tokar (NRC), July 12, 1983.

objectives are met are (1) the use of an improved B_4C absorber rod tube material to eliminate stress corrosion cracking during the lifetime of the assembly and (2) replacement of some B_4C absorber rods with solid hafnium absorber rods. In addition, there are other material and dimensional changes, including a reduction in sheath wall thickness and a change in the pin and roller materials from Stellite to other materials discussed in reference 13. Other variables included the location of the hafnium rods, the type of tubing used for B_4C rods and the use of clad versus unclad Hf.

Due to the complexity of the HICR test program (as evidenced by the large number of variables to be examined), a meeting (Ref. 4) was held with GE and PECo to discuss the program in Peach Bottom 3 as well as the overall R&D program, analyses, surveillance, etc. performed or underway by GE in support of the HICR design. The purpose of the meeting was actually two-fold:

1. To support the proposed amendment to the Peach Bottom 3 operating license to permit HICR use.
2. To support the generic use of HICRs in BWRs.

Because the generic review is much broader in scope than could be accommodated by the tight schedule required for Peach Bottom 3 Reload 5, this safety evaluation addresses only the issues involving the six surveillance HICRs. The results of the generic review will be reported elsewhere as a safety evaluation of the GE topical report, NEDE-22290 (Ref. 3).

With regard to the Peach Bottom 3 Cycle 6 use of the six Type II HICRs, the key issues concerned the potential effects of the changes in component materials and dimensions. The safety considerations involved are discussed below for each design change.

<u>Pins and Rollers</u> - As indicated in EPRI NP-2329 (Ref. 13), the pin and roller materials currently in use in BWRs are cobalt-base alloys (Haynes 25 and Stellite 3, respectively). Because cobalt-60 is an isotope that contributes significantly to plant radiation buildup, there is an incentive to replace

the cobalt alloys with non-cobalt alloys and thus reduce personnel radiation exposure during plant maintenance. In EPRI NP-2329 is described an extensive program at GE to qualify substitute non-cobalt alloy control rod pin and roller materials. Wear resistance measurements in a simulated BWR environment (excluding irradiation), coupled with impact strength an corrosion tests, indicate that the non-cobalt alloys have equivalent or better wear resistance, superior impact strength and similar corrosion resistance to the conventional cobalt alloys. Though the effects of irradiation were not investigated in those tests, reactor tests have been initiated at a control cell BWR and at a conventional core BWR. We conclude that the substitution of the non-cobalt alloys for Haynes 25 and Stellite 3 pins and rollers in the six Type II surveillance HICRs is acceptable, based on the results of the tests described in EPRI NP-2329 and our expectation that (a) the surveillance described on page S-5 of EPRI NP-2329 will be carried out, (b) the results of that surveillance will be reported in a timely fashion, and (c) surveillance of the six HICRs in Peach Bottom 3 will also be conducted and reported.

Control Rod Tubing Material - As indicated on page 2-2 of NEDE-22290 (Ref. 3), the B_4C absorber rod tubing for the Type I (production version) control rods is a high purity Type 304 stainless steel, while the Type II (development) control rods will also contain some high purity Inconel 600 as an alternate absorber tube material. Both of these alloys have undergone extensive qualification testing and evaluation including laboratory testing, correlation of field performance with intergrannular stress corrosion cracking susceptability tests, and assessment of archival materials. In addition, an extensive surveillance program, including visual examinations, dimensional measurements, eddy current testing, neutron radiography, isotopic determinations, and steam corrosion testing (see p. 5-10 of Ref. 3) is planned. Based upon the information provided in Ref. 3 and in the meeting described in Ref. 9, we conclude that the use of the new absorber tube alloys is acceptable for the six Type II HICRs. We expect to be informed of the results from the HICR surveillance program on the absorber tube materials as those results relate to the potential performance of the production version HICRs.

Absorber Material - As indicated in NEDE-22290 (Ref. 3), three of the B_4C absorber rods per blade (12 in each control rod assembly) in the present BWR 2-4 D lattice CRA design will be replaced with solid hafnium rods. In the Type I production version HICRs the Hf rods are unclad and located at the tip positions of each blade. The three main concerns related to the use of Hf rods involve (a) the increase in weight, (b) the thermal expansion of Hf relative to the absorber cladding material, and (c) the corrosion resistance of unclad Hf.

With regard to the increased weight resulting from the higher density of Hf relative to the B_4C it replaces, the reduction in blade sheath thickness (and weight) compensates for the increase in absorber material weight. The resultant sheath thickness falls within the range of GE design experience, and the increased fuel channel clearance should reduce potential fuel channel interference. From a mechanical design standpoint, therefore, there is reasonable assurance that the design changes related to the increased weight of the absorber material have been adequately accounted for in the six Type II HICRs. The planned surveillance of the HICRs should provide confirmation of this.

With regard to the thermal expansion and irradiation growth considerations, the coefficient of thermal expansion of Hf is approximately half that of 304 SS and Inconel 600 (the B_4C absorber tubing materials), and is comparable to an alternate cladding material used for some of the Hf rods in the Type II HICRs. Inasmuch as only a few Type II rods will have the alternate cladding material, any adverse effects, which are not anticipated, should not be significant. The irradiation growth of hafnium is expected to be small. Bare hafnium absorber rods in the Peach Bottom - 2 reactor have shown virtually no change in length or diameter after 18 months service. Since dimensional measurements will be made of the Hf rods at 18-24 month intervals as part of Type II HICR surveillance program, the irradiation growth will readily be monitored.

With regard to the corrosion of hafnium in a BWR environment, there is significantly more information regarding PWR use of hafnium (because of naval reactor use). GE did present some data (Refs. 3 and 4), however. Those data showed that the corrosion behavior of hafnium in high temperature water and steam is superior to that of Zircaloy-2. In addition, an experimental, bare Hf control rod in Peach Bottom-2 has shown little corrosion after 1.5 years exposure (Refs. 14 and 15). The planned Type II HICR surveillance program is intended to include metallographic examinations of the Hf rod hydriding behavior and corrosion characteristics. We conclude, therefore, that the corrosion behavior of the Type II HICR Hf rods has been adequately addressed for PB-3 cycle 6 operation.

2.7 Fuel System Design Conclusions

We have reviewed the information submitted on the cycle 6 operation of Peach Bottom Unit 3, including the design, analysis, testing, and proposed surveillance of a pressurized test assembly, four lead test assemblies, and six Type II Hybrid Control Rods. We find the PB-3, reload 5 proposed refueling and related Technical Specification changes acceptable from a mechanical design standpoint.

3.0 Nuclear Design

The nuclear design of the proposed reload was performed by the approved methods of reference 8 including that of the lead test assemblies. The nuclear parameters for the reload are within the range of those normally seen for BWR reloads and are acceptable.

4.0 Thermal and Hydraulic Design

The objective of the thermal-hydraulic review is to confirm that (a) the thermal-hydraulic design of the core has been accomplished using acceptable methods, (b) the design provides an acceptable margin of safety from conditions which could lead to fuel damage during normal and anticipated operational transients, and (c) the design is not susceptible to thermal-hydrualic instability.

ABSTRACT

The General Electric Hybrid I Control Rod Assembly (HICR) is designed to increase blade life and to eliminate absorber rod tube cracking during assembly lifetime. The HICR mechanical and nuclear functions are identical to that of the General Electric BWR 2-4 D lattice Control Rod Assembly. This report describes the design, and evaluations and analyses performed to demonstrate the safety of the HICR.

FOREWORD

This Licensing Topical Report is intended to aid reactor licensees in obtaining revisions to technical specifications so that the HICR can be used as a replacement for the all-B_4C control rod assemblies presently in use.

The first HICR's are planned for installation in the Peach Bottom 3 reactor, prior to Cycle 6. The results of the safety analysis in the Peach Bottom 3 reload 5 licensing amendment submittal (Y1003J01A54, December 1982) are unchanged for the use of HICR in reload 5 core during cycle 6 operation.

The effect of the HICR on reload analyses is generic in nature. Therefore, all BWR supplemental reload licensing submittals are unchanged if HICR's are used.

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1. INTRODUCTION AND SUMMARY

1.1 Introduction

This report describes the design and safety evaluation of the General Electric Hybrid I Control Rod Assembly (HICR). HICR's are intended to be standard replacement control rod assemblies for the General Electric BWR 2-4 D lattice in operating reactors. The HICR's form, fit and function are identical to that of the blade it replaces. The HICR is designed to increase control rod assembly life and to eliminate cracking of absorber tubes containing boron carbide (B_4C). The essential differences between the new HICR and the BWR 2-4 D lattice control rod assemblies currently in use are:

a) improved B_4C absorber rod tube material to eliminate cracking during the lifetime of the control rod assembly, and

b) some B_4C absorber rods are replaced with solid hafnium absorber rods to increase blade life.

Other minor material and dimensional changes are described in detail later in this report.

1.2 Summary

The design description and analyses in this report demonstrate that the HICR satisfies the performance and safety requirements for use as a direct replacement for the BWR 2-4 D lattice control rod assembly. The design bases for the HICR are given in Section 3.

Section 4 describes the physical, chemical and irradiation properties of the HICR materials. Section 5 sets forth the HICR design evaluation including mechanical, nuclear, thermal, hydraulic performance, prototypical tests, and surveillance.

Section 6 contains evaluations showing that scram speed, scram reactivity linear heat generation rate (LHGR), maximum critical power ratio (MCPR) and maximum average planar linear heat generation rate (MAPLHGR) design limits are not affected when the HICR is used in BWR cores.

2. DESIGN DESCRIPTION

2.1 Configuration

The HICR design configuration is identical to that of the present BWR 2-4 D lattice control rod assembly shown in Figure 2-1. As shown in Figure 2-2, three solid hafnium absorber rods replace the B_4C absorber rods in each wing. In the production version, Type I, the hafnium absorber rods are in the tip positions, * The reason for this placement is explained in Section 2.3.

The total HICR weight is the same as that of the assembly it replaces. The weight is the same because the sheath wall thickness is reduced from to offset the increased weight of the hafnium rods. The sheath wall thickness is within GE's experience base with sheaths on operating reactor control rod assemblies that include the sheath wall thickness used in the present BWR 5 control blades and the thickness for the BWR 6 control rod assemblies. The sheath wall thickness reduction results in an overall blade width reduction of thus increasing the clearance between the HICR and fuel channels without compromising structural integrity. Because there are no significant design configuration or envelope differences, the HICR is directly interchangeable with the present BWR 2-4 D lattice design control rod assemblies and is therefore compatible with existing NSSS hardware.

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*General Electric Company Proprietary Information has been deleted.

2.2 Materials

2.2.1 Pins and Rollers

The pin and roller material will be changed from respectively. This design improvement eliminates cobalt-bearing stellite material. The qualification of these materials is based on the results of an extensive Electric Power Research Institute (EPRI) program at GE to determine alternate materials for control rod use. Results of this program, for application to pin and rollers in a BWR environment.

2.2.2 Control Rod Tubing Material

The B_4C absorber rod tubing for the Type I (production version) control rods is a high purity Type 304 stainless steel. The Type II (development) control rods, will also contain some high purity Inconel 600 as an alternate absorber tube material. Both of these materials have undergone extensive qualification testing that has confirmed their improved resistance to intergranular stress corrosion cracking (IGSCC) when compared to the presently used commercial grade type 304 stainless steel.

2.3 Absorber Material

Three of the B_4C absorber rods per blade (12 in each control rod assembly) in the present BWR 2-4 D lattice control rod assembly design will be replaced with solid hafnium rods. These replacement rods are located at the tip positions of each blade in the production version (Type I, see Figure 2-2). A few surveillance HICR's (Type II) will be constructed with the hafnium rods

(see Figure 2-2). Post-irradiation examination of these Type II assemblies will be used to obtain data on the new B_4C rod material properties at high B_4C burnup (the 3 tip positions). Hafnium rods are used to maintain the weight of these assemblies identical to that of Type I configuration and to obtain additional corrosion and irradiation growth data in the BWR water environment.

A

limited number (approximately 4) of the HICR's will be fabricated with the Hafnium absorber rods

resistance of those rods. As part of the surveillance program (see paragraph 5.4), would be inspected after about 1.5 and 3 years of operation.

Typical HICR parameters are given in Table 2-1. For comparison, the parameters of the currently-used control rod assemblies are included in that table.

TABLE 2-1

TYPICAL PARAMETERS

CURRENT CONTROL ROD ASSEMBLY AND

HYBRID I CONTROL ROD ASSEMBLY

(GE Company Proprietary)

		HICR	
	Current Assembly	Production	Surveillance
Control Rod weight, pounds	218	218	218
Absorber Rod - B_4C			
Number per control rod	84	72	72
Length (inches)	143	143	143
Diameter (inches)	0.138	0.138	0.138
Density (grams/cm³)	1.76 (Nominal, 70% theoretical)	1.76 (Nominal, 70% theoretical)	1.76 (Nominal, 70% theoretical)
Absorber Rod - Hafnium			
Number per Control Rod	0	12	12
Length (inches)	-	143	143
Diameter (inches)	-	0.188	0.188
Density (grams/cm³)	-	13.3	13.3
Type	-		
Absorber Rod (B_4C) Cladding			
Material	Commercial 304 S.S.	High Purity 304 S.S.	High Purity 304 S.S.′ and
Outside Diameter (inches)	0.188	0.188	0.188
Wall thickness (inches)			
Sheath thickness (inches)			
Pin Material			
Roller Material			

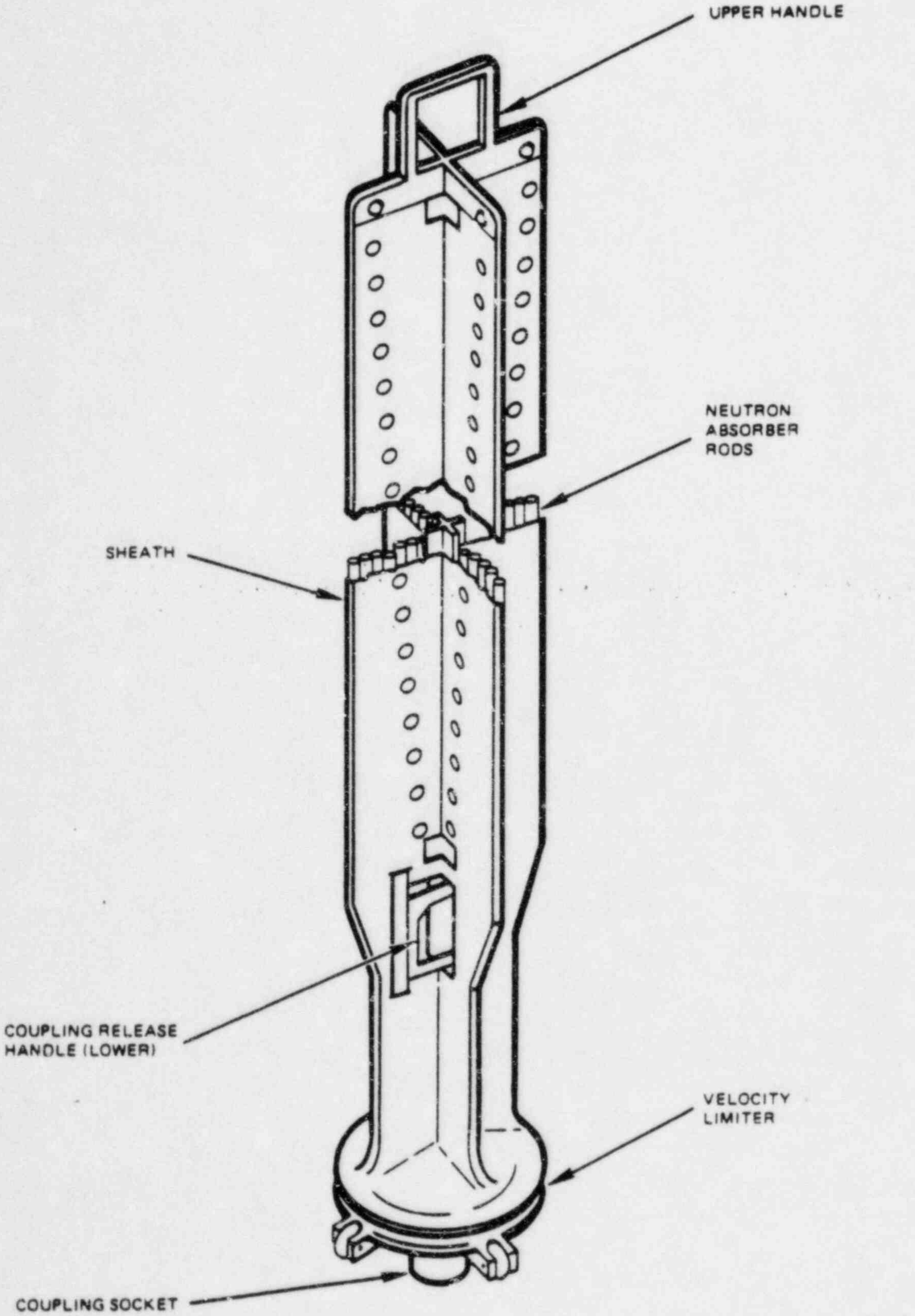

Figure 2-1. Typical Control Rod Assembly

Figure 2-2. Absorber Rod Configuration
(GE Company Proprietary)

3. DESIGN BASIS

The HICR is designed to meet the following criteria:

(a) Mechanical, hydraulic and thermal performance shall be equal to the present product line control blades

(b) The HICR geometry shall not be changed in such a manner as to alter current interface relationships with fuel channels, fuel support castings, guide tubes and control rod drive mechanisms.

(c) The HICR shall be designed to function in a reactor environment with the temperature and pressure transients encountered during normal, upset, emergency and faulted conditions. Stresses and deformations resulting from those transients shall be considered when determining the acceptability of the design.

(d) The HICR shall be designed to withstand the loading encountered during handling, shipping, anticipated activities during reactor operation and environmentally induced loads.

(e) The mechanical life of the HICR is reached either when 1) the internal helium pressure from the Boron-10 (neutron, alpha) reaction results in

resulting in irradiation induced B_4C swelling and stresses in the absorber rod tubing.

(f) The nuclear life of the HICR is reached when the boron depletion results in a 10 percent loss in relative control rod worth ($\Delta K/K$) in the top quarter section of the blade.

4. MATERIALS EVALUATION

4.1 Chemical, Physical, and Mechanical Properties of Hafnium

4.1.1 Corrosion Properties in High Temperature Water

Available data in the literature, as well as experimental data obtained by the General Electric Nuclear Energy Operation, demonstrate that the corrosion behavior of hafnium in high temperature water and steam is superior to Zircaloy-2 as shown in Figure 4-1 and Table 4-1. Pertinent corrosion data are shown in Figures 4-2 to 4-5 and in Tables 4-2 to 4-5. In addition, an experimental, bare hafnium control rod used in Peach Bottom-2* for 1.5 years showed little evidence of corrosion The data and experience demonstrate that acceptable corrosion resistance in high temperature water and steam exist for hafnium in BWR control rod applications.

4.1.2 Physical Properties of Hafnium

A compilation of physical properties of hafnium that are expected to be germane to control rod application is given in Table 4-6. These data indicate acceptable performance for Hafnium in the BWR environment.

*See NEDO-24231, Revision 1 "Proposed Peach Bottom Atomic Power Station Unit 2 Alternate Absorber Control Blade Test Program," January, 1980.

4.1.3 Mechanical Properties of Hafnium

A. Unirradiated

The tensile properties of hafnium are approximately equivalent to those of Zircaloy-2 and are adequate for control rod application. A summary of mechanical properties of hafnium that are pertinent to absorber rod application is given in Table 4-7. Note that the strength properties of unirradiated hafnium exceed those of annealed, Type 304 stainless steel, and the ductility of that hafnium is somewhat lower.

4.1.4 Irradiated Properties of Hafnium

Available data combined with recent studies of 11 hafnium absorber rods from Peach Bottom-2 indicate that

The post-irradiation properties are summarized in Figures 4-6 and 4-7 as follows:

1)

2)

3)

4)

5)

4.1.5 Conclusion

The available evidence demonstrates that the chemical, physical, and mechanical properties of hafnium are acceptable for BWR service.

4.2 Absorber Rod Tubing Materials

The high purity Type 304 SS chemistry selected was based on both field experience and laboratory test data.

TABLE 4-1

CORROSION OF HAFNIUM - CRYSTAL BAR

(Weight Increase in mg/dm^2)

(GE Company Proprietary)

TABLE 4-2

CORROSION OF HAFNIUM

Days -- mg/dm²

	No.	7	14	28	42	46	70	84	98	112	140	163	168	191	196
540°F Water	1	1.4	2.0	1.7	1.7	2.0	2.6	-	2.3	2.8	2.3	2.0		2.8	
	5	1.1	1.7	2.3	2.3	1.7	2.3	-	2.6	2.3	2.3	2.3		2.8	
	9	0.9	1.4	1.7	1.4	1.7	1.1	-	2.3	2.0	2.3	1.7		2.8	
	11	0.6	1.4	1.4	1.1	1.1	0.9	-	1.7	0.9	1.1	1.1		2.0	
	Avg.	1.0	1.6	1.8	1.6	1.6	1.7	-	2.2	2.0	2.0	1.8		2.6	
680°F	2	2.8	4.0	4.8	4.8	5.1	5.7	-	6.8	7.7	6.5		6.8		8.3
	6	2.3	3.7	4.6	4.9	4.6	5.4	-	5.7	6.8	6.0		6.8		8.0
	7	2.8	3.4	4.3	4.9	4.8	5.4	-	6.3	7.4	6.3		6.8		8.0
	12	2.0	3.4	4.3	4.0	4.3	4.6	-	5.4	6.8	6.0		6.0		7.4
	Avg.	2.5	3.6	4.5	4.7	4.7	5.3	-	6.1	7.2	6.2		6.6		7.9
750°F	3	4.5	7.7	13.1	14.5	17.4	20.8	26.5	30.8						
	4	4.8	8.5	12.8	14.2	16.4	19.8	25.5	29.8						
	8	3.7	7.7	12.8	13.9	16.7	20.1	26.1	30.9						
	10	3.1	7.1	11.4	11.9	14.5	17.9	24.7	29.0						
	Avg.	4.0	7.8	12.5	13.6	16.3	19.7	25.7	30.1						

TABLE 4-3

CORROSION PROPERTIES

(GE Company Proprietary)

TABLE 4-4

Oxide Film Thickness of Irradiated Hafnium

(GE Company Proprietary)

TABLE 4-5

MEASUREMENTS OF HAFNIUM SAMPLES CORROSION TESTED

FOR 24 HOURS AT 500°C IN 1500 PSI STEAM

(GE Company Proprietary)

TABLE 4-6
PHYSICAL PROPERTIES OF HAFNIUM

(GE Company Proprietary)

TABLE 4-7

HAFNIUM MECHANICAL PROPERTIES (UNIRRADIATED)

(GE Company Proprietary)

Figure 4-1. Comparison of the Corrosion Behavior of Hafnium and Zircaloy
(GE Company Proprietary)

Figure 4-2. Corrosion Properties of Hafnium in High Temperature Water and Steam (GE Company Proprietary)

Figure 4-3. Weight Gain Versus Nitrogen Content for Hafnium in 680°F Water at Saturation Pressure
(Numbers on curves refer to exposure time in days)
(GE Company Proprietary)

Figure 4-4. Weight Gain Versus Nitrogen Content for Hafnium in 700°F Steam at 1,500 psi
(Numbers on curves refer to exposure time in days)
(GE Company Proprietary)

Figure 4-5. Corrosion of Hafnium
(GE Company Proprietary)

Figure 4-6. Effect of Fast Neutron Exposure on Tensile Properties of Hafnium, 300°F Tests (GE Company Proprietary)

Figure 4-7. Effect of Fast Neutron Exposure on Tensile Properties of Hafnium, 600°F Tests (GE Company Proprietary)

5. DESIGN EVALUATION

5.1 Mechanical Evaluation

5.1.1 Thermal Expansion and Irradiation Growth of Hafnium Rods

The coefficient of thermal expansion of hafnium is approximately one half of that of 304 stainless steel and Inconel 600. Calculations, using conservative heat generation rates, thermal conductivities and heat transfer coefficients, show that

Therefore, the thermal expansion of the hafnium rods will be less than the B_4C absorber rod tubing, and will not interfere with the handle and velocity limiter.

The irradiation growth characteristics of Type 304 stainless steel are known to be very minor at the low control rod temperatures and neutron fluences ($<10^{22}$n/cm^2) that will be experienced by the HICR's. Irradiation growth of hafnium is expected to be small.

5.2 Nuclear Evaluation

5.2.1 Reactivity Worth

The reactivity worth of the HICR was calculated using

is a neutron transport Monte Carlo program and is primarily designed for use in calculating nuclear parameters associated with thermal reactors. In general, can solve eigenvalue, fixed source and neutron shielding problems.

The neutron flux, isotopic reaction rates, group-averaged cross-sections, and leakages are calculated in three-space dimensions and over the energy range from 0 to 10 MeV. The reaction types considered are fission, capture, inelastic and n-2n scatter, elastic scattering with isotropic or anisotropic angular distributions, and thermal elastic scattering. The energy distribution of the neutrons is continuous; however, the cross-sections are averaged over up to 2000 microscopic energy groups. There are special provisions so that resonance cross-sections can be calculated by the code for each neutron energy using the Doppler-broadened Breit-Wigner single-level formula. The isotopic material cross-sections are processed from the ENDF/B format tapes.

input geometry model represented an infinite array of controlled fuel in a reactor core geometry. No axial leakage was considered. The same reload fuel assembly was used in each calculation and only the control rod assembly parameters, moderator density, and temperatures were changed for the comparison calculations to the standard all B_4C control rod. For all calculations the HICR geometry contained three solid hafnium rods in the outer position of each control rod assembly wing. The results of the comparison calculations between the HICR and the all-B_4C control rod assembly are summarized in Table 5-1.

Therefore, the HICR will have no significant impact on core and fuel operation when used as a replacement for the current standard all-B_4C control rod assemblies.

5.2.2 Experimental Results

A series of critical experiments utilizing a current standard B_4C control rod assembly and an all-hafnium control rod assembly were performed at the Nippon Atomic Industry Group (NAIG) critical facility. The purpose was to provide critical experiment data for benchmarking the Monte Carlo program with hafnium cross-section libraries and to obtain measurements of the key nuclear parameters namely: the critical water heights, the rod-by-rod fission distribution in the fuel bundle adjacent to the control rod assembly, and the thermal neutron flux profile adjacent to the control rod assembly surface.

The NAIG critical assembly was arranged with a fully inserted central control rod assembly surrounded by a symmetric array of fuel bundles to simulate a region in a BWR core. Figure 5-1 shows a plan view of the critical configuration utilized for the control rod assembly experiments. The fuel rods were clad in aluminum and contained 2.02 wt% enriched U-235 fuel pellets.

Figures 5-2 and 5-3 compare the rod-by-rod fission distribution for the standard B_4C and all-hafnium control rod assembly. These distributions were obtained by gamma scanning the fuel rods. The uncertainty in the relative rod powers was determined to be ±2% by multiple scanning of each rod. Figures 5-2 and 5-3 show that the fission distribution along the

Copper strips (for activation analysis) were located radially at the core midplane starting at the control rod assembly sheath surface and extending into the water gap. The position of the copper strip is shown in Figure 5-1. The uncertainty in the copper activation analysis was based on the counting statistics and ranges from 0.4 to 1.4% depending upon the location of the strip segment. Figure 5-4 shows that the

The results of these experiments provide critical benchmarks for the Monte Carlo program (Section 5.2.3) and illustrate a minimum expected impact on local power and flux distributions with the presence of an all-hafnium control rod assembly. Similarly, an even smaller impact is expected for the HICR which is a mix of B_4C and hafnium absorber materials.

5.2.3 Methods Qualification

The computer program was used to perform benchmark calculations of the NAIG control rod assembly criticals. The calculated $k_{eff's}$ are contained in Table 5-2 (experiments 8 and 9), as well as other uranium criticals. Analyses of the uranium critical calculations (experiments 1 through 7) indicate that

for thermal reactor experiments (Reference 1). The calculated critical k_{eff} for both the standard B_4C control rod assembly and the all hafnium control rod assembly indicate the to well within the statistical uncertainty. As a result, no additional eigenvalue corrections are required with the use of hafnium cross-section libraries.

results in Section 5.2.2. Figure 5-5 is typical of the agreement seen between the local rod power with a B_4C and all hafnium control rod assembly. The conclusion is that the program accurately predicts the eigenvalue as well as the three dimensional flux distributions.

5.2.4 Fluence Limitations

The nuclear lifetime of the HICR is reached when the top quarter segment reaches a 10% $\Delta k/k$ reduction in reactivity worth, the same criteria used for the existing all-B_4C control rod assemblies. The end-of-life reactivity worth reduction will account for any effects of Boron-10 depletion and B_4C swelling. Tracking of the HICR will be performed in a manner similar to that for the current control rod assemblies, whereby the process computer accumulates the fuel exposures adjacent to the control rod assembly which are then correlated to absorptions in the control rod assembly.

5.2.5 Summary

The current practice by General Electric in Standard Lattice Physics methods is to model the all-B_4C control rod assemblies as non-depleted. The effects of control rod depletion on core performance during any one fuel cycle are small and are corrected for by the critical eigenvalue normalization process performed for each fuel cycle. Section 5.2 demonstrates, through the use of a benchmarked Monte Carlo computer code, that a non-depleted HICR has direct nuclear interchangeability with a non-depleted all-B_4C control rod assembly. The HICR also has the same end-of-life reactivity worth reduction limit as the all-B_4C control rod assembly. As a result, the HICR can be used without change in the current lattice physics treatment of control rod assemblies and current design procedures.

References

1. C. M. Kang and E. C. Hansen, "ENDF/B-IV Benchmark Analyses With Full Spectrum Three Dimensional Monte Carlo Models," paper presented in November 1977 at the winter San Francisco meeting of the American Nuclear Society (ANS), Vol. 22, p. 891.

TABLE 5-1.

REACTIVITY WORTH COMPARISON

(GE Company Proprietary)

TABLE 5-2 - CRITICAL BENCHMARKS
(GE Company Proprietary)

5.3 Thermal Hydraulic Evaluation

During normal operation the maximum internal heat generation rate in the hafnium absorber rods is calculated to be This maximum heating occurs in and is based on a bounding evaluation of the gamma heating due to the fuel and to the self-absorption of hafnium gamma rays. A more representative value of

Heat transfer was calculated using a simplified slab model that neglects vertical and radial temperature gradients. For the coefficient between the absorber rod and water and between the steel sheath and water a convective heat transfer coefficient of was used. That coefficient is more than a factor of 3 below the coefficient predicted by the Jens-Lottes or the Thom correlations.

With the conservative slab model, the maximum heat generation rate and the low heat transfer coefficient, the maximum calculated hafnium absorber rod surface temperature is The internal temperature is only slightly higher. The maximum temperature of the absorber rods containing B_4C is not significantly different from that of rods in the currently used control rod assemblies.

5.4. Surveillance Program

Six Type II HICR's, (see Figure 2-2) will be irradiated in control cell core positions in a selected BWR(s). It is intended to remove three of those HICR's after 18-24 months of irradiation and the remaining three HICR's after an additional 18-24 months. It is planned to start the irradiation period in 1983.

After irradiation, the HICR's will be removed from the core and visually examined at the reactor. Absorber rod lengths will be measured prior to sectioning. Sections of absorber rods will be examined in appropriate hot lab facilities at a location other than the reactor.

Planned hafnium rod examinations include:

a) Dimensional measurements
b) Metallography (hydriding behavior and corrosion characteristics)
c) Mechanical properties
d) Isotopic determinations

Planned B_4C absorber rod examinations include:

a) Visual examination
b) Dimensional measurements
c) Eddy current testing

d) Neutron radiography

e) Isotopic determinations

f) Steam corrosion testing to determine tubing cracking threshold

To complement the above examinations and other extensive in-reactor surveillance and testing programs currently in progress, an additional surveillance program will be performed for the Type I HICR. This will be a long-term surveillance program to provide confirmation of expected performance, including environmental effects, for extended residence time in the BWR. For this surveillance program, the Type I HICR will be placed in a core location where the maximum average B-10 depletion over any 3-foot segment will be less than 4% per cycle. The surveillance program will be performed at the first two non-control cell core plants. Since all the surveillance and test programs previously described provide data on the HICR for up to 6 years of operation, the first visual examination will commence in the sixth year of operation. Subsequent examinations will occur every other cycle thereafter.

Data from these examinations will be used in the development of future new control rod assemblies.

Figure 5-1. Core Configuration with Control Rod (Quarter Core)
(GE Company Proprietary)

Figure 5-2. Fission Distribution Along Bundle Edge
(GE Company Proprietary)

Figure 5-3. Fission Distribution Along Bundle Diagonal
(GE Company Proprietary)

Figure 5-4. Radial Flux (Thermal) Distribution in Water Gap Adjacent to Control Rods
(GE Company Proprietary)

Figure 5-5. Fission Distribution Along Fuel Bundle Edge
(GE Company Proprietary)

6. SAFETY EVALUATION

6.1 Accident Evaluation

The HICR weight and envelope are identical to those of the current control rod assemblies. The mechanical and nuclear properties of the HICR do not differ from those of the current assemblies in any manner that might be significant in a safety evaluation during normal or accident conditions.

Accordingly, the HICR's can be used to replace the currently used BWR 2-4 D lattice assemblies without additional considerations beyond those used in the safety analyses for the current assemblies. Additional discussion follows:

6.2 Mechanical Evaluation

Except for minor differences as described in Section 2, the HICR is mechanically identical to the BWR 2-4 D lattice control rod assemblies for which many reactor years of safe operating experience are available. The HICR sheath thickness is intermediate between that for the BWR 2-4 D lattice assembly and that for the BWR 5 control rod assembly and is identical to that for the BWR-6 assembly.

Accordingly the mechanical safety analysis for the HICR is enveloped by the mechanical safety analysis for the BWR 2-4 D lattice control rod assembly and that for the BWR 5 control rod assembly.

6.3 Thermal Evaluation

During loss of coolant accident (LOCA) conditions the maximum internal heat generation in the hafnium absorber rods is

after the pipe break. These heating rates are based on gamma heating from the fuel and from self-absorption in the hafnium. For purposes of the heat transfer calculation, the heat generation rate was approximated by

This approximation bounds the expected exponential decay of the internal heat generation.

The temperatures are acceptable for the absorber rods because hafnium corrosion properties are better than zirconium properties and the allowable maximum zirconium temperature after a LOCA is 2200°F.

6.4 Reactor Core Response Evaluation

The HICR design has been evaluated against the current control rod assembly design for comparison with LHGR, MCPR and MAPLHGR limits. The HICR weight and rod worth are the same as those for the currently used control rod assembly. Therefore, the scram speed and scram reactivity are also the same. It follows then that the LHGR, MCPR and MAPLHGR limits are not affected by the HICR.