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AN ENGINEERING REVIEW
OF THE
CORE SUPPORT STRUCTURE
OF THE
GAS COOLED FAST BREEDER REACTOR

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1.0 INTRODUCTION

The General Atomic Company program for the development of gas cooled fast breeder reactors (GCFR) has been underway for several years, with support for the design and development activities coming from ERDA and the U. S. electric utilities. The program has progressed to the point where conceptual designs have been prepared for a 300 MWe demonstration unit and a 750 MWe prototype unit, and pre-conceptual design parameters established for a commercially sized 1200 MWe unit. Development programs have been planned, and significant progress made in respect to the objectives of many of these. The GCFR technology has now grown to the point where sufficient information has been generated to permit meaningful independent evaluation of the concepts, design criteria, design approach, and proposed design solutions for many of the reactor's systems, components, and structures.

To provide a technical review and evaluation function for the GCFR program, ERDA contracted with Westinghouse to perform engineering reviews of selected aspects of the GCFR. For Fiscal Year 1977, the GCFR core support structure was selected by ERDA for such a review and evaluation.

The GCFR core support structure, described in Section 3.0, must provide the function of supporting, spacing, and aligning fuel, blanket, and control assemblies in a safe and secure fashion throughout all plant operating modes. Whereas the core support structures in light water reactors, the LMFBR, and the European version of the GCFR are located below the cores, with assemblies "resting" on the support plate and guided and aligned by some combination of upper support structure and/or radial restraints, the GCFR core support structure is situated above the core with the fuel assemblies "latched to" and suspended from the support plate with no radial or lower restraints.

The unique approach to supporting fuel and blanket assemblies provided the incentive for the core support structure to be selected as the first review item. It is not the purpose of this effort to critique the rationale for selection of a

top or bottom core support, but rather to review the General Atomic Company design criteria, approach to the design and analysis, and suggested development program to determine if, based on the experience of Westinghouse in developmental and commercial nuclear programs, the proposed core support structure can ultimately be built to meet all of its operational and safety related functions. To assist in a determination of the adequacy of the design concept, the reviewers attempted to keep the following questions in mind:

- Are the criteria used for the design adequate?
- Does the design meet the criteria?
- Is there adequate ligament strength in the structure?
- In a depressurization accident, where large pressure drops can occur across the structure, are the resulting displacements tolerable?
- Will the structure withstand seismic loads typical of those required in safety analyses of nuclear plants?
- Is the seismic response of the cantilever supported core compatible with meeting the control rod system speed of insertion requirements during a seismic event?
- The fuel and blanket assemblies are suspended from, and locked into, the core support structure with no further constraints. Is this adequate for seismic conditions, fuel bowing, hydrodynamic excitation forces, etc.?
- The mechanical system for locking and unlocking fuel assemblies in the core support structure imposes requirements on the structure itself. Are these requirements adequately accounted for?
- The loads associated with deceleration of control rods are taken by the core support structure. Are these loads adequately accounted for?

- It is contemplated by GA that each fuel element not removed at a refueling will be rotated 180° to compensate, in the next cycle, for the bowing accumulated in the previous cycle. This is proposed in lieu of providing core restraints to prevent gross bowing of fuel elements during burnup. Is this proposed solution workable?
- Are there reactivity effects, that have not been accounted for in the safety analysis, that may be expected due to the use of a core with no mechanical restraints, during both normal and abnormal conditions?
- Because of the high power density in the FBR core it is anticipated that high cooling gas velocities will be required to extract the heat. The high velocity gas streams exiting from the unsupported outlet nozzles at the lower end of the cantilever supported fuel assemblies could cause dynamic lateral excitation of the assemblies. Has this problem been evaluated and is the predicted response amplitude acceptable from mechanical and reactivity feedback viewpoints?
- Can the core support structure be manufactured in commercial sizes? What would be a representative commercial manufacturing cost?

The engineering review did not include any performance of detailed calculations or analyses, but rather used the judgment of personnel experienced in the art to comment on the GA criteria, calculational techniques and results, and to identify potential problem areas not adequately considered. The results of the review are presented in the following sections of this report. The report incorporates consideration of written comments by the General Atomic Company on an earlier draft of this document. The GA comments are included, in their entirety, as Appendix B.

2.0 SUMMARY AND RECOMMENDATIONS

The review of the core support structure of the gas cooled fast breeder reactor (GCFR) covered such areas as the design criteria, the design and analysis of the concepts, the development plan, and the projected manufacturing costs. These, described in the body of this report, are summarized in the paragraphs below. In addition, certain recommendations are provided in this section to establish a basis for future work on the GCFR core support structure.

2.1 DESIGN CRITERIA

The design criteria document (Reference 1, attached as an Appendix A) establishes functional, performance, design, safety, and interface requirements which the core support structure must satisfy. The review indicated that many paragraphs within the document have inadequacies and/or need clarification to properly define the structure's requirements. These shortcomings are in the definition of the components of the structure, in the definition of loading conditions and the method of combining loads, in the seismic criteria, in the quantitative limits for variables (e.g., gas leakage, reactivity changes, etc.), and in the specification of materials of construction.

The General Atomic Company comments (Appendix B) indicate that it is not their practice to list all interfacing components and all anticipated conditions in the design criteria, but rather to incorporate them in a design specification. If this procedure is followed, the sections of the design specification which list the interfacing components and anticipated conditions should at least be referenced in the design criteria section. This is especially true if the correspondence is distributed outside the originating organization, where the words "design criteria" may be interpreted as something different from what was intended. Moreover, the words "all anticipated loading conditions" is not very definitive - if the design criteria is not specific regarding load inputs as basic as thermal loads, control rod deceleration loads and seismic loads, it will become so general that it will probably be useless.

Specific recommendations pertaining to the design criteria include the following:

- (1) Similar to the design criteria document, Reference 1, criteria governing the seismic design and the materials selection of the core support structure must be developed.
- (2) The design criteria must be more firmly established in the following major areas:
 - All the components that constitute the core support structure must be completely defined.
 - All the loading conditions to which the core support structure could be subjected to must be completely defined.
 - Interface requirements must be defined.
 - Methods for combining the loads during the design and operating conditions must be developed.
 - Quantitative limits must be established for the response of the core support structure during all anticipated conditions.
 - Variances in material properties of the core support structure through its design life must be adequately accounted for.

2.2 DESIGN AND ANALYSIS

The review of the design and analysis of the core support structure considered the mechanical design, fuel element attachment, structural analysis, the seismic model, grid plate shielding, reactivity insertion mechanisms, and fuel element rotation.

2.2.1 Mechanical Design

The mechanical design of the core support structure and the means of fuel element attachment is far from complete and undergoing revisions and modifications as part of the design evaluation process. Specific comments and recommendations for successfully completing the design of the GCFR core support structure include:

- (1) Perform tests to investigate flow induced vibrations of the compression tubes and tension rods in the inlet plenum.

- (2) Perform tests to investigate the fuel element assemblies natural frequency and flow induced vibration at the anticipated velocities.
- (3) Investigate specific methods of fabricating the core support structure and grid plate. These parts will have to be made from several pieces and welded to obtain the large sizes required.
- (4) Consider using a shrink fit or a swaging operation instead of a press fit on the fuel element support inserts.
- (5) A normal metallic "O" ring for controlling coolant by-pass flow will have the groove designed to permit about 20 percent deformation. The groove currently shown will not be satisfactory. In this specific application, a gas filled "O" ring might be used to compensate for compression set which must be expected. Installation of the "O" ring will be difficult since the groove is on the lower side of a very heavy part.

Consideration should be given to the use of temporary clips to be used only during installation. The groove could also be machined in the mating part so that gravity will assist in holding the "O" ring during installation of the grid plate.

- (6) The radial keys will be difficult to install on the lower side of the grid plate. They will perform the same function on the outside diameter near the top edge. They may be machined to fit at assembly for control of the radial clearance or separate shims may be used.
- (7) The hold down bolts should be designed to prevent parts from backing off and falling into the reactor cavity.
- (8) Materials selection and proof tests cannot be overstressed. Specific problem areas are as follows:
 - Latch materials and hard facing for the fuel elements and tension rods.

- The sealing surface for individual fuel elements against the lower surface of the grid plate.
 - Springs for the tension rods.
- (9) In order to assess radiation damage to materials, surveillance samples should probably be built into a dummy radial blanket unit for ease of removal and minimum detrimental effects on performance of the reactors.
- (10) The pressure equilization system should be designed in detail to be sure there is no impact on the other parts which interface with this system. The development program should be strongly hardware oriented so that actual parts are fitted together to study the attachment and fitting problems which are necessary to the successful design of a pressure equalization system.

2.2.2 Structural Analysis

The approach to the performance of structural analysis of the GCFR core support as well as the comparison study of displacements of analytical and experimental models of the grid plate, were evaluated and commented on in detail. The evolving nature of the design has resulted in simplifications in the analyses that are acceptable at this stage, but which would require modification in later work. The review of the comparison between the analytical and experimental displacements of the model grid plate shows differences in the axial deflection greater than desirable, indicating shortcomings in the analysis and/or the modeling technique. For instance, at $r = 0$, the difference is about 5.4 percent and at $r = 6$ inches the difference is about 10.7 percent. More importantly, it is shown that the real region of interest in the experimental model is $6 \leq r \leq 10$ inches, where the comparison is stated, in Reference 5, to be poor.

The following recommendations are made with respect to the structural analysis of the core support structure:

- (1) Insofar as grid plate contributions to core reactivity changes are considered, the pressure loading analysis of the core support structure (Reference 7) should be performed with more exact boundary conditions. Additionally, geometric relations between the grid

plate displacement and core reactivity changes must be more accurately used (see Section 5.3).

(2) The stress analysis of the core support structure should also include:

- A separate analysis considering only the mating surfaces between the grid plate and the core assemblies.
- A thermal stress analysis of the core support structure.
- The evaluation of the effects of core jet pressure loadings on the static and dynamic response of the free suspended core should be confirmed by flow tests or full scale mock-ups of GCFR core assemblies.
- An evaluation of the effects of interactions between the grid plate and the core assemblies.

2.2.3 Nuclear Interactions

The nuclear interactions with the core support structure include radiation effects on the structure itself (shielding), reactivity insertion mechanisms, and the fuel element rotation program for counteracting fuel bowing.

The design of the grid plate shielding incorporated in the fuel elements, as described in Section 3.3, is believed to be adequate to meet the design criterion which requires that the core support structure shall have a minimum residual ductility of 10 percent based upon total elongation after a service life of thirty years at a capacity factor of 80 percent. This statement applies to the region of the grid plate that lies directly above the fuel and control assemblies.

It was not possible to evaluate the adequacy of the shielding of the grid plate in the region above the blanket assemblies. No shielding of the grid plate in this region was apparent from the available documents, and no shield analysis was found for this region. However, it seems clear that if there is a shielding problem in this region it can be resolved using shielding of the type proposed for the fuel elements.

The use of a top supported, unrestrained core leads to concerns about reactivity additions resulting from pressure-induced grid plate defections leading to fuel element motion which can potentially increase core reactivity. This mechanism has been recognized and a maximum reactivity insertion criterion, for this phenomenon, has been established. The uncertainties in the structural analysis which define the degree of grid plate displacement make it impossible to assess how well this reactivity criterion is being met.

The proposed rotation of fuel elements by 180° during refueling operations is considered to be a potentially workable, albeit stop-gap solution to the problem of preventing gross bowing of the fuel elements which requires considerable effort to assess its practicality. Proof testing will be required to determine the behavior of an element subsequent to rotation. Analyses of the fuel behavior, fuel-cladding mechanical interaction and structural response of the cladding during step power changes are needed to establish which elements can be rotated, the reactor startup program to limit fuel-cladding mechanical interaction to acceptable values, and the required development testing. In addition, the effects of bowed fuel elements on fuel transfer operations and handling tools must be considered.

2.3 DEVELOPMENT PLAN

The basic philosophy of the development plan for the core support structure (Reference 4) appears to be that the design will be based primarily on analysis, with some supportive experimental testing. The plan considers the development of the structural design criteria, the structural analysis and testing, the seismic response analysis and testing, the backup core support system, and the material compatibility evaluation for the core support structure. Major concerns pertaining to the development plan are identified in the following areas:

- All of the components that constitute the core support structure have not been included in its definition, and development plans have not been made for the solution of potential problems which these components may encounter.

- A development plan has been made for the criteria to be used in the structural design of the core support structure. Similar plans should be made to obtain the criteria governing the seismic design and criteria to be used in the selection of materials for the core support structure.
- The development philosophy should include extensive testing to complement the analyses.
- The development plan should include a task to ascertain the effect of interactions between the grid plate and the core assemblies. For example, the effects of flow-induced vibrations and seismic excitations of the core assemblies on the perforated grid plate should be determined.
- Consideration should be given to test verification of the seismic scram capability.
- The conical surfaces along the bottom surface of the grid plate should be separately addressed. This is because these surfaces are subjected to severe conditions such as high neutron fluence from the core, high thermal stresses, interaction effects due to lateral movements of the core assemblies, and the tendency to self-weld with the core assemblies.
- The development plan for the seismic response test should be expanded to include the effects of thermal bowing of the fuel elements on the grid plate.
- The backup core support system should be analyzed and/or tested to assure that it is functioning with the primary core support system nonfunctioning.
- The materials compatibility evaluation should be expanded to include seals. A development plan should be formulated to explore the possibility of using coatings between metal-to-metal mating surfaces to prevent self-welding in the helium environment.
- Thermal striping appears to be a fundamental problem with any breeder reactor. The problem stems from the large lifetime swing in the heat generation rate within the blanket assemblies. This leads to large differences in the temperature of coolant streams exiting from the core

and blanket assemblies. These streams mix in the region downstream of the core. Structures located in the mixing zone are subjected to many cycles of rapidly changing surface temperature. This condition (known as thermal striping) has been shown to produce structural failure by the action of high cycle thermal fatigue. In CRBRP, extensive outlet plenum mixing tests were required to determine the magnitude, frequency and downstream extent of thermal striping conditions. It is recognized that the lower heat transfer coefficient of Helium can minimize the problem. The possibility, however, of a fluctuating stream of hot gas impinging a metal surface is likely to cause problems if the mass of the part relative to the natural frequency of the oscillations is in the proper range.

- The "in reactor" insulation should be included in the development program. It is particularly important that the sealing arrangement proposed for the insulation be simulated in the development tests. (Note that PFR has experienced a substantial increase in the rate of accumulation of creep-fatigue damage due to a malfunction in an insulation and in the vessel/head gas cooling system.)

2.4 MANUFACTURING COSTS

The cost of manufacturing the core support structures for 300 MWe, 750 MWe, and 1200 MWe plants was estimated. These estimates, as defined in detail in Section 7.0, are based on the cost of materials and fabrication services in accordance with commercial (i.e., LWR) nuclear practice, and do not include any design costs for the structure or special tooling that may be required for fabrication. Based on the design information currently available, the projected preliminary price estimates (expressed in 1977 dollars) are as follows:

	<u>(300 MWe)</u>	<u>(750 MWe)</u>	<u>(1200 MWe)</u>
Core Support Barrel	\$ 591,000	\$ 847,000	\$1,445,000
Core Support (Grid) Plate	863,000	1,305,000	2,479,000
Core Support Plate Inserts	<u>370,000</u>	<u>375,000</u>	<u>678,000</u>
TOTAL:	\$1,824,000	\$2,527,000	\$4,602,000

3.0 DESCRIPTION OF THE CORE SUPPORT STRUCTURE

The core support structure of the 300 MWe gas cooled fast breeder reactor is briefly described here with the intention of providing a reference for the review and evaluation that follows in subsequent sections. The description is taken from References 4 and 10 and is based on information contained in Reference 2; and is provided without comment.

The GCFR core support structure consists essentially of two principal structural members: the reactor grid plate and the reactor grid plate support cylinder.

The function of the grid plate, Figure 1, is to rigidly hold the fuel and blanket elements in position so that the positive reactivity insertion associated with the plate deflection-related core distortion is limited to values that satisfy the plant safety requirements. The grid plate must also be designed to carry the dead weights of the control rod drive and locking mechanisms, the upper shield assembly, and the flow-induced pressure load. In addition, the grid plate provides flow passageways for the pressure equalization system.

To maintain the grid plate and the core attached to it in the required position in the reactor cavity, the grid plate is seated in and supported by a flange on the lower end of the grid plate support cylinder, Figure 2, which extends downward from the liner of the PCRVR top penetration. In addition to carrying the grid plate load, the support cylinder also performs the function of a flow separator for the core inlet and outlet paths. The connections between the grid plate and the support cylinder and between the core elements and the grid plate will be designed to limit the amount of coolant gas passing between the core elements or bypassing the entire core to a small, acceptable amount.

The core support structure will be subject to temperature changes in the inlet helium and heating by nuclear radiation emanating from the core. The entire structure will be shielded and the support cylinder will be insulated so that the maximum temperature and thermal stresses are maintained within acceptable limits

Figure 1. Reactor Grid Plate

Figure 2. Grid Plate Support Structure

during steady-state operation and transients. Moreover, protection of the core support structure against failures due to excessive radiation-induced loss of ductility will be provided by adequate shielding.

3.1 REACTOR GRID PLATE

The grid plate is a thick circular disk which has circular perforations throughout the central portion to accommodate the bayonet sections of the core elements, and is surrounded by a narrow solid outer rim. The plate has orientation keyways to fit their counterparts on the lower flange of the support cylinder. Seals are provided at the connections between the core elements and the grid plate and between the grid plate and the support cylinder to limit coolant bypass leakage. In addition, small vertical holes are drilled through the grid plate to provide the flow passages for the pressure equalization system.

The grid plate, shown in Figure 1, is made of 316SS and has the following dimensions:

Thickness (in.)	24.0
Outer radius (in.)	66.875
Radius of perforated region (in.)	61.0
Average temperature during operation ($^{\circ}\text{F}$)	660.0
Young's modulus of solid material (psi)	25.1×10^6
Poisson's ratio	0.3
Number of core assemblies	211
Number of orientation keyways	6

3.1.1 Core Element Clamping and Reactivity Effect

In the design of the GCFR demonstration plant, the upper end of each core element is rigidly clamped to the grid plate, as shown in Figure 3. This core arrangement ensures a negative thermal bowing component in the power coefficient and avoids the complicated thermomechanical interactions inherent in more redundant support arrangements. The principal requirement with this type of core support is that the grid plate and support cylinder be rigid enough to accurately hold

Figure 3. Core Element Clamping Arrangement

the core elements so that the core geometry can be adequately controlled for all credible disturbing forces such as pressure differentials and seismic excitations.

Each core element is rigidly clamped to the grid plate by an individual breach lock type coupling designed for pre-loading the element into the grid plate with a force of about 1814 Kg (4000 lb) exerted across the mating conical surfaces. The weight of the element and the pretension load are carried by a tension tube to an actuating mechanism located above the reactor cavity closure of the PCRVR. The reaction to this tensile load is returned via a compression tube to the grid plate insert.

The lower end of the thrust tube forms a collet. As the element pretension load is applied, the thrust tube applies more force to the grid plate insert, which tends to center the collet with respect to the hole in the grid plate. The collet in turn tightens on the element extension to center the upper end of the element with respect to the collet. The upper end of the breach dogs at the end of the tension tube and the locking surface inside the upper end of the element extension are tapered. The tensioning force therefore causes these surfaces to seek a center location. The net effect is that when the locking operation is performed and the tension tube is pulled up with the proper load, the upper end of the element is firmly held in a central position with respect to the hole in the grid plate. Thus, the upper end of the element is rigidly held between the upper end and the lower surfaces of the grid plate.

Because the core elements are firmly clamped, the deflection of the grid plate caused by the applied pressure load due to the pressure drop across the core plus the dead weight will result in the core elements splaying slightly, thus causing an increase in active core volume. When the plant operates at a steady-state power output, the reactor is in the critical condition. If there is a sudden reduction of applied forces, e.g., a partial or total loss of coolant circulation, the deflection of the grid plate is reduced. As a result, a positive reactivity insertion ensues owing to the decrease of the active core volume concomitant with the inward movement of the core elements. In view of the plant safety requirement associated with the positive reactivity insertion, the grid plate must be designed

to be sufficiently stiff so that the deflection-related reactivity change is acceptably small.

3.1.2 Pressure Equalization System Passages

The pressure equalization system is designed to nearly equalize the static gas pressure across the cladding of the rods of the core elements and thereby to limit the pressure differential across the cladding. For the pressure equalization system, there are two connections which lead from each core element to the grid plate. These connections are located on the tapered shoulder of the core elements and align with the holes in the chamfered edge of the grid plate hole. These passages permit the fission product gases from the core elements to be aspirated through the grid plate connections into the helium purification system.

3.1.3 Coolant Flow Leakage

The seal between each core element and the grid plate separates the entrance and exit reactor plenum pressure regions. Leakage at these seals degrades plant performance because any primary coolant that leaks through bypasses the core element. Because the mating surfaces may also be the load-bearing surfaces for clamping the core element to the grid plate, forces applied to the core elements by vibrations, seismic excitation, differential thermal expansion, and irradiation-induced swelling are to some extent transmitted to and resisted at the seal and the mating surfaces of the core element and the grid plate. Furthermore, the seal materials and surfaces must operate for the lifetime of the grid plate in high-temperature, high-purity helium, with an allowance for periodic core element demounting, rotation and remounting. In the same manner, a sealing ring is needed to fit between the grid plate and the lower flange of the support cylinder to prevent coolant leakage through the mating surfaces. The dead weight and the pressure drop forces acting downward on the grid plate will seal the contact surfaces and limit coolant bypass leakage.

3.1.4 Grid Plate to Support Structure Connection

In the present reference design, the grid plate is mounted on the lower flange of the support cylinder, as shown in Figure 2. Six radial keys on the flange, which fit into keyways in the grid plate, assure precise radial and azimuthal alignment between the grid plate and the support cylinder. They also provide accommodation for differential thermal expansion during temperature transients and severe seismic excitation without allowing axial displacement or rotation. However, the nature of this connection is similar to that of the grid plate, which is simply supported by the cylinder. This results in the largest deflection of the grid plate, which is undesirable since the grid plate is deflection limited. To reduce the deflection, the grid plate may be clamped to the support cylinder by bolting. Since the grid plate and the lower end of the support cylinder are essentially in the same thermal environment, thermal stresses arising from the clamping are considered to be small. Nevertheless, further investigation is required.

3.2 GRID PLATE SUPPORT STRUCTURE

The grid plate support structure is a large thin-walled cylinder. It includes a flange at the lower end to support and anchor the grid plate and the load acting downward on it and a flange at the upper end to transfer the load to the reactor cavity liner and the PCRV.

The grid plate support structure, shown in Figure 2, is made of 316SS and has the following dimensions:

Thickness (in.)	1.75
Outer radius (in.)	70.375
Length from top of the upper flange to top of grid plate (in.)	120.0
Operating temperature ($^{\circ}\text{F}$)	650.0

3.2.1 Grid Plate Support Structure to PCRV Connection

A flange on the upper end of the support cylinder is designed to transfer loads to the reactor cavity liner and into the PCRV. With this arrangement, the core support structure is held in position in the reactor cavity by loads acting downward on the grid plate. To prevent rotation, lateral movements, and any possible upward movement of the structural assembly resulting from accidental flow reversal across the core, a number of retaining brackets are installed above the upper support flange. To reduce thermal bending stresses in the support cylinder beneath the upper flange, the friction between the mating surfaces should be minimized to permit relative radial displacements due to differential thermal expansion. In addition, a seal is required between the upper flange and the mating flange attached to the PCRV liner to limit primary coolant bypass leakage.

3.2.2 Thermal Insulation

In the demonstration plant reference design, the primary coolant flows inside the support cylinder at a core inlet temperature of 596K (613°F), down to the inlet passages in the grid plate, and through the reactor core. The outer surface of the support cylinder is exposed to the core outlet coolant at a temperature of 823K (1022°F). To reduce the heat exchange between the core inlet flow and the core outlet flow, the outer surface of the support cylinder is clad with thermal insulation over its entire length. By doing this, the support cylinder is essentially maintained at the core inlet temperature. This results in reduction of thermal stresses generated by otherwise high-temperature gradients in the support cylinder.

3.3 CORE SUPPORT STRUCTURE SHIELDING

The core support structure must survive the neutron fluence coming from the GCFR core. If no shielding were provided, the highest fluence would occur in the bottom of the grid plate in the area directly above the core. To protect this region of the grid plate, B_4C shielding is provided in two volumes within each fuel assembly and each control assembly. Figure 4 shows the two volumes as dotted regions; this figure has been reproduced from Reference 2. The upper cylindrical annular B_4C volume, indicated by Section B-B, is partially inserted

Figure 4. Location of B_4C Grid Plate Shielding in a Control Assembly

into the grid plate; the bottom of the grid plate is located at the level of the outside taper between the upper cylindrical part of the fuel assembly and the lower hexagonal part. The lower B_4C volume indicated in Section A-A is a tapered, hexagonal, annular region. It is located in the space between the bottom of the grid plate and the top of the fuel rods. These two B_4C volumes provide an almost-complete enveloping layer of shielding over the bottom of the grid plate in the region directly above the core; small gaps occur in the spaces between adjacent fuel assemblies.

3.4 BACKUP CORE SUPPORT SYSTEMS

In compliance with the general design principles adopted for the GCFR, backup means are provided to maintain core geometry and position in the reactor cavity in the unlikely event of malfunction or structural failure of the core support structure. Protection against the consequences of failure of the grid plate or its support structure is provided by an entirely independent system consisting of individual latching mechanisms and their housings. Normally, these mechanisms clamp the elements to the grid plate with their upper parts "floating" with the housings above the central cavity closure. If the grid plate were to drop a fraction of an inch, part of each mechanism within its housing would "bottom out" on the housing in such a way that the core weight and pressure drop load would be carried by the latching mechanism tension tubes and transferred through the housings to the top of the PCRV closure.

3.5 MATERIAL AND FABRICATION

The core support structure is intended to be a non-replaceable component and shall therefore be designed to function in the reactor system for a life span of thirty years. As a reactor internal, the core support structure will receive considerable neutron irradiation during its lifetime, even though it will be protected by adequate shielding. It is an established fact that neutron irradiation can cause changes in the mechanical properties of materials, e.g., irradiation-induced metal swelling, enhanced creep, and loss of ductility. This influences the structural performance of the core support structure. Consequently, the selection of material

for the core support structure should take into consideration not only its mechanical strength at elevated temperatures, but also its capability to withstand prolonged exposure to neutron irradiation. The effect of the helium coolant on material properties is also important in the selection of material. Because of the inevitable presence of impurity gases such as H_2 , CO , and CH_4 in the helium coolant, individual elements or components within the alloy material selected may be oxidized or reduced and carburized or decarburized depending on the temperature, impurity gas levels, gas partial pressure ratios, etc. The gas-metal reactions could impair the mechanical properties of the materials, e.g., strength, creep rupture, fatigue, and crack growth rates. These factors must be considered when selecting materials which must be compatible with the helium coolant environment.

At present, type 316 austenitic stainless steel is the choice for the core support structure. Type 316 austenitic stainless steel is widely accepted for construction of nuclear reactors since its nonirradiated mechanical behavior is well known and its irradiated mechanical properties are currently being developed by the LMFBR material development program. There is also a substantial amount of information from the gas-cooled reactor projects, especially those in some European countries on the effects of the helium coolant on 316 stainless steel. The preference for type 316 over type 304 stainless steel is due to the fact that type 316 stainless steel appears to have better postirradiation ductility at 644 to 700K (700°F) to 800°F).

Selection of materials is also dependent on the component fabrication method. Although the support cylinder is a simple structure comprising a thin circular cylindrical shell, the grid plate is a complicated, large structure with requirements for accurate positioning of fuel and blanket elements, vent connections, and internal vent holes. Although the grid plate could be formed by casting, a forging may be better because of the requirements for internal vent passages; in a casting, porosity may provide a leakage path. However, a casting with inserts providing the passages may be considerably cheaper and should be considered.

4.0 DESIGN CRITERIA

The General Atomic Company design criteria for the core support structure of the 300 MWe gas cooled fast breeder reactor (Reference 1) is reproduced in its entirety in Appendix A. The evaluation comments below are presented in relationship to the pertinent paragraphs of the criteria document.

1.0 SCOPE

This document provides the general design criteria for the core support structure for the 300 MWe Gas-Cooled Fast Breeder Reactor (GCFR) Demonstration Plant.

Comments

- No Comment

2.0 FUNCTIONAL REQUIREMENTS

The core support structure consists of the core grid plate and the core support cylinder. The grid plate is supported by a flange on the lower end of the grid plate support cylinder, and a flange on the upper end of the support cylinder transfers the loads to the locally thickened reactor cavity liner and into the PCRV shear anchor.

Comments

- The core support structure is stated to consist of the core grid plate and the core support cylinder. The core support structure also consists of the following:
 - (1) The flange supporting the grid plate on the support cylinder,
 - (2) The flange connecting the support cylinder to the Prestressed Concrete Reactor Vessel (PCRV),
 - (3) The radial keys,

- (4) The bolts connecting the grid plate to the support cylinder,
- (5) The shear anchor connection between the support cylinder and the PCRV,
- (6) The seals which inhibit bypass gas flow.

If the scope of the design criteria is to cover the entire core support structure, the definition of the structure should be modified to include the above listed minor components. If an additional document, such as a "design specification" is used to list interfacing components and anticipated operating conditions, that document should be specifically referenced.

2.1 CORE SUPPORT GRID PLATE

The primary functions of the grid plate are to:

- (a) Hold the fuel, control and blanket assemblies firmly in position so that the reactivity change associated with core distortion due to the deformation of the structure is within the values given by Section 4.6.

Comments

- No Comment

- (b) Interface with such components as the control-rod drives, the locking mechanisms, and the pressure equalization system vent lines for all anticipated conditions.

Comments

- Examples of interfacing components are given. If there are other interfacing components, they, too, should be defined here or the appropriate reference cited.
- The anticipated conditions during which the components must interface should either be defined here or the appropriate reference cited.

- (c) Transmit Loads to the grid plate support cylinder due to all anticipated loading conditions.

Comments

- The anticipated loading conditions should either be defined here or the appropriate reference cited. Load transmission structural components are not identified.

2.2 GRID PLATE SUPPORT STRUCTURE

The primary functions of the grid plate support structure are to:

- (a) Support the grid plate and to transfer the loads of the core assemblies and their locking mechanisms, its own weight and the downward pressure load on the grid plate to the PCRV for all anticipated loading conditions.

Comments

- Deadweight and pressure loads are mentioned. Other loads such as: (1) Thermal loads; (2) Control rod deceleration loads; and (3) Seismic loads should also be included. If the design criteria is not specific regarding loads as basic as thermal, control rod deceleration, and seismic loads, it will become so general that it will be close to useless.
- The anticipated loading conditions should either be defined here or an appropriate reference cited.
 - (b) Transmit load due to the differential pressure acting radially on the core support cylinder to the PCRV.

Comments

- In addition to the differential pressure loads on the support cylinder, other loads must be transmitted to the PCRV such as, for example, the thermal loads.

- (c) Provide gas bypass sealing at the grid plate and PCRV liner interfaces.

Comments

- No Comment

3.0 PERFORMANCE REQUIREMENTS

3.1 BYPASS FLOW

During the reactor normal operating condition (pressure differentials, temperature, etc.) as specified in 5050100000, the leakage between the grid plate and the core support cylinder, and between the core support cylinder and the liner shall be minimized and shall not exceed 0.2 % of the total gas flow.

Comments

- A quantitative limit is put on the gas leakage during normal operating condition. If this limitation is established for any purpose other than for establishing a plant efficiency characteristic, then the allowable leakage under "non-normal" operating conditions must be specified.

3.2 MAINTENANCE

The core support structure shall function without maintenance or replacement during the design life specified in 5010100000.

Comments

- The design life is stated, elsewhere to be thirty years. More appropriately it should be stated here.

- By these words, it is implied that all constituents of the structure, including gas seals must last for the design life without maintenance or replacement.

3.3 IN-SERVICE INSPECTION

Access shall be provided to meet the in-service inspection requirements of Section XI, Division 2 for GCFRs planned for issue during Summer 1977 for a one-year trial use and comment period. Inspection provisions will accommodate the current requirements for visual examination and material surveillance of core support structures. For the locations in the core support structure with high predicted fluence, sufficient material shall be obtained during fabrication to produce coupons from which test specimens shall be fabricated.

Comments

- It is assumed that Section XI, Division 2 for GCR's will also require a baseline inspection.
- It is stated that "inspection provisions will accommodate the current requirements for visual examination . . ." Are these current requirements part of the Summer 1977 issue of Section XI, Division 2 for GCR's? No mention is made in the design description of the core support structure as to how the test specimens will be held, where they will be located, and how they will be removed and tested.

4.0 DESIGN REQUIREMENTS

4.1 APPLICABLE CODE CRITERIA

The core support structure shall be designed as a nuclear component in accordance with the following rules:

- (a) The applicable edition of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, "Requirements for Class 1 Components," and the applicable parts of Subsection NG, "Requirements for Core Support Structures."

Comments

- Subsection NB requires that Section NA be used for the material properties.

- (b) RDT E15-2NB (Supplement to Section III), Requirements for Class 1 Components." The Liquid Metal Fast Breeder Reactor Materials Handbook, Nuclear Systems Material Handbook. TID-26666 shall be used to obtain material properties data not available from the above sources. Other material property data used shall be justified in the design report.

Comments

- No Comment

General

The design of the core support structure under seismic conditions must follow a seismic criteria document. This had not been referenced.

4.2 VIBRATION

The core support structure shall be designed to preclude impairment of operability and nuclear stability due to mechanical or flow induced vibrations. The core support structure shall be designed in such a way that neither flow induced vibration nor acoustic induced vibration will result in unacceptably high vibration loads which could result in excessive fatigue damage or wear to the structure.

Comments

- It is not clear whether operability and nuclear stability are affected by acoustic induced vibrations.
- The restriction on the level of flow induced vibrations and acoustic induced vibration loads is related to excessive fatigue or wear. Mechanical vibration and seismic vibration loads must also be accounted for. Additionally, quantitative limit for excessive fatigue damage or wear must be established or related to code allowables. It is unclear whether the excessive fatigue damage or wear is for a ductile material or for a brittle material, especially since the core support structure tends to become brittle with continued operation.

4.3 GENERAL REQUIREMENTS

The core support structure shall be designed so that installation and removal can be achieved through the upper central opening in the PCR. The alignment of the grid plate penetration shall be designed such that the combination of the fuel assembly tolerance and the grid plate tolerance minimize the contact between fuel assemblies.

The grid plate shall be designed as a deflection-limited structure to preclude a positive reactivity insertion in excess of that specified in 5040100000, Plant Safety Criteria, in the event of a Design Basis Depressurization Accident (DBDA).

The maximum design temperature for the core support structure at operating condition shall be 50°F above the helium inlet temperature specified in 5210000000. The temperature for the core support structure for plant transient conditions shall be calculated using parameters from 5050200000. The design differential pressure across the grid plate is specified in 5210000000, and the design life for the core support structure shall be 30 years.

The core support grid plate shall be designed to accommodate the pressure equalization system (the system for the transmission of fission gases from

the core assemblies to the helium purification system). The core support cylinder shall be insulated in such a way as to maintain the structure at essentially core-inlet temperature.

Comments

- More appropriately, the grid plate must be designed as both a deflection limited as well as a stress-limited structure, especially since the ligament stresses at its bottom surface could be high.
- In designing the plate, material property changes that could take place over the operating life of the grid plate must be taken into account.
- It is unclear whether the Design Basis Depressurization Accident (DBDA) causes the maximum grid plate deflection. Would other postulated accidents cause a greater grid plate deflection? Is the simultaneous occurrence of a DBDA and other transients such as an operating basis earthquake possible?
- The design life of thirty years for the demonstration plant appears to be reasonable. However, commercial size plants will more likely have a design life of forty years and this must be accounted for while establishing design criteria for such plants.
- It is unclear how pressure equalization through the grid plates is to be accomplished and what effect it will have on the design of the plate.

4.4 ENVIRONMENT

The normal environment conditions for the support structure components are specified in the relevant parts of 5950100000, Plant Operating Conditions; 5050200000, Plant Transient Conditions.

Comments

- No Comment

4.5 RADIATION DAMAGE

The core support structure shall have a minimum residual ductility^(a) of 10% based upon total elongation after a service life of 30 years at a capacity factor^(b) of 80% as determined from specimens complying to ASTM-E8 or equivalent.

The minimum residual ductility requirement shall apply to the weld material as well as to the base metal.

Comments

- If the seals and connecting components form part of the core support structure system, radiation damage limits must also be established for them.

4.6 REACTIVITY CHANGE

The maximum reactivity change due to the loss of coolant pressure shall be limited to 10¢.

In the event of Safe Shutdown Earthquake, an equivalent of 10¢ step reactivity insertion or an oscillating reactivity change with an amplitude of 60¢ shall be permitted; this amplitude is the combined effect of the movement of the core support structure and the core components.

Comments

- A quantitative limit for the reactivity change due to loss of coolant pressure and an SSE has been set. Quantitative limits due to other transients such as an OBE must also be established.

(a) Defined as post-irradiation material at the end of life which is $\Delta l/l$ 10% at point of rupture, where l is original length, and Δl is the elongation of a 2-inch gage length specimen as specified in ASTM E199 and E 8.

(b) Defined as:

$$\frac{(\text{main unit energy generated})}{[(\text{period hours}) (\text{maximum dependable capacity})]}$$

4.7 LOADING CONDITIONS

The structure shall be designed for all the normal operating, upset, emergency and faulted conditions specified in 5050100000 and 5050200000. As a minimum the following load combinations shall be used in the design and analysis.

- (a) Operational Basis Earthquake (OBE) or one-half Safe Shutdown Earthquake (1/2 SSE) as specified in 5040400000, Seismic: Nuclear Steam Supply System, combined with dead weight, pressure drop across the structure, and the thermal loads.

Comments

- OBE is considered in the load combinations for the design and upset conditions; but not for the normal, emergency, and faulted conditions. Thermal loads are considered only during the upset and normal conditions, but not the other conditions. These specifics are not stated.
- The justification for substituting 0.5 SSE for an OBE is not apparent.

- (b) Safe Shutdown Earthquake (SSE) are specified in 5040400000 combined with dead weight and the pressure drop across the structure.

Comments

- An SSE usually is considered in the load combinations for the faulted condition only.

General:

- The term "operating condition" is ambiguous.
- The method of combining the deadweight, pressure, SSE, OBE, and temperature loads are not defined for the different operating conditions.

4.8 ALLOWABLE STRESSES

The design stress-intensity values for metallic construction materials shall be obtained from ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NB.

Comments

- The term "allowable stresses" appears to be applicable to Class 2 and 3 nuclear components, but not Class 1 nuclear components. Perhaps the title here is better stated as "Stress Criteria."
- The stress intensity values are in subsection NA and the stress intensity limiting values are in subsection NG of the ASME Code, Section III.

4.9 MATERIALS

Materials selected for the core support structure shall be compatible with the environment as specified in 5040500000. In addition the grid plate material shall be selected (or protection provided) such that self welding between fuel assembly and grid plate does not occur.

Comments

- Materials selection must be according to subsection NG of the ASME Code, Section III.

5.0 SAFETY REQUIREMENTS

5.1 DESIGN CLASSIFICATION

The Core Support Structure is a Seismic Category 1 item per 5040100000, Plant Safety Criteria.

Comments

- The core support structure is stated to be a Seismic Category 1 structure, but the seismic criteria which it must satisfy are neither defined nor referenced in the documents available for review.

6.0 INTERFACE REQUIREMENTS

The design of the core support structure is restricted by requirements imposed by components or adjacent structures. The major interface requirements are tabulated below:

<u>Major Interface</u>	<u>Document No.</u>	<u>Interface Requirements</u>
Overall Plant Design	5010100000	Determine design life of core support structure.
PCRVR Cavity Liner and Closures	5110200000	Loads at liner attachments.
PCRVR Thermal Barrier	5110400000	Thermal barrier thickness and attachment requirements.
Primary Coolant System	5210000000	Design temperature and P.
Fuel Assembly	5170100000	Space envelope and loads.
Control Assembly	5170200000	
Blanket Assembly	5170300000	
In-Core Thermal Probe	5170500000	Requirement for core outlet temperature probes.
Plant Safety	5040100000	Defines safety requirements for core support structure, and classification of accidents.
Pressure Equalization System	5230200000	Transmission of fission gases from the fuel elements to the purification system.
Upper Shield Assembly	5130300000	Nuclear heating and irradiation.
Radial Shield Assembly	5130500000	
Plant Transient Conditions	5050200000	Operating conditions.
Plant Operating Conditions	5050100000	Operating conditions.
In-Service Inspection	5080200000	See In-Service Inspection of Core Support Structure.
Seismic: NSSS	5040400000	Seismic requirements.
Environmental Qualification of NSS Equipment	5040500000	Material environmental compatibility.

Comments

- None of these documents were provided for review, therefore no comments can be made regarding the adequacy of the design criteria.

7.0 REFERENCES

<u>Reference</u>	<u>Document Section</u>
ASME Code Section III, Division I, Subsection NB	4.1a, 4.8
ASME Code Section III, Division I, Subsection NG	4.1a
ASME Code Section XI	4.1a
ASTM E8	4.5a
ASTM E139	4.5a
LMFBR TID-26666	4.1b
RDT E15-2NB	4.1b

Comments

- ASME Code Section XI: The document section is 3.3.

5.0 REVIEW OF THE DESIGN AND ANALYSIS OF THE CORE SUPPORT STRUCTURE

This section contains the Westinghouse evaluation of the design and analysis of the core support structure for the 300 MWe gas cooled fast breeder reactor as performed by the General Atomic Company. Specifically, Section 5.1 contains the review of the mechanical design, Section 5.2 the fuel element attachment, Section 5.3 the structural analysis, Section 5.4 the comparison study of the grid plate displacements, Section 5.5 the seismic model, Section 5.6 the grid plate shielding, Section 5.7 the reactivity insertion mechanisms that are unique to an unrestrained core, and Section 5.8 the rotation of the fuel elements. The review identifies adequacies/inadequacies and problem areas in the design and analysis, and additionally, makes specific recommendations.

5.1 MECHANICAL DESIGN

5.1.1 Core Support Structure

The basic design of the core support structure is simple and functional. The applied loads should not cause excessive deflections if stresses are maintained at reasonable levels for all loading conditions. Major concerns which must be addressed are discussed below with suggestions for consideration as the design is firmed up.

5.1.1.1 Natural Frequency

The entire assembly is suspended from the top surface of the pressure vessel and its natural frequency could amplify the vibration of parts supported by the grid plate.

5.1.1.2 Thermal Gradient

The thermal gradient across the thickness of the support could be of concern if the insulation is not properly selected and applied.

5.1.1.3 Materials Surveillance

Section 3.3 of Reference 1 identifies the need for surveillance samples to be placed in areas of high level radiation to assess the effect of radiation on the different parts of the reactor. Nowhere in the information available for use in evaluating the design is placement of the samples discussed. One basic problem with any instrumentation is placement without effecting performance. The specific location of the surveillance samples as well as the method of attachment should be spelled out in detail early in the design to be certain the samples are properly located and offer minimal degradation of performance. Care must also be taken to be certain the samples are actually a part of the material used to make the part being evaluated. Several duplicate samples may be required for each part to be tested since a sample taken out of the reactor is very difficult to replace and a time versus property curve would be very desirable - therefore multiple samples are necessary for initial installation with a suitable means for retrieval.

5.1.1.4 Pressure Equalization

The fuel pressure equalization system represents many deep drilled holes and an elaborate tubing system to collect gases from the fuel elements and direct them into the helium purification system. The system should be studied in detail. The large number of potentially loose parts and assembly difficulties may justify a manifolded sandwich construction held in place by the element support inserts. The equalization system will probably have a greater impact on the mechanical design of the grid plate and element support inserts than is generally recognized. Plumbing of the type required to manifold the gases is usually left until last, often with very painful results. Special inserts may be required and pre-bent and welded tubing may be necessary and should be designed before the other parts are finalized. Milled interconnecting slots between drilled passages might be considered as an alternate method with tubing as a final termination of the equalization system. The specific design does not appear to be finalized in this area and this system should be another part of an integrated total design rather than an "add-on" after the more interesting work is completed.

5.1.2 Grid Plate

The grid plate will be difficult to manufacture due to its large size. Depending upon plant rating, the plate may have to be fabricated in more than one part, with proper welding and heat treatment prior to machining. The deep, small-diameter holes required for the fuel pressure equalization system and the beveled surfaces on the lower side of the plate which mate with the fuel assemblies to form a seal will offer a challenge to the manufacturer.

5.1.2.1 Tolerances

The close center position and the close tolerance on the inside diameter of the individual holes on the top side of the grid plate must be held tight to permit a press fit of the element support inserts.

A six inch nominal size hole with an interference fit will run from 15 to 32 thousandths interference, according to the 19th edition of Machinery Handbook, as a minimum range, and could run much heavier depending on the ligament between holes. The press fit will require large forces and with such forces the possibility for damage to the grid plate is increased. The use of a shrink fit or a swaging operation performed from the lower side of the grid plate might be considered as a desirable alternate to the press fit. Any method of attachment must be sufficiently strong to resist the forces exerted by the refueling machine as it inserts elements from the bottom.

5.1.2.2 "O" Ring Seal

The seal on the lower side of the grid plate as shown in Figure 1 has a groove which is not properly sized for what is apparently a 3/8 inch diameter cross section metallic "O" ring.

The depth of the groove should provide for approximately 20 percent deformation and the "O" ring should be soft plated to provide a more effective seal.

The "O" ring should be gas filled to compensate for compression set assuming a safe method of installation can be found. As presently shown the seal would probably be held in place by an adhesive while the grid plate was installed. It could also be held in place by clips which would preclude its being gas filled- or the groove could be machined in the mating part.

The "O" ring seal will be affected by differential thermal expansion and could actually "slide" on the mating surface reducing its effectiveness.

5.1.2.3 Radial Keys

The grid plate is held in place both radially and azimuthally by six radial keys which also accommodate differential thermal expansion. The design of these keys as represented on Figure 2 cannot accomplish these requirements. If the keys are tight enough to locate the grid plate radially, then expansion of the grid plate relative to the core support cylinder will crush the space keys or destroy the grooves, probably by forcing the grid plate upward while stretching (or deforming) the hold-down bolts.

Sufficient radial clearance could be provided for the six keys to prevent interference during normal differential thermal expansion. Actual radial centering could then be controlled by using shims on the upper edge of the grid plate.

The radial keys could be moved to the top of the grid plate where they would be much more easily installed and could be designed to ride upward in specially designed keyways if thermal differential expansion becomes excessive. This reduction would also provide more space for the "O" ring seal.

5.1.2.4 Hold-Down Bolts

The hold-down bolts shown in Figure 2 must be sized primarily for seismic considerations in the vertical direction. If the six keys are moved as suggested above, larger diameter bolts could be accommodated.

No method of preventing the threaded fasteners from backing out and falling has been suggested in the information examined. This is a problem which should be addressed early in the design to make certain the method selected is acceptable. Threaded fasteners are avoided whenever possible inside the reactor pressure vessel. When they must be used, some method of assuring that the threads cannot back out has been required. A further requirement has been that in the event of a fractured bolt the pieces are retained instead of being allowed to drop into the reactor cavity. This requirement will almost certainly be placed on the gas cooled breeder and should be covered in the design.

5.2 FUEL ELEMENT ATTACHMENT

5.2.1 Clamping Mechanism

The fuel element clamping arrangement as currently shown in Figure 3 must be engaged blind and the fuel element must be properly oriented to engage and lock.

5.2.2 Positioning Latches

The rotational positioning tolerances in the loading machine and the tension rod must both be machined closely enough to assure engagement, and latch members must be positioned vertically to permit the tension rod to rotate and engage the lower edge of the latch surface on the fuel assembly. (It would be helpful during the engagement operation if the vertically engaging edges were beveled to assist in overcoming misalignments.)

5.2.3 Vertical Travel

There must be sufficient vertical travel in the tension rod to engage the latch on the fuel assembly including enough overtravel to permit the necessary rotation. The clamping action which holds the fuel element securely must pass through the tension rod into the compression tube to the element support insert.

5.2.4 Vertical Expansion Loads

Unless the compression tube and tension rod assembly are allowed to "float" vertically, the differential thermal expansion between these parts and the pressure vessel, grid support and grid plate stackup could impose a load on the grid plate through the insert on the top surface.

5.2.5 Element Support Insert

The element support inserts must be machined to sufficiently tight tolerances to provide a press fit in the top of the grid plate. The total force required to press these parts in place is appreciable and some thought might be given to using a shrink fit or swaging the parts in place from the opposite (lower) side of the grid plate. The hexagon shape on the top surface is useful in indexing the inlet vanes which support the center section of the insert used for clamping the fuel assemblies in place. The entire weight and clamping force to hold the fuel and blanket assemblies in place must be supported by the four slender supports attached to the center ring. All the vibration and seismic loads are transferred through these same supports as well as the weight of the tension rod and compression tube used to clamp the fuel or blanket assemblies in place.

5.2.6 Fuel Element Seal

The design provided for review showed the fuel assemblies clamped between the beveled surface on the bottom of the grid plate and the latch surfaces on the tension rod. This design has apparently been revised to use piston rings for by-pass flow sealing rather than the conical surface at the bottom of the grid plate. A properly designed piston ring could satisfy the sealing requirement; however, the conical sealing surface which has been retained as a centering support point must also function as a lead-in for the piston ring. This requires a gradual taper to compress the piston ring. The piston ring must also have rounded edges to prevent galling of the lead-in surface. Testing should be used to confirm the specific dimensions and materials of this design.

5.2.7 Fuel Element Insertion

Although fuel insertion is indirectly related to core support structure design, the rigidity of the fuel assemblies must be investigated in order to evaluate the feasibility of handling the assemblies from the lower end. The ability of the upper end to follow the motion of the refueling machine should be investigated.

5.2.8 Materials Selection

Material selection for these fuel/grid plate mating surfaces is critical. Testing proposed or in progress should help locate areas and provide a sound basis for proper selection of materials to be used in the final design.

5.2.9 Spring Material

The spring material in the latching mechanism is another area of concern. Springs for use at 600°F are a very real engineering and metallurgical problem. There are several materials to choose from, but the stresses must be kept relatively low to insure a reasonable life which means that the spring will be larger than would normally be expected for a given application. The effects are not well understood or documented for spring materials at high temperature and radiation levels combined. It is suggested that a speciality spring company be consulted early in the design phase and that springs be mounted on replaceable parts if at all possible.

5.3 STRUCTURAL ANALYSIS OF THE CORE SUPPORT STRUCTURE

The review is itemized to first provide brief descriptions of information contained in Reference 7 and then followed by the review comments.

5.3.1 Core Support Structure Loading Conditions

The structural analysis of the GCFR Core Support Structure considers only lateral pressure loading on the perforated grid plate, and internal and external pressures on the core support cylinder.

Comments

The analysis should include other loadings/effects such as, for example,

- Thermal stresses.
- Seismic effects and flow induced vibration effects.

- Interaction effects between the core assemblies and the grid plate.
- Neutron irradiation effects on the material properties of the core support structure.
- Irradiation induced swelling, creep, and loss of ductility.

5.3.2 Core Support Structure Components Analyzed

The analysis includes the grid plate, the grid support structure (including the upper and lower flanges and the core support cylindrical shell), and the structural interaction between them.

Comments

- The analysis does not include the radial keys, the bolts connecting the grid plate to the support cylinder, the shear anchor connection between the support cylinder and the prestressed concrete reactor vessel, or the seals which inhibit bypass gas flow.

5.3.3 Grid Plate Analysis Method

The analysis of the perforated grid plate is performed using the concept of the equivalent solid plate.

Comments

- The grid plate under consideration has two kinds of holes: bayonet holes for the core assemblies (of diameter 6.375 inches and 211 in number), and holes for the pressure equalization system (of diameter 0.25 inch and "numerous" in number). Since, apparently, the equivalent solid plate concept cannot be applied to a plate with more than one type of hole, a determination must be made as to the magnitude of the error incurred in the application of the concept to the grid plate analysis.

- The bayonet holes occupy 75 percent of the perforated region of the grid plate.

5.3.4 Grid Plate Analysis Models

Three analyses of the perforated grid plate are performed (all using the concept of the equivalent solid plate) and compared: The first uses isotropic thin plate theory, the second uses isotropic thick plate theory, and the third uses transversely isotropic thick plate theory. All three analyses assume the outer solid rim of the plate to be simply supported. A comparison of the numerical values of the stresses (excluding the outer solid rim effect) as obtained from the three different analyses indicates that they are approximately the same. A comparison of the numerical values of the displacements (including the outer solid rim effect) as obtained from the three different analyses indicates that the thin plate analysis yields a centroidal axial displacement 32.6 percent lesser than the isotropic thick plate analysis. The transversely isotropic thick plate analysis yields a maximum radial displacement value approximately 32 percent greater than the isotropic thick plate analysis, while the centroidal axial deflection is 21 percent lesser.

Comments

- The effect of the outer solid rim is taken into consideration (treating it as an isotropic ring) by incorporating the redundant moments and stresses into the displacement solution. However, this effect has not been incorporated into the stress solution; this must be done since the ligament stresses in the grid plate can be computed from the stress solution.
- The ligament stresses are not evaluated.

5.3.5 Grid Plate Boundary Conditions

All three plate analyses make the same assumptions, and essentially follow the same procedure. The perforated portion of the grid plate is assumed to be homogenous, with modified elastic constants, and is treated as simply supported. The effect of redundant edge moments and edge radial forces is computed separately. The outer solid rim is then analyzed separately as simply supported on the outer edge. The redundant moments and radial forces at the interface of the perforated region and the solid rim are solved for by simultaneously equating the edge radial displacement of the perforated region to that of the solid rim, and by equating the edge slope of the perforated region to the centroidal slope of the outer solid rim.

Comments

- The boundary condition $w(r = R, z = 0) = 0$ used to evaluate the constant of integration in the expression for the axial deflection of the perforated region is incorrect. Here $w(r, z)$ is the axial deflection of the plate. The correct expression should be:

$$w(r, h/2) = 0.$$

- The use of the boundary condition

$$-\theta^p \Big|_{\substack{z=0 \\ r=R_1}} = \theta^r \Big|_{\substack{z=0 \\ r=R_1}}$$

appears incorrect. The quantity on the left hand side represents the slope at the edge of the perforated region while the quantity on the right hand side is stated, in the derivation, to be the slope about the centroid of the solid ring. The slopes should be equated at the same radial location.

- In the analysis of the perforated region, wherein transversely isotropic thick plate theory is used, the edge boundary conditions:

$$\int_{-h/2}^{h/2} \sigma_{rr} (R,z) dz = 0,$$

$$\int_{-h/2}^{h/2} \sigma_{rr} (R,z) z dz = 0$$

are incorrect. The right hand sides of these two equations must be the redundant radial force H and the redundant moment M, respectively, between the perforated region and the solid outer ring.

- The outer solid rim of the grid plate should be analyzed more rigorously; preferably using three dimensional isotropic theory for a solid of revolution instead of treating it as an isotropic ring.

5.3.6 Grid Plate Deflection Related to Reactivity Change

The primary purpose of the analysis of the grid plate is to determine the reactivity change in the core due to grid plate displacement. The reactivity change in the core due to grid plate deflection is calculated to be -6.96¢ due to a 42 psi pressure load on the grid plate. This is less than the plant safety requirement of -10¢ reactivity change.

Comments

- By extrapolation, a pressure greater than 60 psi on the grid plate will result in a reactivity change greater than -10¢. This indicates that measures must be taken to assure that the pressure on the grid plate does not approach 60 psi.

- The increment of core radius at the core midplane is derived from Figure 5 (taken from Reference 7) as

$$\Delta R = L \frac{\partial w}{\partial r} (R_c, 0)$$

where R is the core radius, and L is apparently the distance from the bottom surface of the grid plate to the core midplane. (This conclusion is based on the fact that the numerical value of L is taken as 63.19 inches in Reference 7 and is approximately the same as 64.36 inches as obtained from Reference 10). The correct equation is either.

$$\Delta R = (L + \frac{h}{2}) \frac{\partial w}{\partial r} (R_c, 0)$$

or

$$\Delta R = L \frac{\partial w}{\partial r} (R_c, h/2)$$

The use of the first of these gives a reactivity change of -8.28¢ (a 19 percent increase over the value of -6.96¢ in Reference 7) while the second gives -7.51¢ (an 8 percent increase over the value in Reference 7). Assuming that the second of the above quotations is valid, a pressure of 56 psi will increase the reactivity change beyond the limiting value of -10¢.

- From Figure 5, it is apparent that thermal bowing of the grid plate would increase ΔR thereby increasing the value of the reactivity change. Furthermore, seismic vibrations could also increase ΔR .

Figure 5. Geometric Considerations for the Grid Plate Deflection

5.3.7 Core Support Cylinder Analysis

The grid plate support structure (the core support cylinder and the upper and lower flanges) is analyzed using strength of materials theory. The highest stress on the outer surface of the core support cylinder is about 49 ksi, but this may be reduced to about 37 ksi by increasing the wall thickness to 2.25 inches (from 1.75 inches).

Comments

- The procedure appears correct.

5.3.8 Interaction Between Grid Plate and Support Cylinder

An evaluation of the interaction effect between the grid plate and support structure by assuming the plate to be thin and by ignoring the solid rim effect has been made. The plate end condition was assumed to be clamped. The end moments (at the clamped edge) were computed and indicate that their sense is such that they would increase the grid plate rotation.

Comments

- Since the true boundary condition of the outer surface of the grid plate would probably simulate a clamped condition in the region of the bolts, it is important that the increase in the grid plate rotation due to the end moments be incorporated while obtaining the expression for the plate axial displacement; this has not been done in Reference 7.
- It is suggested that the structural interaction between the grid plate and its support structure be more accurately determined.

5.4 REVIEW OF THE COMPARISON STUDY OF THE GRID PLATE DISPLACEMENTS

The review is itemized to first provide brief descriptions of information contained in Reference 5 and then followed by the review comments.

5.4.1 Grid Plate Models Used for Comparison

A comparison of the grid plate displacements is made between the results of an experimental model, a finite element model, and the analytical model of Reference 7. The test model was an aluminum alloy plate of 20 inches diameter and 3.6 inches thickness. The pattern of perforations had a pitch of 1.03 inches and a hole diameter of 0.953 inch. The maximum applied pressure was 60 psi.

The finite element model used two dimensional axisymmetric elements and the SAP-IV computer code. The perforated region of the plate was treated as a transversely isotropic equivalent solid plate (9 inches in radius) with modified elastic constants, surrounded by an isotropic region (1 inch thick).

Comments

- In view of the concern expressed (Sections 5.3.1 and 5.3.6) about the interaction effects of the lateral movements of the core elements on the grid plate, it is suggested that an experimental program be undertaken with the core elements attached to the grid plate and measurements made of the grid plate deflection due to the lateral splaying of the elements.
- Since expressions for the ligament stresses were not explicitly obtained (see Section 5.3.4), it is suggested that a testing program be undertaken to determine the ligament stresses by attaching strain gauges to the grid plate ligaments.

5.4.2 Boundary Conditions for the Comparison Models

The boundary conditions for the analytical model (Reference 7) and the finite element model simulate a simply supported condition. However, the edge of the experimental model was clamped between two steel rings through eight preloaded bolts. The partial clamping effect at the edge was analytically treated and incorporated in the test results for comparison. This was done by determining the ratio of the axial deflection at the center of the grid plate due to pressure loading to the reduction in axial deflection at the center of the grid plate due to clamping effect; and then adding a proportionate amount of the analytical displacement to the measured displacement thereby obtaining the experimental displacement corresponding to the simply supported edge condition.

Comments

- The expression W_1 used for the axial deflection at the center of the grid plate and due to pressure loading is obtained from isotropic thin plate theory and not from the transversely anisotropic thick plate analysis for which this comparison was intended. The value of the aforementioned ratio should therefore be 0.4544 instead of 0.4834.

5.4.3 Comparison Results

The axial displacement at the center of the grid plate due to a pressure loading of 60 psi is computed from the analytical solution of Reference 7 to be 9.387 mils. The measured deflection at the center of the grid plate is stated to be 4.7 mils. The effect of the edge clamping is taken into account and the experimental deflection is obtained as $4.7 + 0.4834(9.387) = 9.238$ mils. The centroidal axial displacement as obtained from the finite element solution is 9.2 mils. The discrepancy between these three values is less than 2 percent.

Comments

- Chuang (Reference 8) states that the axial displacement was measured at the bottom surface of the grid plate and not at the mid-surface. However, the comparison has been made between $w(r=0, h=0)$ for the analytical models and $w(r=0, z=h/2)$ for the test results. A quick calculation (Equation (4) of Reference 5) indicates that $w(r=0, z=h/2) = 9.555$ mils. The experimental deflection is then obtained (following the procedure of Reference 5) as $4.7 + 0.4544(9.555) = 9.1$ mils. This results in a discrepancy of 5.37 percent between the analytical and experimental results. At the location $r = 6$ inches, the analytical deflection is 4.906 mils and the experimental value is 4.379 mils; the difference is consequently 10.73 percent instead of 5.76 percent. Although there appears to be a conservative analysis for plate deflection, the important locations where plate slopes are of interest have not been adequately treated (see Section 5.4.4).

5.4.4 Validity of Comparisons

Comparisons of the axial deflection are made between the three models to a radial location of $r = 6$ inches. The maximum discrepancy is stated to be 6 percent in the region $0 \leq r \leq 6$ inches. Comparative values are not presented in the region $6 \leq r \leq 10$ inches; but it is stated that the comparison is not good in this region.

Comments

- All comparisons have been made between $w(r,0)$ for the analytical models and $w(r,h/2)$ for the experimental model. Consequently the discrepancy should be greater than the indicated 6 percent (see Section 5.4.3).

- In Reference 7, the ratio of the core radius to the outer radius of the grid plate was given as $R_c/R_o = 44.75/66.875 = 0.6692$. In the test model, the radius ratio at which the comparison between the results becomes poor is given as $6.0/9.625 = 0.6234$. (Note: A 3/8 inch wall thickness of the outer solid rim is clamped; therefore the outer radius of the grid plate is taken as $10 - 3/8 = 9.625$ inches). Since the slope at $r = R_c$ is used to compute the reactivity change (Reference 7), it appears (Reference 5) that the comparison of the axial deflection may become poor in the region of interest.

5.5 REVIEW OF THE SEISMIC MODEL FOR THE CORE SUPPORT STRUCTURE

The review is itemized to first provide brief descriptions of information contained in Reference 6 and then followed by the review comments.

5.5.1 Seismic Model for Core Support Structure

A new modeling technique for the core support structure is under development. The technique will permit one to determine the effect of the core elements, hanging as cantilevered beams from the perforated plate, on the grid plate vibration. The amplitudes and stresses of individual core elements can be computed, and the relative motions between core elements can be evaluated.

The seismic analysis of the core support structure is divided into three parts: (1) analysis of the grid plate; (2) analysis of the core element; and (3) model synthesis. The first part is performed in depth, the second is discussed, while the third is briefly described.

Comments

- The seismic analysis of the core support structure must include the core support cylinder; especially since this is a flexible component and will therefore effect the vibrations of the grid plate and consequently of the core elements.

5.5.2 Seismic Model for Grid Plate and Core Elements

In the analysis of the grid plate, the core elements are assumed to be rigid rods and their effect is incorporated by taking their mass and rotary inertia into effect. The plate is analyzed as a simply supported orthotropic thick plate. The effect of the outer solid rim is taken into account. A numerical example indicates that the frequency of the grid plate is 8 cps.

Comments

- The displacement solution (Equations (2), (10), (11), (12), and (19)) of Reference 6 is shown to be independent of both the tangential coordinate and of time t . This is incorrect. For example, the displacement solution as given by equation (19) must include trigonometric functions of the tangential coordinate θ (see Reference 2 cited in Reference 6).

5.5.3 Seismic Analysis of Core Elements

The core element analysis is performed using the finite element technique; beam elements are used in the SAP-IV computer code. Two cases are analyzed: one in which the core elements are clamped to the lower surface of the grid plate and the other in which they are pinned. For the first case, the fundamental frequency of fuel elements is computed to be 11.2 cps and for the blanket elements 9.4 cps. For the second case, the corresponding frequencies are 9.3 cps and 7.7 cps, respectively.

5.5.4 Coupling Between Grid Plate and Core Elements

In the brief description of the modal synthesis, it is stated that the coupling effect between the core elements and the grid plate must be determined since the frequencies of the two systems are in the same range. These effects will be considered in future work using the SAP-IV computer code.

5.6 GRID PLATE SHIELDING

The design criterion (Reference 1) for the grid plate shielding says that "the core support structure shall have a minimum residual ductility of 10 percent based upon total elongation after a service life of thirty years at a capacity factor of 80 percent." R. E. Perkins and R. J. Cerbone note (Reference 2) that "the precedent for such a conservative design criterion has been set, for example, in the fast flux test facility (FFTF) and the Clinch River breeder reactor (CRBR) programs because of a lack of structural design and analysis methods." The design criterion as stated above is adequate for the shield design for the 300 MWe demonstration reactor. As indicated earlier, however, commercial reactors are frequently specified to have a design life of forty years with an 80 percent capacity factor. Perkins and Cerbone have performed (Reference 2) damage function analyses for several potential configurations of the grid plate shielding. Their analyses require, as input, an energy-dependent damage function plus the energy-dependent neutron fluence at the grid plate. In order to be conservative they have used an upper-bound damage function, plus a factor of two bias on the calculated neutron fluence. With these conservative assumptions they show that the design criterion can be met using the grid plate shielding configuration shown in Figure 4.

The primary uncertainty in the neutron fluence calculations of Perkins and Cerbone was attributed (Reference 2) to the effect of axial streaming in the rod lattice on the source emerging from the axial blanket. This being the case, some concern arose as to whether bowing of the fuel elements was being considered in the analyses of the axial streaming. A crude calculation was done which indicated that bowing of the fuel assemblies might increase the axial leakage of uncollided neutrons by a factor of about fifteen. However, the uncollided fluence with bowed rods was estimated to be less than one-tenth percent of the total neutron fluence at the grid plate, so it was concluded that bowing of the fuel rods should not affect the results obtained by Perkins and Cerbone. Experiments aimed at verifying the calculations of the energy and angular dependence of the neutron flux emerging from the GCFR fuel rod lattice are being considered (Reference 3) and should be encouraged. Hopefully, the results of the experiments will permit a reduction in the shielding.

The factor of two bias in the calculated neutron fluence is judged to be sufficient to cover the uncertainties in the calculation. Accordingly, it is concluded that the analysis of Perkins and Cerbone is adequate to demonstrate that it will be possible to provide adequate shielding for the grid plate.

The analyses by Perkins and Cerbone (Reference 2) apply to the area of the grid plate that is directly above the fuel and control assemblies. No analyses were found for the region of the grid plate above the blanket assemblies. The document available did not indicate any shielding between the top of the UO_2 in the blanket rods and the bottom of the grid plate. Analyses may be required to establish what shielding, if any, is required for the grid plate in the area above the blanket assemblies. This is not a problem that is crucial to decisions on the viability of the GCFR core support because it is clear that shielding such as that shown in Figure 4 can be used in the blanket assemblies if required.

5.7 REACTIVITY INSERTION MECHANISMS THAT ARE UNIQUE TO AN UNRESTRAINED CORE

Consideration was given to reactivity insertion mechanisms which are unique to the GCFR core support arrangement. The unusual features of this arrangement are the lack of support at the bottom of the fuel assemblies and the lack of a restraint system to restrict lateral movements of the free ends of the assemblies. The objective of the review was to see if any reactivity insertion mechanism could be found which had not been heretofore considered.

5.7.1 Loss of a Control Rod Cluster From a Control Assembly

If the support rod for a control rod cluster were to break, the cluster would drop. The design of the control assembly is such that the cluster would be caught by a hard stop at the lower end of the assembly. The cluster would not escape from the control assembly; hence, a reactivity insertion will not occur.

5.7.2 Loss of a Control Assembly

If the supports for a control assembly fail, then the assembly will drop or be expelled from the core. General Atomic has considered this possibility (Reference 10) and has found that the maximum reactivity insertion would be +25 cents. Normally, other control clusters would be moved to compensate for this reactivity insertion and little would happen. If the other control clusters were not moved and if the scram circuits failed to operate, then the power level would rise to an equilibrium level determined by the negative temperature coefficient of reactivity. At this equilibrium level, the temperatures of the remaining fuel assemblies have not exceeded the fuel safety limit, and there is no propagation of the problem.

5.7.3 Fuel Bowing and Splaying

The effect of the fuel bowing and splaying on reactivity has been studied (Reference 9) by General Atomic. The magnitudes of the effects are small (7¢ for deadweight deflection of the grid plate, 9¢ for pressure load on the grid plate, 6¢ for thermal gradients in the fuel assemblies) and the time constants are long (7 minutes for the pressure effect and about 15 seconds for thermal bowing) so there should be little problem in compensating for these effects by moving the control rods.

The design of the control assembly allows sufficient clearance between the control rod clusters and their guide tubes so that binding should not occur when the assembly is bowed.

5.7.4 Lateral Oscillations

General Atomic personnel have considered (Reference 9) the reactivity effects of earthquake induced core oscillations. Using the conservative assumption that all fuel elements oscillate coherently inward and outward from the core center, the reactivity oscillates between +65¢ and -65¢. The frequency is about 9 Hz. Reactor trip signals would be activated during the first millisecond and trip

would occur after 0.2 second. It is claimed that only very small changes in core temperatures would occur and that even if the control clusters failed to move, the asymptotic temperatures in the fuel and the cladding would not exceed damage limits. Nevertheless, the review results in the conclusion that the seismic scram capability is a major area of feasibility uncertainty in the cantilever core support system. Large amplitude seismic motion of the core can cause impacts affecting the guide duct. In the case of CRBRP, which has smaller lateral motions of the guide duct because of the multi-level support, seismic impact loads made the results from the original analysis of seismic scrams marginal. An extensive development testing program was undertaken to demonstrate the effectiveness of the fluid damping and squeeze film lubrication of the surrounding sodium. Factoring these results into the seismic scram analysis produced acceptable results. The GCFR, however, has no fluid medium in the core to supply damping or squeeze film lubrication. This, coupled with the large amplitude seismic motion of the core, could make the seismic scram capability of the GCFR marginal.

5.7.5 Thermal Shield Segment(s) Tip Against Radial Blanket

The GCFR grid plate does not support the inner thermal shield. It is supported from the outer thermal shield. If the supports fail in such a way that a segment or several segments of the inner thermal shield tip against the radial blanket, then the neutron reflection from the thermal shield will increase causing an increase in GCFR reactivity. Inasmuch as the inner thermal shield is separated from the core by two radial blanket assemblies, the reactivity effect should be very small. This postulated event cannot lead to a significant reactivity insertion.

5.7.6 Dynamic Excitation

An example of a possible positive reactivity feedback mechanism is illustrated in Figure 6. In Figure 6, it is assumed that full power-flow ratio conditions are established at low (say 40%) power. Full thermal loads therefore occur at this power. The unbalanced pressure in the exit nozzle exerts a normal force P_j on the end of the fuel assembly. This force increases as the power (and associated

Figure 6. Forces on Fuel Element

flow) increased from 40 percent to 100 percent. The temperature gradients do not, however, increase since power-flow matching is achieved at 40 percent power. The result is that the radial component of P_J (P_L) causes the "splayed" fuel assemblies to bow towards the core P_L giving a positive reactivity feedback.

The jet force could also produce vibration by the "Becks problem" mechanism (statically unstable but dynamically stable - see Timoshenko, Theory of Elastic Stability, page 154).

5.8 ROTATION OF FUEL ELEMENTS

It is contemplated that fuel elements not removed at a refueling will be rotated 180° to compensate, in the next cycle, for the bowing accumulated in the previous cycle. This is proposed in lieu of providing core restraints to prevent gross bowing of fuel elements during burnup. Consideration has been given to the workability of this rotation scheme.

The proposed solution to compensate for fuel element (assembly) bowing, although potentially workable, is considered not practical because of the extended refueling time and the difficulty in reassembly of the bowed ducts. The UKAEA is now investigating restrained cores for CFR rather than the free bowing cores used in PFR (which is similar in its bowing behavior to the GCFR). Two aspects of the solution are not currently proven and have resulted in a decision not to rotate elements during early operation of FFTF and CRBRP. These aspects are summarized below, together with potential analyses and/or tests required to ultimately establish the limits for which the solution is workable.

5.8.1 Core Restraint Performance

Undoubtedly, the elements in GCFR core will be subject to neutron flux gradients and temperature gradients. These gradients combined with the pressure loading on the ducts and the effects of the neutron flux itself, will cause the elements to bow and dilate (i.e., bulge out at the midflat portion of the hexagonal duct relative to the corners). Bowing of the elements is a function of the

three dimensional temperature profiles in the duct walls, the axial (along duct) and radial (cross duct) flux profiles in the duct walls, and the irradiation induced swelling and creep characteristics of the duct material. Duct dilation is a function of the pressure loads on the duct and the parameters given above. Both duct dilation and bowing must be considered if duct to duct contact is to be avoided. Rotation of the ducts compounds the situation and the net effect of the complex interaction of all the variables involved is a relatively large uncertainty on the time at which ducts must be rotated to preclude interacting with their neighbors or to preclude excessive reactivity changes. It should also be noted that the degree of bow experienced in the first fuel irradiation cycle is approximately 50 percent of that which would be expected during subsequent cycles. This is due, as shown in Figure 7, to the linear dependence of swelling on fluence and the fact that in 20 percent cold worked Type 316 stainless steel the incubation fluence is of the order of $4 \times 10^{22} \text{ n/cm}^2$ ($E < 0.1 \text{ MeV}$).

Quantitative analyses are needed to establish how long the core can be operated before element rotation is necessary and the subsequent behavior after element rotation. These analyses must consider uncertainties in the various input parameters and correlations and the effects of unplanned fuel replacements or other changes in the fuel management scheme resulting from unexpected practical problems, e.g., fuel element leaks. Proof testing to determine the behavior of an element subsequent to rotation will probably be required. General Atomic has analyzed (Reference 9) the bowing of the fuel elements induced by differential thermal expansion and irradiation swelling. The results look encouraging in that when the rotation scheme is used, interference is limited to a maximum of about 0.15 inch and is limited to about six or nine locations. More recent swelling correlations may remove the interference. No analyses were found for the dilation effect.

$$\Delta v_2 \approx 2 \Delta v_1$$

Figure 7. Effect of Fluence on Fuel Swelling

5.8.2 Fuel Rod Performance

The fuel and blanket rods in the CGFR design which was reviewed featured very small initial fuel-cladding gaps (0.002 and 0.0015 inch, respectively) and fission gas venting. These features will result in significant fuel-cladding mechanical interaction (FCMI) loading of the cladding relative to that in LMFBR fuel rods. Because there are radial flux and power gradients in the core, element rotation will necessarily result in power increases in some rods which will, in turn, magnify the FCMI. The degree to which breeder fuel elements can withstand such loading is not well established at this time, but reactor startup at slow enough rates can mitigate the transient FCMI effects.

Analyses of the fuel behavior, FCMI, and structural response of the cladding during step power changes are needed to establish which elements can be rotated, the reactor start-up program to limit FCMI to acceptable values, and the required development testing. Such analyses have been performed for LMFBR's and confirmatory testing has been planned.

6.0 REVIEW OF THE DEVELOPMENT PLAN FOR THE CORE SUPPORT STRUCTURE

The review of the General Atomic Company development plan for the GCFR core support structure (Reference 4) is contained in this section. The review follows the following format: the titles are those used in the development plan (Reference 4), the statements/synopses are based on information contained in the plan and the comments represent the result of the review.

6.1 GENERAL

The GCFR core support structure consists essentially of two principal structural members: the reactor grid plate and the reactor grid plate support cylinder.

Comments

- The core support structure should also consist of the following components:
 - (1) The flange supporting the grid plate on the support cylinder.
 - (2) The flange connecting the support cylinder to the PCR.V.
 - (3) The radial keys.
 - (4) The bolts connecting the grid plate to the support cylinder.
 - (5) The shear anchor connection between the support cylinder and the PCR.V.
 - (6) The seals which inhibit bypass gas flow.
 - (7) The retaining brackets.

Consequently, these components of the core support structure must be included in the development plan, also.

The grid plate must --- be designed to carry the dead weights of the control rod drive and locking mechanisms, the upper shield assembly, and the flow-induced pressure load.

Comments

- Other grid plate loads should be included in the plate design. Examples of these are (a) seismic, (b) thermal, (c) grid plate dead-weight, (d) interactive loads between the grid plate and support cylinder, and (e) interactive loads between the grid plate and core assemblies (such as loads resulting from flow-induced excitations and seismic excitations of the core assemblies).

There is considerable similarity between the GCFR and Clinch River breeder reactor core support structures, especially in the grid plate --- . The GCFR core support structure can be based upon the design, analytical, and manufacturing experience of the Clinch River breeder reactor to ensure the integrity of the GCFR core support system.

Comments

- In applying experience gained from the design of the core support structure for the Clinch River Fast Breeder Reactor (CRBR) to that of the GCFR, care must be taken to account for the differences in environment (helium for the GCFR and liquid sodium for the CRBR).

6.2 STRUCTURAL DESIGN CRITERIA

The objective of this work is to establish structural design criteria which will facilitate the design of the GCFR core support structure using analytical methods with a minimum of experimental verification required.

Comments

- Considering the complexity of the analysis, it appears reasonable to assume that substantial experimental verification will be required for the analyses.

Modes of structural failure --- should encompass ductile rupture from short-time loading.

Comments

- Modes of structural failure should include brittle rupture due to short time loading. Although the core support structure may not be brittle at the end of its operating life, the consideration of both ductile and brittle rupture due to short time loading will scope the problem of structural failure due to short-time loading.

6.3 STRUCTURAL ANALYSIS AND TEST

6.3.1 Structural Behavior of the Grid Plate

(The objectives of this work are to validate the applicability of stress analyzing the multiply perforated grid plate as a solid plate with modified material properties). This facilitates the analysis of the structural behavior of the grid plate and the prediction of the stresses, strains, and deflections under GCFR loading and environmental conditions.

Comments

- The grid plate has two different kinds of perforations: holes to accommodate the bayonet sections of the core elements and axial holes for the pressure equalization system. A determination should therefore be made as to whether any assumptions are being violated in applying the theory of the equivalent solid plate with modified moduli

to the analysis of the grid plate. If the theory is not valid, an attempt should be made to evaluate the error incurred in its use.

- The analysis should consider the fact that after a period of operation, the grid plate will have non-homogenous properties induced by nuclear irradiation.
- The effect of interactive forces between the core elements and the grid plate must be incorporated in the grid plate structural analysis.

The three-dimensional anisotropic elasticity theory will be used to analyze the equivalent solid plate of the grid plate.

Comments

- In using three-dimensional anisotropic elasticity theory to analyze the equivalent solid plate, material properties in three mutually perpendicular planes will be required. There is no description of how these material properties will be obtained.

The results from the analytical development will be verified by scale-model tests. The applied load could be a uniformly distributed force and/or concentrated forces, and simply supported and clamped edges could be the boundary conditions.

Comments

- The effect of interactive forces between the core elements and the grid plate should also be incorporated in the grid plate test.

6.3.2 Structural Interaction Between Grid Plate and Support Cylinder

The objective of this work is to investigate the effect of local flexibility of the support cylinder on the stresses and deflections of the grid plate and thereby to optimize the means of connection between the grid plate and the support cylinder.

Comments

- In considering the interaction between the grid plate and the support cylinder, a structural evaluation of the components connecting the grid plate to the support cylinder must be made; i.e., the bolts, keys, and seals (if they are load bearing) must be analyzed. Similarly an evaluation of the components connecting the support cylinder to the PCRV must be made.
- Since the grid plate deflection is of importance in reference to the reactivity change in the core, the probable fact that the core support cylinder acts as an elastic support to the grid plate should be taken into consideration in the grid plate analysis.

6.3.3 Thermal Stress Analysis

The objective of this work is to demonstrate that existing methods are adequate to evaluate thermal stresses and deformation of the core support structure under steady-state and transient operating conditions.

Comments

- The objective should be to actually perform the thermal stress analysis rather than to demonstrate that the analysis can be done on the basis of existing methods.

The thermal stress analysis will consist of the following work items:

- (1) Development of a heat transfer model for the grid plate and the support cylinder in order to accurately determine the steady-state and transient temperature distributions.
- (2) Development of time-independent and time-dependent thermomechanical models of the core support structure for the evaluation of thermal stress and deformations during steady-state and transient operating conditions.

Comments

- Considering the complexity and the criticality of the thermal stress problem, it is assumed that numerical methods of solution will be used.

Substantial literature is cited which could be applicable to the steady-state thermal stress problem of the support cylinder if it had a radial temperature distribution. (Synopsis)

Comments

- Since the temperature distribution in the cylinder is a function of both radial and axial location, it appears that the literature that is cited can only be used as a basis for the solution of the thermal stress problem and not for the actual solution itself.
- Neither a method of analysis nor literature is cited which could be used to solve the steady-state thermal stress problem of the grid plate.
- No method of analysis is proposed for the solution of the thermal stress problem of any component of the core support structure during transient conditions.

6.3.4 Creep-Fatigue Damage Analysis

In the work scope, it is stated that the structural life of the core support structure will be determined by defining the cyclic loading conditions from all anticipated transient operating conditions and by selecting an acceptable creep-fatigue interaction model by reviewing and evaluating existing literature. (Synopsis)

Comments

- In performing the creep and fatigue analysis, no statement is made as to the constitutive equations that will be used for the creep analysis or of the availability of stress-cycle curves for the fatigue analysis.

In the background for the problem, literature is cited that appears to depict the considerable interest in this subject. The following statement is also made, "the mechanisms associated with creep-fatigue interaction are not clearly known at this time." (Synopsis)

Comments

- The literature that is cited does not appear to provide specific methods of analysis which could be used in the solution of the problem under consideration.
- From the description of the background for the problem, it is apparent that the creep-fatigue damage analysis of the core support structure is formidable and, perhaps, of limited value. Additionally, in the light of the above quotation, it is perhaps better to determine the creep-fatigue damage through an experimental program rather than through an analytical program.

6.3.5 Analysis of Irradiation Effects

The objective of this work is to evaluate the effect of neutron irradiation on materials behavior and thereby to establish the neutron fluence limit for the core support structure.

Comments

- It is not explicitly stated how irradiation induced embrittlement will be related to neutron fluence, but it is implied that this will be done experimentally.

In the work scope it is stated that a "development of analytical models for analysis of irradiation-induced swelling and enhanced creep" will be made. (Synopsis)

Comments

- No description is made as to what the analytical models will include and no literature is cited.

General Comment

- Irradiation effects on the materials of the seals should be determined to assure the proper functioning of the seals through their design life.

6.4 SEISMIC RESPONSE ANALYSIS AND TEST

6.4.1 Seismic Response Analysis

A dynamic structural model of the core support structure will be developed with the effect of the boundary conditions and interaction of the core elements included.

Comments

- No statement is made as to whether a finite element model or closed form analytical model will be used or how the damping coefficients will be obtained.

The literature that is cited appears to describe the current state of the art of seismic resistance design. (Synopsis)

Comments

- The literature does not appear to specify methods of analysis which could be used to determine the seismic response of the core support structure.

6.4.2 Seismic Response Test

It appears that a seismic response test will be performed on a scale model of the grid plate with core elements. (Synopsis)

Comments

- It is not clear whether the test will also be performed on a scale model of the entire core support structure.
- It is gratifying that regardless of the GA comment (Appendix B, Section 5.7.4) that "no seismic scram feasibility uncertainty is considered to exist," seismic testing is planned that will include..." Scram tests of the complete control rod system under maximum seismic deflection conditions will be part of the control and drive system development program..." (Appendix B, page B-6, 5th concern).

In the test, the effect of an irradiated environment will be considered by accounting for irradiation induced swelling of the core assemblies.
(Synopsis)

Comments

- Thermal bowing of the core elements must also be accounted for as should a means for verifying the scram capability of the core under seismic conditions.

6.5 BACKUP CORE SUPPORT SYSTEMS

General Comments

- The backup core support system must remain functional after the failure of the primary core support system. The failure mechanism must first be postulated and then the continued functioning of the backup core support system during and after these operating conditions must be assured through analysis and/or test. If the operating conditions which could cause failure of the primary core support system cannot be hypothesized, it appears appropriate that analyses and/or tests be performed for all operating conditions with the backup core support system functioning and the primary core support system nonfunctioning. These operating conditions should include seismic and flow induced excitations.

6.6 MATERIAL COMPATIBILITY EVALUATION

The effects of the helium coolant environment on the materials to be used for the core support structure are to be determined. While no experimental program is envisaged, information obtained in the development of high temperature helium gas cooled reactors will be used. (Synopsis)

Comments

- In addition to determining the effects of the helium environment on structural materials such as austenitic stainless steels, an evaluation of the effects of the helium environment on the material of the seals must also be determined.

Concern is expressed that in a helium environment, the mating surfaces of austenitic steels tend to fret, gall, and self-weld. Since the contact surfaces between the core assemblies may weld to the grid plate and, consequently, refueling could be a problem. (Synopsis)

Comments

- This potential problem could be circumvented by coating the mating surfaces with a material such as titanium nitride that is hard and does not diffuse. The coating material, however, should first be tested to assure that it is compatible with its environment and does not pose other problems.

Concern is expressed about possible gas-metal reactions that could occur and change the material properties such as rupture stress, creep fatigue, and crack growth rates. It is stated that an assessment of the material property changes will be made. (Synopsis)

7.0 CORE SUPPORT STRUCTURE MANUFACTURING COST

One measure of the practicality of a design concept is the projected manufactured cost of the item. With this in mind, estimating price data was prepared for the core support structures for the GCFR in sizes suitable for the demonstration plant (300 MWe), prototype plant (750 MWe), and commercial size plant (1200 MWe). The price estimates were prepared by the Westinghouse Pensacola Division, supplier of core support structures for Pressurized Water Reactors. The estimates therefore reflect the conditions of a commercial market.

It should be recognized that the estimates are based on preliminary and incomplete design information; therefore, when more complete information is available, final estimates could vary considerably from those presented below.

The resulting preliminary price estimates, expressed in 1977 dollars, are as follows:

	<u>(300 MWe)</u>	<u>(750 MWe)</u>	<u>(1200 MWe)</u>
Core Support Barrel	\$ 591,000	\$ 847,000	\$1,445,000
Core Support (Grid) Plate	863,000	1,305,000	2,479,000
Core Support Plate Inserts	<u>370,000</u>	<u>375,000</u>	<u>678,000</u>
Total	\$1,824,000*	\$2,527,000*	\$4,602,000*

* It is necessary to acquire tooling to fabricate these components. Since this tooling will vary considerably with the fabricator's existing facilities, these estimates DO NOT include any factors for tooling requirements.

The preceding estimated prices are based on the following assumptions and/or conditions:

- The following tabulation of dimensions indicates the approximate sizes of the parts used in these price estimates:

	(300 MWe)	(750 MWe)	(1200 MWe)
<u>Grid Plate Support Structure Assembly Length</u>	156.0"	213.0"	186.0"
<u>Core Support Barrel</u>			
<u>Upper Flange</u>			
O.D.	151.6"	180.0"	227.4"
I.D.	137.2"	162.8"	213.0"
Thickness	7.2"	7.2"	7.2"
Length	12.0"	12.0"	14.0"
<u>Barrel Section</u>			
O.D.	140.7"	167.0"	222.0"
Thickness	1.7"	2.1"	3.0"
Length	108.0"	165.0"	128.0"
<u>Lower Skirt</u>			
O.D.	140.7"	167.0"	222.0"
Thickness	3.0"	3.5"	4.5"
Length	24.0"	24.0"	30.0"
<u>Lower Flange</u>			
O.D.	140.7"	167.0"	222.0"
I.D.	128.7"	152.6"	207.6"
Thickness	6.0"	7.2"	7.2"
Length	12.0"	12.0"	14.0"

	(300 MWe)	(750 MWe)	(1200 MWe)
<u>Core Support (Grid) Plate</u>			
O.D.	133.7"	159.0"	210.0"
Thickness	24.0"	24.0"	30.0"
No. of Element Holes	211.0"	217.0"	397.0"
<u>Core Element Inserts</u>	211.0"	217.0"	397.0"

- The Core Support Barrel portion of the structure will be fabricated from four basic parts:

- Top Flange (ring forging)
- Mid barrel (plate Material)
- Bottom skirt (plate material)
- Bottom flange (ring forging)

These parts will be welded together and machined.

- The core support plate (grid plate) will be fabricated from forged material as described below:

- In the 300 MWe size the grid plate will be a one piece 316SS forging.
- In the 750 MWe size, the grid plate will be a two piece 316SS forging welded together with a hot wire TIG process utilizing a special Westinghouse Pensacola Division welding technique.
- In the 1200 MWe size, the grid plate will be a three piece 316SS forging welded together with a hot wire TIG process utilizing a special Westinghouse Pensacola Division welding technique.

- In obtaining raw material estimates, it was noted that steel industry experience with 316SS was limited or nonexistent, especially in large forgings and ingots.
 - There are only two United States sources for the large disc forgings, and one of these could cease to melt stainless steels in the near future. Only one supplier submitted estimates on the multiple piece forgings required in the 750 MWe and 1200 MWe grid plates.
 - Three domestic sources submitted estimates on most of the ring forgings; however, again one possible supplier could cease stainless steel production in the future. One domestic ring forging supplier prefers to use billet material supplied by a Canadian steel company.
- The grid plate would be fabricated and machined as described below:
 - After welding, in the cases where multiple forgings are used for raw material, the grid plate forging would be finished machined on all surfaces.
 - Flow and vent holes would be machined into the forging.
 - The machined grid plate would receive a flame-sprayed hard facing application, as specified. Westinghouse Pensacola does not have manufacturing experience or systems in place to provide the hard facing application; and it is the opinion of its welding engineers that this would be a very difficult application and would require an extensive development program. The above estimates are based upon the assumption that General Atomics will develop a suitable method of flame spraying the hard facing material on the grid plate. The above estimates include using this method to be developed at the rate of two flow holes per eight hour shift.
 - This estimate assumes that no special welder qualification requirements are needed for the carbide powder application for hard facing.

- After application of the hard facing material, the treated surfaces will be ground to provide a good surface finish for the element insert seating.
 - This estimate assumes no special nondestructive examination of the final ground surface will be required.
 - The above estimates do not include supplying venting piping or hardware.
 - The element support inserts would be final machined from 316SS castings. After machining, specified areas of the element inserts require hard facing and the comments in the above five paragraphs are applicable. Installation of these inserts into the grid plate is not included in these estimates as this cannot be accomplished until the venting piping and hardware are installed.
 - The grid plate will be installed into the core support barrel prior to shipment. This will be a bolted assembly.
- All three core support structures will be shipped with core support barrel and the grid plate assembled as one piece.
 - It is the Westinghouse Pensacola Division experience that both the 300 MWe and the 750 MWe sizes could be shipped in the horizontal position via rail.
 - The 1200 MWe structure would require shipment via barge and, therefore, could be shipped in the vertical position.
 - All three structures would be supported with a shipping fixture during the shipment. The ends would be protected with plywood covers, and the entire assemblies would be wrapped in a 6 mil polyethylene and covered with a canvas during shipment..
 - No special atmosphere, evacuation, or desiccant use has been included in the preparation for shipment.

- Price estimates are based upon the fabricator securing all raw materials.
- Price estimates are based upon the prime contractor or the system designer supplying all the necessary drawings, specifications, and design information in a format suitable for direct use by the fabricator to secure raw materials and produce the components required.
 - The necessary design calculations and the applicable stress reports will be generated by the system designer.
 - Fabricator will produce the components on a build-to-print basis.
- Price estimates are based upon using in-place manufacturing procedures, methods, standards, processes, and systems currently used in the commercial production of reactor vessel internals for light water-pressurized water reactors.
- Price estimates are based upon all customer inspections being conducted as witness points, not hold points, as currently used and accepted in present day production practice.
- Price estimates are based upon fabrication in accordance with Section III ASME boiler and pressure vessel code.
 - Components are classified as core support structures coded NG.
 - Welding processes and qualifications will be in accordance with Section IX of the ASME B & PV code. No special welder qualification requirements are needed for the carbide power application for hard facing.
 - Nondestructive Examination processes and procedures will conform to Section V of the ASME B & PV code. No special nondestructive examination of the final ground surface will be required.
- Price estimates are based upon product acceptance at the fabricator's factory and the fabricator's responsibility ceases at the f.o.b. point.

- Price estimates are based upon a limited fabricator's warranty which will warrant only the work performed on the contract as meeting the requirements of applicable drawings and specifications, and that work shall be free from any defects in workmanship. The fabricator will not be responsible for any end use of the product. The maximum extent of the fabricator's liability will be limited to the value of the contract.
- Price estimates are f.o.b. cars point of shipment, freight not included.
- Price estimates are based upon terms of payment being progressive payments which will be made monthly throughout the procurement and fabrication cycle.

8.0 REFERENCES

- (1) A. S. Chuang, "Design Criteria: Core Support Structure (300 MWe GCFR)," Document No. 5130100000, GA-1194 (Rev. 1), Preliminary Issue, General Atomic Company (March 31, 1977).
- (2) R. G. Perkins and R. J. Cerbone, "Physics Design for the 300 MWe GCFR Demo Plant Grid Plate Shielding," GA-A14124, General Atomic Company (September, 1976).
- (3) R. G. Perkins and R. J. Cerbone, "Grid Plate Shielding Design Confirmation Experiment Test Requirements," No Document Number, General Atomic Company (March 8, 1977).
- (4) K. H. Chang and A. S. Chuang, "Development Plan for the GCFR Core Support Structure," GA-A13938, General Atomic Company (August 16, 1976).
- (5) A. S. Chuang, K. H. Chang, and C. E. Washington, "Comparison Study on GCFR Core Support Grid Plate Displacement among Experimental, Analytical, and Finite-Element Models," GA-A14325, General Atomic Company (March, 1977).
- (6) L. E. Penzes, K. H. Chang, and G. E. Lee, "Seismic Model of the Gas-Cooled Fast Breeder Reactor Core Support Structure," GA-A13397, General Atomic Company (August 15, 1975).
- (7) K. H. Chang, A. S. Chuang, and C. E. Washington, "A Structural Analysis of the GCFR Core Support," GA-A14098, General Atomic Company (September, 1976).
- (8) Telephone conversation between A. S. Chuang, General Atomic Company, and R. Fernandes, Westinghouse Advanced Energy Systems Division (July 11, 1977).
- (9) A. Torri and M. J. Driscoll, "Reactivity Insertion Mechanisms in the GCFR," GA-A12934, General Atomic Company (April 10, 1974).
- (10) Letter from L. J. Kube, General Atomic Company, to L. Welshans, U. S. Energy Research and Development Administration (June 22, 1977).

APPENDIX A

GENERAL ATOMIC COMPANY DESIGN CRITERIA: CORE
SUPPORT STRUCTURE (300 MWe GCFR) DOCUMENT NO.
5130100000, 3/31/77

GA 1194 (Rev. 1)

DESIGN CRITERIA: CORE SUPPORT STRUCTURE (300 MWe GCFR)

Approval Level 5

Document No. 5130100000

Issue A

Date

ISSUE SUMMARY

Issue	Date	Prepared by	Approvals		Purpose of Issue
			Name	Date	
A		A. Chuang <i>A. Chuang</i>	<i>C. E. Washington</i> C. E. Washington	3/31/77	Initial Issue
			<i>L. J. Kube</i> L. J. Kube	3/31/77	
			R. H. Simon		
<p style="text-align: center;">CAUTION</p> <p>This report contains possible patentable information and should not be further disseminated without the approval of The Document Management Branch, ERDA Technical Information Center, Oak Ridge, Tenn., or The Assistant General Counsel for Patents, ERDA Headquarters.</p>					

Quality Level: 1

PRELIMINARY

General Atomic Company

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1.0 SCOPE

This document provides the general design criteria for the core support structure for the 300 MW(e) Gas-Cooled Fast Breeder Reactor (GCFR) Demonstration Plant.

2.0 FUNCTIONAL REQUIREMENTS

The core support structure consists of the core grid plate and the core support cylinder. The grid plate is supported by a flange on the lower end of the grid plate support cylinder, and a flange on the upper end of the support cylinder transfers the loads to the locally thickened reactor cavity liner and into the PCRV shear anchor.

2.1 Core Support Grid Plate

The primary functions of the grid plate are to:

- (a) Hold the fuel, control and blanket assemblies firmly in position so that the reactivity change associated with core distortion due to the deformation of the structure is within the values given by Section 4.6.
- (b) Interface with such components as the control-rod drives, the locking mechanisms, and the pressure equalization system vent lines for all anticipated conditions.
- (c) Transmit loads to the grid plate support cylinder due to all anticipated loading conditions.

2.2 Grid Plate Support Structure

The primary functions of the grid plate support structure are to:

- (a) Support the grid plate and to transfer the loads of the core assemblies and their locking mechanisms, its own weight and the downward pressure load on the grid plate to the PCRV for all anticipated loading conditions.
- (b) Transmit load due to the differential pressure acting radially on the core support cylinder to the PCRV.
- (c) Provide gas bypass sealing at the grid plate and PCRV liner interfaces.

3.0 PERFORMANCE REQUIREMENTS

3.1 Bypass Flow

During reactor normal operating condition (pressure differentials, temperature, etc.) as specified in 5050100000, the leakage between the grid plate and the core support cylinder, and between the core support cylinder and the liner shall be minimized and shall not exceed 0.2% of the total gas flow.

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3.2 Maintenance

The core support structure shall function without maintenance or replacement during the design life specified in 5010100000.

3.3 In-Service Inspection

Access shall be provided to meet the in-service inspection requirements of Section XI Division 2 for GCFRs planned for issue during Summer 1977 for a one-year trial use and comment period. Inspection provisions will accommodate the current requirements for visual examination and material surveillance of core support structures. For the locations in the core support structure with high predicted fluence, sufficient material shall be obtained during fabrication to produce coupons from which test specimens shall be fabricated.

4.0 DESIGN REQUIREMENTS

4.1 Applicable Code Criteria

The core support structure shall be designed as a nuclear component in accordance with the following rules:

- (a) The applicable edition of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, "Requirements for Class 1 Components," and the applicable parts of Subsection NG, "Requirements for Core Support Structures."
- (b) RDT E15-2NB (Supplement to Section III), "Requirements for Class 1 Components." The Liquid Metal Fast Breeder Reactor Materials Handbook, Nuclear Systems Material Handbook TID-26666 shall be used to obtain material properties data not available from the above sources. Other material property data used shall be justified in the design report.

4.2 Vibration

The core support structure shall be designed to preclude impairment of operability and nuclear stability due to mechanical or flow induced vibrations. The core support structure shall be designed in such a way that neither flow induced vibration nor acoustic induced vibration will result in unacceptably high vibration loads which could result in excessive fatigue damage or wear to the structure.

4.3 General Requirements

The core support structure shall be designed so that installation and removal can be achieved through the upper central opening in the PCR. The alignment of the grid plate penetration shall be designed such that the combination of the fuel assembly tolerance and the grid plate tolerance minimize the contact between fuel assemblies.

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The grid plate shall be designed as a deflection-limited structure to preclude a positive reactivity insertion in excess of that specified in 5040100000, Plant Safety Criteria, in the event of a Design Basis Depressurization Accident (DBDA).

The maximum design temperature for the core support structure at operating condition shall be 50°F above the helium inlet temperature specified in 5210000000. The temperature for the core support structure for plant transient conditions shall be calculated using the parameters from 5050200000. The design differential pressure across the grid plate is specified in 5210000000, and the design life for the core support structure shall be 30 years.

The core support grid plate shall be designed to accommodate the pressure equalization system (the system for the transmission of fission gases from the core assemblies to the helium purification system). The core support cylinder shall be insulated in such a way as to maintain the structure at essentially core-inlet temperature.

4.4 Environment

The normal environmental conditions for the support structure components are specified in the relevant parts of 5050100000, Plant Operating Conditions; 5050200000, Plant Transient Conditions.

4.5 Radiation Damage

The core support structure shall have a minimum residual ductility (a) of 10% based upon total elongation after a service life of 30 years at a capacity factor (b) of 80% as determined from specimens complying to ASTM-E8 or equivalent.

The minimum residual ductility requirement shall apply to the weld material as well as to the base metal.

4.6 Reactivity Change

The maximum reactivity change due to the loss of coolant pressure shall be limited to 10¢.

In the event of Safe Shutdown Earthquake, an equivalent of 10¢ step reactivity insertion or an oscillating reactivity change with an amplitude of 60¢ shall be permitted; this amplitude is the combined effect of the movement of the core support structure and the core components.

(a) Defined as post-irradiation material at the end of life which is $\Delta l/l = 10\%$ at point of rupture, where l is original length, and Δl is the elongation of a 2-inch gage length specimen as specified in ASTM E139 and E8.

(b) Defined as:

$$\frac{(\text{main unit energy generated})}{[(\text{period hours})(\text{maximum dependable capacity})]}$$

Notations in this column indicate where changes have been made

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4.7 Loading Conditions

The structure shall be designed for all the normal operating, upset, emergency and faulted conditions specified in 5050100000 and 5050200000. As a minimum the following load combinations shall be used in the design and analysis.

- (a) Operational Basis Earthquake (OBE) or one-half Safe Shutdown Earthquake (1/2 SSE) as specified in 5040400000, Seismic: Nuclear Steam Supply System, combined with dead weight, pressure drop across the structure, and the thermal loads.
- (b) Safe Shutdown Earthquake (SSE) as specified in 5040400000 combined with dead weight and the pressure drop across the structure.

4.8 Allowable Stresses

The design stress-intensity values for metallic construction materials shall be obtained from ASME Boiler and Pressure Vessel Code, Section III Division 1, Subsection NB.

4.9 Materials

Materials selected for the core support structure shall be compatible with the environment as specified in 5040500000. In addition the grid plate material shall be selected (or protection provided) such that self welding between fuel assembly and grid plate does not occur.

5.0 SAFETY REQUIREMENTS

5.1 Design Classification

The Core Support Structure is a Seismic Category I item per 5040100000, Plant Safety Criteria.

6.0 INTERFACE REQUIREMENTS

The design of the core support structure is restricted by requirements imposed by components or adjacent structures. The major interface areas are tabulated below:

<u>Major Interface</u>	<u>Document No.</u>	<u>Interface Requirements</u>
Overall Plant Design	5010100000	Determine design life of core support structure.
PCRV Cavity Liner and Closures	5110200000	Loads at liner attachments.
PCRV Thermal Barrier	5110400000	Thermal barrier thickness and attachment requirements.
Primary Coolant System	5210000000	Design temperature and ΔP .

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6.0 INTERFACE REQUIREMENTS (Continued)

<u>Major Interface</u>	<u>Document No.</u>	<u>Interface Requirements</u>
Fuel Assembly	5170100000	Space envelope and loads.
Control Assembly	5170200000	
Blanket Assembly	5170300000	
In-Core Thermal Probe	5170500000	Requirement for core outlet temperature probes.
Plant Safety	5040100000	Defines safety requirements for core support structure, and classification of accidents.
Pressure Equalization System	5230200000	Transmission of fission gases from the fuel elements to the purification system.
Upper Shield Assembly	5130300000	Nuclear heating and irradiation.
Radial Shield Assembly	5130500000	
Plant Transient Conditions	5050200000	Operating transients and cyclic conditions.
Plant Operating Conditions	5050100000	Operating conditions.
In-Service Inspection	5080200000	See In-Service Inspection of Core Support Structure.
Seismic: NSSS	5040400000	Seismic requirements.
Environmental Qualification of NSS Equipment	5040500000	Material environmental compatibility.

7.0 REFERENCES

<u>Reference</u>	<u>Document Section</u>
ASME Code Section III, Division 1, Subsection NB	4.1a, 4.8
ASME Code Section III, Division 1, Subsection NG	4.1a
ASME Code Section XI	4.1a
ASTM E8	4.5a
ASTM E139	4.5a
LMFBR TID-26666	4.1b
RDT E15-2NB	4.1b

Notations in this column indicate where changes have been made

APPENDIX B

GA COMMENTS ON THE WESTINGHOUSE REVIEW
OF THE CORE SUPPORT STRUCTURE

A draft version of this report was submitted to the U. S. Department of Energy on November 11, 1977 for the purpose of review and comment prior to publication. Comments were prepared by the General Atomic Company and given to DOE. These comments, with their transmittal letter, are reproduced in the pages that follow.

A review of the comments was made and, where deemed appropriate, changes made to the body of the report. It should be recognized, however, that the Westinghouse review was based on the documents and design information made available and referenced in Section 8. Additional documents and design analysis, referenced in the General Atomic comments, were not available to Westinghouse and could not be evaluated, nor could conclusions attributed to those additional documents be used to modify the comments, criticisms, or suggestions resulting from the engineering review.

GENERAL ATOMIC COMPANY
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April 14, 1978

Mr. Louis M. Welshans
Chief, Fast Gas Reactors Branch
Division of Nuclear Research and Applications
Mail Station F-305
U.S. Department of Energy
Washington, D.C. 20545

Subject: GA Comments on the Westinghouse Review of the
Core Support Structure, GCFR Program, Contract
EY-76-C-03-0167, Project Agreement No. 23

Dear Lou:

Attached is a summary of the GA comments on the subject report. The general feeling about the report is that Westinghouse performed a very thorough review, and most of the questions raised are being addressed or are planned to be addressed.

We were particularly interested in the Westinghouse emphasis on the necessity for a thorough seismic response program. We concur in the stated development philosophy that we should "include extensive testing to complement the analysis," and we welcome suggestions for programs not already in our development plans.

We were also pleased at the inferred conclusion that both the 300 MWe plant and a larger commercial sized plant grid plate could be fabricated, although we are aware there are many difficulties in the process.

Many of the Westinghouse comments were based on an incomplete knowledge of the areas other than the core support, such as the core and control rod drive mechanisms. This results from GA's difficulties in supplying a consistent set of information due to the naturally uneven progression of the design. There are also difficulties in supplying information that confines the investigation to the core support system without unduly involving the other components that interact with the core support system.

Please feel free to have Westinghouse people communicate with us further if the comments need clarification.

Very truly yours,

C. E. Washington

C. E. Washington
Senior Project Engineer
GCFR Program

CEW/bjm
Att.

cc: J. B. Radcliffe; SAN-SD
J. J. Scoville; HBA

B-3

SECTION 2.0 FUNCTIONAL REQUIREMENTS

The core support structure consists of two major components, namely, the grid plate and the core support cylinder. The minor components such as bolts, seals, shear anchor, and flanges attached to the core support cylinder are understood to be included.

SECTION 2.1 (b) CORE SUPPORT GRID PLATE

At General Atomic (GA) it is not customary to list all interfacing components and all anticipated conditions in the design criteria. These will be defined in the design specification.

SECTION 2.2 (a) GRID PLATE SUPPORT STRUCTURE

Thermal loads, control rod deceleration loads, and seismic loads are included in "all anticipated loading conditions."

SECTION 2.2.1 MECHANICAL DESIGN

COMMENT 2

Tests are scheduled to investigate the fuel element assemblies natural frequency and flow induced vibration as documented in the GCFR Core Development Program, Ref. 1.

COMMENT 8

Proof tests of the flow by-pass seals between the core assemblies and the grid plate and the adhesion characteristics of various material couples representative of the seating surface between the core assemblies and the grid plate, with and without hard facing, have been in process for some time. Progress to date in these areas are as reported in the GCFR Quarterly Progress Reports.

COMMENT 10

The detail design of the pressure equilization system (PES) is in process and has been for some time. In recognition of the need for early establishment of the PES, the development program for the PES, Ref. 2, was identified well in advance of most other GCFR components. It is not envisioned that the PES will have no impact upon other components. The PES development program is designed to identify the detail requirements that need to be imposed.

SECTION 2.2.2 STRUCTURAL ANALYSIS

The static and dynamic effects of the helium jet exiting the core assemblies have been evaluated and have been shown to be negligible. The static effect of the jet was first evaluated in 1971 (Ref. 3) when it was shown that the maximum static deflection due to the jet thrust was 0.0002 inch,

resulting in a static stress at the clamped end of the assembly of less than 20 psi. The dynamic analysis performed in 1975 (Ref. 4), indicated that the assemblies are unconditionally stable to aeroclastic instabilities for coolant velocities up to 50 times greater than maximum flow.

Although considerable confidence is held in these analytic results, there are plans to perform flow tests on full scale mock-ups of GCFR core assemblies. It is considered inevitable that some very low amplitude vibration will occur as a result of coolant flow. However, the resulting stresses will be so low that fatigue will not be a problem, and the displacements will be so small that in the event that impacts between assemblies occur, any wear that may result will certainly be inconsequential.

SECTION 2.2.3 NUCLEAR INTERACTIONS

The effects of the "coolant jet forces at the exit of the unrestrained fuel assemblies" have been previously identified, quantified, and shown to be insignificant as noted in the response to Section 2.2.2, Comment 2.

Proof testing to determine the behavior of assemblies subsequent to 180° rotation performed to prevent gross bowing is an item that requires further consideration. In the present Core Development Program, there are plans to perform this type of proof test during the initial period of operation of the demonstration plant. This plan was considered to be valid when the development plan was designed. Since that time, however, new information has emerged which indicates the primary core structural material (20% cw type 316 ss) undergoes appreciably more irradiation induced swelling than thought at the time the development program was established. Consequently, a proof test of flow duct bowing as the result of irradiation induced swelling with 180° rotations needs to be further evaluated.

The needed analyses of fuel behavior, PCMI effects and cladding structural response resulting from the step power changes that accompany the assembly rotations, have been performed and indicate acceptable conditions. In addition, an irradiation test has been initiated in the EBR-II reactor to experimentally evaluate the effects of step power increases on proto-typical GCFR fuel rods.

The effects of bowed fuel assemblies on fuel transfer operations and handling tools are being considered in the design of the operations and equipment.

SECTION 2.3 DEVELOPMENT PLAN

Many of the major concerns itemized in this section are core related and are covered in the established development programs, Ref. 1 and 2. Applicable responses are as follows:

o 4th Concern (Core Assembly Flow Induced Vibrations)

The core development plan, Ref. 1, currently includes tasks associated with determining the effects of flow induced vibrations and seismic excitations of the core assemblies. Any significant loading effects upon the grid plate would, of course, be included in the grid plate analyses.

o 5th Concern (Seismic Scram Capability)

Test verification of the seismic scram capability in terms of control rod insertability in the presence of seismic induced deflections, as well as all other forms of assembly distortions, are currently part of the core assembly test plans. Scram tests of the complete control rod system under maximum seismic deflection conditions will be part of the control rod drive system development program which has not yet been finalized. Also see response to Section 5.7.4.

o 6th Concern (Core Assembly Seating Surface)

Evaluations of the conical surfaces along the bottom edge of the grid plate are being separately addressed as part of the core development program. For example, adhesion tests for a variety of material couples have been completed which do demonstrate the desirability of some form of hard facing to minimize material transfer. Sample hard faced materials have been procured and testing will commence shortly.

o 7th Concern (Core Assembly Thermal Bowing)

The effects of thermal bowing of the fuel assemblies on the grid plate during a seismic response test are not considered to be particularly important since the thermal gradients are in the direction to generally reduce the interactions between assemblies. The interactions between core assemblies are being addressed but it should be noted that the presence of, or lack of, interactions between core assemblies would not have a pronounced effect upon the grid plate. If anything, the existence of interactions between core assemblies and the ensuring damping would be generally expected to reduce the magnitude of the reactions transmitted to the grid plate. Potential impacts between adjacent core assemblies during a seismic response is the area of most concern, but does not particularly affect the grid plate.

o 9th Concern (Seal Material Compatibility)

If the seal materials compatibility concern is directed at the flow by-pass seals on each core assembly, it should be noted that the core development program does address this question and, as noted earlier, tests are already in progress to evaluate the self-welding characteristics of various material couples in the helium environment.

o 10th Concern (Thermal Striping)

The effects of thermal striping in the GCFR where the core outlet coolant flows into a very large cavity beneath the core is significantly of less concern than in the LMFBR. There are virtually no structures in the mixing zone that would be subject to large numbers of cycles of rapidly changing surface temperatures. Furthermore, the significantly lower heat transfer coefficient of the GCFR helium coolant, as compared to the LMFBR sodium coolant virtually eliminates the potential for rapidly changing surface temperature of structural components.

SECTION 3.1 BYPASS FLOW

The limit on gas leakage is specified for plant efficiency purposes, and therefore, the other operating conditions are not of any particular concern.

SECTION 4.0 DESIGN CRITERIA

Section 3.3 In-service Inspection

1. The GA document should have been worded "to meet the intent of" Section X, Division 2 for GCRs (not GCFRs). The Code in its present form does not apply directly to gas breeder reactors, as stated in the Foreword.
2. Westinghouse (WH) comments on baseline inspection: Yes, a baseline inspection will be performed.
3. If and when the current Code is extended to include GCFR it can be presumed that present examination and surveillance methods will be applicable to core support structures of this reactor type.
4. The location of test specimens within the core cavity is a degree of design detail yet to be addressed. Similarly, placement and retrieval of these items are detailed features of plant operation outside the scope of the subject design criteria.
5. Monitoring of these seals to assess continued integrity is not required because they are not safety related equipment thus it is not a function of inservice inspection.

Section 4.1 Applicable Code Criteria (a)

Subsection NB requires that Section NA be used for the material properties.

Section 4.1 Applicable Code Criteria (b)

The seismic loading conditions are indicated in Section 4.7 of Ref. 5.

Section 4.2 Vibration

"Nuclear stability" refers to the reactivity changes caused by vibrations of the grid plate.

"Mechanical vibration" refers to the vibration induced by mechanical means, such as circulators or pumps.

The limits established for the flow and acoustic induced vibrations are general limits. The specific limits will be based on the Code allowables for stress and fatigue.

"Flow induced vibration" refers to all vibrations caused by the helium flow.

Acoustic vibration damage to the core support is not likely to be a problem. The most probable source is noise generated by the circulator. The effects of acoustic vibrations will be addressed in more detail at a later time.

Section 4.3 General Requirements

The comments made by WH are correct; in principle, the grid plate must meet deflection-limits and stress-limits. However, the reactivity change by the deflection of the grid plate governs the design.

The material property changes during the operating life of the grid plate have been taken into consideration as required in Section 4.1.

The Design Basis Depressurization Accident (DBDA) occurring simultaneously with either an operating basis earthquake or with a safe shutdown earthquake is a requirement and will be included in the Design Criteria.

Flow reversal can happen only if there is a hole in the reactor vessel greater than 75 sq. in. Owing to the design of the PCRV, and the use of flow restrictors, a free flow area above 75 sq. in. is not a credible event.

Thirty years design life is for the demonstration plant only.

In the presently favored concept the PES vent lines will be collected into a manifold plate which is seated on top of the grid plate. These design details are yet to be worked out.

Section 4.5 Radiation Damage

The seals are not a safety related item and at this time, the material of the seals has not been selected. When this is done, irradiation damage will be considered.

Section 4.6 Reactivity Change

Will be established at a later time.

Section 4.7 Loading Conditions (a)

These combinations will be specified in the design specification.

The use of 1/2 SSE in lieu of an OBE is consistent with earlier NRC Regulatory Guides. This is now obsolete, and the GA design criteria are being revised.

Section 4.7 Loading Conditions (b)

The ASME Code Section III does not specify how to classify or combine the SSE. This is a licensing issue.

Section 4.8 Allowable Stresses

"Allowable stresses" used in the GCFR core support structure criteria is a general item, which applies to any class components; the specific term "design stress intensity" was used in Ref. 5.

Section 4.9 Materials

The core support structure material is required to be selected in accordance with Subsection NG of the ASME Code, Section III. In addition, the material chosen shall be compatible with the GCFR environment.

Section 5.1 Design Classification

The basic seismic criteria are in Section 4.7 of Ref. 5. The more detailed criteria will be established later.

Seismic Criteria are only defined in the higher level document 50401 00000 "Plant Safety Criteria".

Section 6.0 Interface Requirements

Interface requirements cover all the interrelation with adjacent components and systems, including flow induced vibration loads and seismic loads.

SECTION 5.1.2 GRID PLATE

In general we agree with the comment. A survey of stainless steel forging suppliers in the U.S. and Japan was made to determine the feasibility of manufacturing a large disc forging. At least one U.S. and one Japanese supplier were found who can manufacture forgings of sizes well in excess of what is needed for the GCFR demonstration plant.

SECTION 5.2.6 FUEL ELEMENT SEAL

The design of the core assemblies has been revised to use piston rings for by-pass flow sealing rather than the conical surface at the bottom and surface of the grid plate. The conical seat has been retained only to function as a centering support point. As previously noted, tests are in progress to develop a satisfactory conical seat.

SECTION 5.2.7 FUEL ELEMENT INSERTION

The fuel assembly rigidity has been investigated to evaluate the feasibility of handling the fuel assemblies from their lower ends. The assemblies have sufficient rigidity as the result of the hexagonal flow duct so there is no concern associated with the ability of the upper end to follow the motion of the lower end.

SECTION 5.2.8 MATERIAL SELECTION

See prior comments concerning material compatibility of conical seating surface between fuel assembly and grid plate (Section 2.3, 6th Concern).

SECTION 5.2.9 SPRING MATERIAL

The temperature at the latching mechanism is only about 600°F and, as such, could not cause relaxation of an appropriately selected spring material.

SECTION 5.3.2 CORE SUPPORT STRUCTURE COMPONENTS ANALYZED

The stress analysis of the radial keys, bolts, and shear anchors will be

covered in the final design report. Reference 6 addresses only the major components which are the grid plate and the support cylinder.

SECTION 5.3.3 GRID PLATE ANALYSIS METHOD

We agree with the comment but, in the presently favored design of the GCFR grid plate the PES holes will be located in the manifold plate on top of the grid plate.

SECTION 5.3.4 GRID PLATE ANALYSIS MODELS

The main purpose of the analysis reported in Ref. 6 was to determine deflections. The correct nominal and ligament stresses is the subject of subsequent analyses.

SECTION 5.3.5 GRID PLATE BOUNDARY CONDITIONS

The boundary condition $W(r=R, z=0) = 0$, which means that the axial deflection of the mid-plane at the edge of the grid plate is zero, was used for simplicity. The expression WH proposed, namely $W(r, h/2) = 0$ is the exact condition, as discussed on Page 44 of Ref. 6.

The boundary condition: $\frac{\partial W^P}{\partial r}(R, z) = \frac{\partial W^S}{\partial r}(R_1, z)$ referred to by WH was not used in the GA report, Ref. 6. The correct boundary condition is shown on Page 54 of Ref. 6, Equation 80 as:

$$-\theta^P \Big|_{\substack{z=0 \\ r=R_1}} = \theta^r \Big|_{\substack{z=0 \\ r=R_1}}$$

This means that the slope of the plate at R_1 (at the junction between the perforated plate and the solid rim) is equal to the slope of the rim at the same location.

From the figure on Page 52 of Ref. 6, R_1 is clearly defined as the radius of the perforated plate.

The boundary condition:

$$\int_{-h/2}^{h/2} \sigma_{rr}(R, z) dz = 0$$

$$\int_{-h/2}^{h/2} \sigma_{rr}(R, z) z dz = 0$$

was never used at the junction of the perforated region and the solid rim, but is used at the edge of the grid plate for the simply supported condition, as shown on Page 38 of Ref. 6.

A simplified analysis was done and a more detailed one, including more exact modeling of the solid rim, will be performed at a later date.

SECTION 5.3.6 GRID PLATE DEFLECTION RELATED TO REACTIVITY CHANGE

The calculation reported in Ref. 6 is for the present design which has a pressure drop of 42 psi. For a different design with a higher pressure the thickness of the grid plate has to be increased to keep the deflection down and satisfy the limit on the reactivity change of 10¢.

The equation $R = L \left. \frac{dw}{dr} \right|_{r=R_c}$ at $z = 0$

is based on the geometry of the mid-plane of the grid plate. L is the distance from the mid-plane of the grid plate to the mid-plane of the core, which for the current design is 76.8 in. For a value of $L = 76.8$ in. and $P = 42$ psi the reactivity change is approximately -10¢. The numbers in Ref. 6 are based on an earlier design with a shorter core.

The reactivity change due to thermal bowing is not an instant phenomenon, and is, therefore, of less consequence than the pressure effect. Thermal and seismic effects are covered in more recent analyses (Ref. 7, 8, and 9).

The effect of seismic vibrations on reactivity through effective changes in the core radius have been analyzed in Ref. 10.

SECTION 5.3.7 CORE SUPPORT CYLINDER ANALYSIS

In the analysis the flange was modelled as an elementary ring and the cylinder as a thin shell, in which case the compatibility condition which was used is correct.

SECTION 5.3.8 INTERACTION BETWEEN GRID PLATE AND SUPPORT CYLINDER

Since the support cylinder is very thin compared to the grid plate (the thickness of the support cylinder is 1-3/4", the thickness of the grid plate is 24"), it is believed that the clamping effect on the grid plate is small. More detailed analysis, however, will be performed to verify this assumption.

SECTION 5.4.1 GRID PLATE MODELS USED FOR COMPARISON

The perforations were not pressurized in the test.

The effects of the core elements on the grid plate has been experimentally measured and the results were presented in Ref. 11.

To measure the ligament stress experimentally is one of the possible static tests being considered.

SECTION 5.4.2 BOUNDARY CONDITIONS FOR THE COMPARISON MODELS

In Section 4.3, Ref. 12, the axial displacements W_1 and W_2 are based on thin plate theory, assuming homogenous properties throughout the plate. The W_1 value is not obtained from equation (4) of Ref. 12. The value of the

ratio $W_2/W_1 = 0.4834$ was obtained on this basis, and is approximate, but is believed to be close to the exact solution. If W_1 were obtained from Equation 4, W_2 should be derived based on the same method.

SECTION 5.4.3 COMPARISON RESULTS

The comment is essentially correct; the deflection was measured from the bottom in the experiment and from the mid-plane in the analysis. The error made by this simplification, however, is only 2.5%, and not 5.37% as claimed by WH.

SECTION 5.4.4 VALIDITY OF COMPARISONS

See discussion in Section 5.4.3.

Westinghouse is correct in stating that the most important location from a reactivity standpoint is at the radial boundary of the core ($r = R_c$). However, in the experimental model there were some local effects introduced through the use of preloaded bolts, such that a comparison of deflections at this location would not have been meaningful.

SECTION 5.5.1 SEISMIC MODEL FOR CORE SUPPORT STRUCTURE

Although not included in the analysis reported in WH's Ref. 11, GA has done other analyses which include the effect of the support cylinder on the seismic response of the core. More detailed analytical work is planned for the future.

SECTION 5.5.2 SEISMIC MODEL FOR GRID PLATE AND CORE ELEMENTS

The orthotropic constants are general in equations 7 through 11 of Ref. 13. For the specific design configuration, Ref. 13, an effective modulus of elasticity and Poisson's ratio were used. These were obtained from the ASME Code using the graph based on the definition of the equivalent solid plate.

In equations 2, 10, 11, and 12 of Ref. 13, the displacement functions ψ_i ($i = 1, 2$) are understood to be general function of the radius (r) and circumferential (θ) coordinates, and time (t), otherwise equilibrium equations 7, 8, and 9 would not contain partial derivatives of θ (tangential) and time (t).

Equations 16, 17, and 18 can be reduced to equations 6, 7, 8, and 9 of Ref. 13.

In equation 19 a subindex r could be included, since equation 19 contains only the radial component of the displacement function W and the transfer shear function H .

SECTION 5.6 GRID PLATE SHIELDING

The current design does not include grid plate shielding in the radial blanket assemblies. Shielding analyses have, however, assumed grid plate shielding in the blanket assemblies. Consequently, it is now planned to revise the blanket assembly design to include grid plate shielding material.

SECTION 5.7.4 LATERAL OSCILLATIONS

The conclusion is reached in this section that "the seismic scram capability is a major area of feasibility uncertainty in the cantilever core support system." This conclusion was reached by comparison with the CRBR core, where even though the lateral motions are smaller because of the multi-level support, seismic impact loads made the seismic scrams marginal. Only by factoring in the effectiveness of the fluid damping and squeeze film lubrication of the surrounding sodium were acceptable results obtained.

The duct interaction effects in the GCFR are somewhat different than in the CRBR. In the case of the GCFR, the control rod and control rod guide duct are wholly contained within a control assembly which includes fuel rods and an outer flow duct encompassing the lot. As a result, interactions between core assemblies during a seismic event would have no significant effect upon the planar shape of the control rod guide duct and the resultant control rod scram capability. Lateral deflections of the guide duct would be expected to occur but these can be accommodated by design to not cause resistance to control rod insertion. Consequently, due to the somewhat different design approach adapted for the GCFR control assembly, no seismic scram feasibility uncertainty is considered to exist.

The scram feasibility during a seismic event is questioned. The following statements are pertinent:

1. The backup shutdown system provides motorized (forced) insertion as well as larger clearances to maximize seismic insertion reactivity.
2. The maximum reactivity insertion during a safe shutdown earthquake is an equivalent step reactivity insertion of $+10\%$. This effect would easily be compensated by the negative power coefficient and result in a new equilibrium condition well below fuel and cladding safety limits. Therefore, a scram delay to the end of a seismic vibration (the longest in recorded history is ~ 20 seconds) is acceptable.

SECTION 5.7.6 DYNAMIC EXCITATION

See response to Section 2.2.2, Comment 2.

SECTION 5.8 ROTATION OF FUEL ELEMENTS

Quantitative analyses have been performed on core assembly bowing and flow duct dilation resulting from irradiation induced metal swelling and neutron

enhanced creep. The analyses have been performed using the latest correlations for the irradiation effects. (The latest correlations used in the analyses are the most up-to-date available even as of this writing.) The latest results indicate that the interactions resulting from assembly bow are within acceptable bounds. In order to maintain the duct dilation within acceptable bounds (1/2 of the nominal interassembly gap) the duct wall has been thickened somewhat. Also see response to Section 2.2.3 regarding "Proof Testing."

SECTION 5.8.2 FUEL ROD PERFORMANCE

The fuel to cladding gaps quoted have been obsolete for some time. In addition, it has been shown quantitatively that the potential for PCMI in the GCFR pressure equalized and vented rod design is less than that in a sealed LMFBR rod design (see Ref. 14).

As noted in the response to Section 2.2.3, not only has confirmatory testing of step power changes been planned for the GCFR as is the case for LMFBR, the GCFR testing has actually been initiated in the EBR-II reactor. Rods that will be subjected to a step power increase are now being irradiated preparatory to relocating for a step power change.

SECTION 6.1 GENERAL

It is commented that pressure loads in the perforations of the grid plate need to be included in the plate design. As the result of using piston rings on the upper end of the core assembly nozzles to control core by-pass flow, pressure loading no longer exists in the perforation holes.

SECTION 6.4.2 SEISMIC RESPONSE TEST

With regard to the comment pertaining to verifying the scram capability, see responses to Section 2.3 (5th Concern) and Section 5.7.4.

SECTION 6.6 MATERIAL COMPATIBILITY EVALUATION

See response to Section 2.2.1, Comment 8.