Department of Energy
Washington, D.C. 20545
Docket No. 50-537
HQ:S:82:157

DEC 22 1982

Mr. Paul S. Check, Director
CRBR Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Check:

ADDITIONAL INFORMATION FROM NOVEMBER 22-24, 1982, MECHANICAL ENGINEERING BRANCH (MEB)/CLINCH RIVER BREEDER REACTOR PLANT (CRBRP) MEETING

Reference: HQ:S:82:143, J. R. Longenecker to P. S. Check, Subject: Meeting Summary, November 22-24, 1982, MEB/CRBRP Meeting, dated December 14, 1982

Enclosed are a new Preliminary Safety Analysis Report (PSAR) Table 3.2-4 and an amended PSAR Section 4.2 that respond to MEB items 3 and 41 from the reference letter. This information will be incorporated in Amendment 75 to the PSAR scheduled for January 1983.

Questions regarding this submittal may be directed to D. Robinson (FTS 626-6098) of the Oak Ridge Project Office staff.

Sincerely,

John R. Longenecker

John R. Longenecker
Acting Director, Office of
Breeder Demonstration Projects
Office of Nuclear Energy

2 Enclosures

cc: Service List
Standard Distribution
Licensing Distribution

D001
1/40

TABLE 3.2-4

PRELIMINARY LIST OF APPLICABLE CODES FOR NON-SAFETY CLASS MECHANICAL SYSTEMS/COMPONENTS

Systems/Components	Applicable Codes and Standards	Location
Reactor Refueling System		
Non-safety Related Equipment	ASME VIII/1, AISC	RCB, RSB
Supports	ASME III/NF3	RCB, RSB
Nuclear Island Maintenance Systems		
Vessels	ASME VIII/1	RCB, RSB
Supports	ASME III/NF3, AISC	RCB, RSB
Piping	ANSI B31.1	RCB, RSB
Valves	ANSI B31.1	RCB, RSB
Steam Generator System		
Water Dump Subsystem	ASME III/3	SGB
SWRPRS Piping (SGB wall to floor stack)	ANSI B31.1	SGB
Normal Chilled Water System		
Chillers Condensers, Evapcrators	ANSI B31.1	SGB
Piping and Valves	ANSI B31.1	Except RCB
Circulating Pumps	Manufacturers standards	SGB

TABLE 3.2-4 (Cont.)

PRELIMINARY LIST OF APPLICABLE CODES FOR NON-SAFETY CLASS MECHANICAL SYSTEMS/COMPONENTS

Systems/Components	Applicable Codes and Standards	Location
Radioactive Waste System		RSB, RWB
Tanks	ASME Sec. VIII/ API 650/620	
Piping & Valves	ANSI B31.1	
Pumps	Manufacturers standard	
Balance of Plant HVAC	ASHRAE SMACNA	PSB, TGB, MSW, CWPH, CB, SRH, FPPH
Main and Auxiliary Steam	ANSI B31.1	TGB, SGB
Heat Rejection System	ANSI B31.1 AWWA	TGB, Water Cooling Tower Yard
River Water System	ANSI B31.1 AWWA	River Water Pump House, Yard
Plant Service Water System		
Normal Plant Water System	ANSI B31.1	SGB/IB, CB, RSB/IB, TGB
Hot Water Heating System PSB,	ANSI B31.1	TGB, SGB, RSB, DGB, MSW
Treated Water System	ANSI B31.1 & Manufacturers standard	TGB, RSB, DGB, MSW, PSB SGB, MB, CB, GH

TABLE 3.2-4 (Cont.)

PRELIMINARY LIST OF APPLICABLE CODES FOR NON-SAFETY CLASS MECHANICAL SYSTEMS/COMPONENTS

Systems/Components	Applicable Codes and Standards	Location
Non-Sodium Fire Protection		
Sprinkler and Spray System	NFPA-13 & 15	RSB, MSW, PSB
Gas Blanket System	NFPA-12A	CB
Portable Fire Protection System	NFPA-10	SGB, CB, DGB, TGB, RSB, MSW, PSB
Dry Chemical Fire Protection	NFPA-17	SGB, CB, DGB, TGB, RSB, MSW, PSB
Fire Detection Alarm System	NFPA-72A, D & E	SGB, CB, DGB, TGB, RSB, MSW, PSB
Feedwater Water and Condensate System		
Feedwater Heater & Deaerator	ASME Section VIII, Division 1	TGB
Piping	ANSI B31.1	TGB
Startup drains piping and equipment	ANSI B31.1	TGB, SGB

UPDATE TO PSAR CHAPTER 4

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CHAPTER 4.0 - REACTOR

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The detailed formulation of the design criteria of Table 4.2-7 and the conditions for which they are applicable, are given in the CRBRP fuel assembly stress report, Reference 171.

INSERT 4.2-17

4.2.1.1.2.3 Requirements for Design Features

In addition to the preceding operational requirements, specific design features shall be incorporated into the fuel and blanket assembly designs to preclude the accident conditions discussed in Chapter 15.4 and any detrimental effects which could adversely affect the attainable design life.

1. Sufficient constraint shall be applied to the fuel and blanket rods to minimize fretting and wear at the support points.

2. The fuel and blanket assembly materials shall be compatible with adjoining materials and environmental conditions during their design lifetime. Where potential for excessive galling or self-welding exists, mating components shall be hard coated.

3. The relative location of the pellet column within the fuel and blanket rods shall be maintained during shipping to prevent damaging reactivity fluctuations during start-up by utilizing a properly designed axial spring support system.

4. The assembly axial support system shall maintain fuel and blanket assembly axial positions under all steady state and transient operating conditions, while providing for differential thermal expansion of the internal structures and the irradiation induced expansion of the assemblies. With the current CRBR baseline design, the limit is 2.5 inches at 70°f.

5. The inlet nozzles for the assemblies, in conjunction with the reactor internals, shall be designed with sufficient aperture redundancy to preclude total inlet blockage and to provide for adequate cooling even after total blockage of one inlet passage.

6. To prevent loading of a fuel or blanket assembly into a position where it is undercooled, the following situations must be prevented by a properly designed discrimination system.

    a. Fuel assembly insertion into a position which it would be undercooled, i.e., a position in which more coolant flow through the assembly is required for heat removal than can be admitted by the flow orifice in that assembly or receptacle.

    b. Fuel assembly insertion into positions in the core that are provided for the control rod assemblies, blanket assemblies and removable shield assemblies, except for those positions where fuel and inner blanket assemblies are intentionally interchangeable.

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New Section

4.2.1.1.2.2.2 Removable Radial Shield Assembly (RRS) Structural Component Design Criteria

The design criteria were selected so that the RRS structural components will satisfy their functional requirements described in Section 4.2.2.1.1.9 at the following operating and loading conditions over their design life within the constraints of the damage severity limits of Table 15.1.2-1.

(A) Operating and Loading Conditions

1. Thermal-Hydraulic:

Steady state and transient conditions shall be considered.

The umbrella transients for the shield assembly include normal, upset, emergency and faulted categories.

An appropriate number of events will be determined for the hottest shield assemblies based upon the designed replacement schedule, since some of the shield assemblies will have less than a 30 year life.

2. Mechanical Loads

For determining loading sources to be considered in the structural evaluation of Shield Assembly Duct structures; core restraint, seismic, thermal and miscellaneous effects are considered

These loadings represent a generalization of all the currently postulated loading mechanisms applicable to CRBRP core assembly duct structures. While some of these loadings may be negligible or inapplicable to the shield assemblies, they nevertheless form the minimum basis for the conservative evaluation of the design adequacy of the shield assembly duct structures. The determination of structural loads shall utilize the environmental conditions in accordance with the requirements of Table 15.1.2-1

3. Nuclear Environment

Table 4.2-69 summarizes the range of neutron environment of the shield assemblies.

(B) Structural Design Criteria

The shield assembly is not currently covered directly by an existing structural design code. The shield assembly structural design criteria are basically similar to those in 4.2.1.1.2.2.1 for fuel and blanket assemblies. The following subsections describe specific requirements which are used for the design of the RRS.

1. General Criteria for All Shield Assembly Structures

The general structural criteria which are used in the shield assembly structural evaluation are defined in Table 4.2-70. The shield assembly structural components shall be designed so that deformation due to mechanical loading, thermal expansion, and neutron irradiation induced swelling and creep does not produce gross interference with adjacent components such that the equipment functional requirements of 4.2.2.1.1.9 cannot be satisfied. The effect of loss of carbon, nitrogen, and alloying elements shall be considered when determining the strength of the material.

2. Component Structural Design Criteria

Shield Assembly Inlet Hardware:

In the inlet hardware (nozzle, transition, orifice, rod support), where maximum temperatures do not exceed 800°F, the base metal is ductile and the time dependent thermal creep is expected to be negligible. Time independent ductile failure modes covered by the

applicable ASME Boiler and Pressure Vessel Code are used as the basis for structural design requirements.

Consequences of shield assembly inlet hardware loss of structural integrity are significantly less severe than those for pressure boundaries and permanent components. Therefore, the emergency stress limits for shield assembly inlet hardware are higher than those given by the ASME B&PV code(1), but the maximum stress intensity shall not exceed the minimum ultimate tensile strength.

Shield Assembly Outlet Hardware:

The material is expected to be ductile and thermal creep may be significant. Both the time-independent and time-dependent ductile failure modes covered by the applicable ASME B&PV code, code cases and RDT standards are used as the basis for the structural design requirements.

The time independent stress limits for normal and uspet condition specified by the ASME B&PV Code are used directly for outlet nozzle design.

Thermal fatigue failure and creep related failures (creep rupture, excessive strain, creep fatigue) are prevented by imposing the appropriate design limits in Table 4.2-70.

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(1) The ASME B&PV Code is applicable to pressure boundaries and permanent components. The shield assemblies are not pressure boundaries and are removable.

3. Shield Assembly Duct

Where a minimum uniform elongation greater than 3% cannot be demonstrated, brittle fracture is considered as a potential failure mode and specific analytical methods to be used in fracture mechanics evaluations are developed.

4. Design Limits for Inelastic Analysis

Because of the inherent simplifications in elastic and simplified inelastic analyses which are offset by additional conservatism in the corresponding limits, inelastic analysis methods may be used in lieu of the elastic analysis methods. The structural criteria are given in Table 4.2-70.

5. Special Weld Requirements

Since all of the joining is accomplished by mechanically pinning the components and seal welding, there are no structural weldments in the shield assembly.

REPLACE WITH INSERT 4.2-127

58 4.2.2.1.1.9 Removable Radial Shielding (RRS)

The removable radial shield assemblies are core assemblies which mechanically interface with the radial blanket assemblies, the core former structure, the inlet and flow bypass modules, and the upper internals. These assemblies have the following functional requirements:

a) Attenuate neutron fluence to levels consistent with 30 year lifetime for peripheral components, based on residual total elongation ductility limits.

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b) Maximize solid volume fraction within the limits required to provide adequate cooling. To obtain adequate shielding with 316SS 80% minimum solid volume is required.

c) Maintain the RRS structural integrity by limiting RRS lifetime to assure 1.0 percent minimum residual ductility based on total elongation. In addition maximum operating steady state plus transient strains must be less than 0.056 percent at a biaxial stress ratio (longitudinal to circumferential) of 1:1 and less than 0.33 percent at a 1:0 ratio. The allowable strain varies approximately linearly with the biaxial stress ratio.

d) The removable radial shield assemblies are to be installed and removed by normal fuel handling equipment.

e) Transmit lateral core restraint loads without contributing significantly to the magnitude of these loads.

f) Provide circulation path for cooling to preserve structural integrity.

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4.2.2.1.1.9 Removable Radial Shield (RRS)

The removable radial shield assemblies are core assemblies (See Figure 4.2-43) which mechanically interface with the radial blanket assemblies, the core former structure, the lower inlet modules, the bypass flow modules, and the upper internals. These assemblies have the following functional requirements:

a) Provide radiation shielding to ensure the structural integrity of the reactor permanent components beyond the radius of the RRS for 30 years.

b) Provide a compact structural unit that can be handled in and out of the reactor by the normal fuel handling equipment.

c) Transmit lateral core restraint loads without contributing significantly to the magnitude of these loads.

d) Provide flow paths ~~for the sodium cooling~~ for cooling of the RRS to preserve structural integrity.

e) Provide a means for locating surveillance specimens in the RRS and a method for their expedient recovery.

The bases for these requirements are described in 4.2.2.1.2.9.

m. Requirement - Provide vertical load reaction to the core barrel.

Bases - The core former structure is subjected to mechanical loads caused by the incremental thermal growth of the fuel assemblies. Due to stick-slip friction of the assemblies and alternating heating and cooling cycles, a net upward force may be developed on the core former ring. Additionally, the SMBDB loadings can cause upward forces on the upper core former.

~~4.2.2.1.2.9 Removable Radial Shielding (RRS)~~

~~a) Requirement - Attenuate neutron fluence to levels consistent with a 30 year lifetime for peripheral components, based on the following residual total elongation ductility limits:~~

~~Shielding Material Remote from Attachments~~	~~5%~~
~~Shielding Material Attachments~~	~~10%~~
~~Core Barrel, Core Formers, and Vessel~~	~~10%~~

~~Bases - The components beyond the radius of the RRS assemblies have thirty-year lifetimes. This includes fixed radial shielding, the core barrel, core formers, and the reactor vessel as the major items. For each of these items, there are limits established for maximum allowable fluence to assure a certain level of residual ductility of the structural material after thirty years. Approximate fluence, (total, E>0 MeV) limits to obtain the above ductility limits (Ref. 1) are as follows:~~

~~Structural fixed radial shielding (with ten percent ductility at 800°F) has a fluence limit of 1.3×10^{22} (n/cm^2).~~

REPLACE WITH $\boxed{\text{INSERT 4.2-151}}$

Amend. 51
Sept. 1979

4.2.2.1.2.9 Removable Radial Shield (RRS)

a) Requirement - Provide radiation shielding to ensure the structural integrity of the reactor permanent components beyond the radius of the RRS for 30 years.

Bases - The structural integrity of the reactor permanent components is based on the requirement that at the end of the 30 year life the minimum total residual elongation of the materials must be not less than 10%.

b) Requirement - Provide a compact structural unit that can be handled in and out of the reactor by the normal fuel handling equipment.

Bases - Efficient and economical means of installing and removing the RRS by the fuel handling equipment. This is achieved by commonality with the fuel and blanket assembly design.

c) Requirement - Transmit lateral core restraint loads without significant contribution to the magnitude of these loads.

Bases - The load buildup at "on power" conditions must be minimized to increase the margin of safety against duct crushing. Also, it is desired to attain lower withdrawal forces at reactor refueling conditions.

d) Requirement - Provide flow paths for the sodium for cooling of the RRS to preserve the structural integrity.

Bases - Coolant flow must be provided inside the assembly to maintain the maximum material temperature below a reasonable limit (~1100°F), and to limit the maximum temperature gradients due to nuclear heating so that the resulting thermal stresses will not exceed the allowable limit.

e) Requirement - Provide a means for locating surveillance specimens in the RRS and a method for their expedient recovery.

INSERT 4.2-151 (Page 2 of 2)

<u>Bases</u> - As described in 4.2.2.1.1.12, surveillance is required to obtain information on materials irradiation damage so that changes in material properties can be monitored and potential deteriorating conditions detected.

REPLACED BY INSERT 4.2-151

~~Nonstructural fixed radial shielding (with five percent ductility at 800°F) has a fluence limit of 2.6×10^{22} (n/cm^2).~~

~~The Core Barrel and Vessel (with ten percent ductility at 800°F) have a fluence limit of 1.3×10^{22} (n/cm^2).~~

~~b) Requirement - Maximize solid volume fraction within the limits required to provide adequate cooling. To obtain adequate shielding with 316 SS eighty percent minimum solid volume is required.~~

~~Bases - Based on 316 SS, the minimum solid volume fraction is 80% to obtain the required residual ductility of the midpoint of the core, with the fixed radial shielding and the core barrel at five percent and ten percent ductility respectively.~~

~~c) Requirement - Maintain the RRS structural integrity by limiting RRS lifetime to assure 1.0 percent minimum residual ductility based on total elongation. In addition maximum operation steady-state plus transient strains must be less than 0.056 percent at a biaxial stress ratio (longitudinal to circumferential) of 1:1 and less than 0.33 percent at a 1:0 ratio. The allowable strain varies approximately linearly with the biaxial stress ratio.~~

~~Bases - The strain limits given above are considered conservative since a safety factor of three was applied to rupture strain predicted from tensile testing to determine allowable strain. This safety factor is consistent with safety factor levels employed in ASME Code criteria. It has been observed from stressed tube experiments that rupture strain varies with stress state. For a 1:1 biaxial stress ratio the predicted rupture strain is only one-sixth that of uniaxial tension tests. Thus this factor has been included.~~

~~The minimum total fluence (E>0 MeV) associated with 1.0 percent residual total elongation is 0.875×10^{23} (n/cm^2) at 800°F and 1.2×10^{23} (n/cm^2) at 900°F (Ref. 29). The use of energy dependent damage functions will increase the fluence limits beyond those given above.~~

~~The minimum RRS lifetimes based on the one percent ductility limit are as follows:~~

RRS Row Number	Lifetime (Years) Depending on Position Within Row
1	4.9 to 8.8
2	8.8 to 16.9
3	15.8 to 31.6
4	30.0 to 61.1

REPLACED BY INSERT 4.2-151

Thus some of the assemblies in row 3 and all of the assemblies in row 4 will not need to be replaced. A surveillance program is expected to extend the lifetimes beyond the minimum predicted lifetimes.

The above strain and ductility limits represent reasonable criteria which reduce complexity and expense of RRS assemblies and increase reliability as compared to less stringent criteria.

d) Requirement - The removable radial shield assemblies are to be installed and removed by normal fuel handling equipment.

Bases - This requirement affords efficient and economical means of installing and removing the removable radial shielding by using the existing fuel handling equipment.

e) Requirement - Transmit lateral core restraint loads without significant contribution to the magnitude of these loads.

Bases - It is desired to minimize load buildup at "on power" conditions to provide a margin of safety against duct crushing. Also, it is desired to attain lower withdrawal forces at reactor refueling conditions.

f) Requirement - Provide coolant paths for circulation cooling to preserve the structural integrity of the assemblies.

Bases - Sufficient coolant must be provided for the following two reasons. The maximum material temperature must be kept within reasonable limits (approximately 1100°F) and the maximum allowable temperature gradients from internal nuclear heating

51 must be determined by examining the allowable thermal stresses.

4.2.2.1.2.10 Core Former Structure (CFS)

a) Requirement - Provide peripheral constraint for the reactor assemblies.

Bases - Positioning of the core former structure relative to the core maintains, as part of the core restraint system, proper core geometry during all modes of operation.

b) Requirement - Provide lateral location and constraint for the lower end of the Upper Internals Structure.

Bases - Maintain alignment of the Upper Internals Structure for proper operation of the reactor control system. Interface constraints are imposed by the In-Vessel Transfer Machine, the reactor vessel,

58 and the rotating plugs.

Amend. 58
Nov. 1980

The RSS assemblies are removable core assemblies (See Figure 4.2-43) having a structure basically similar to the fuel and blanket assembly structures consisting of inlet nozzle, hexagonal duct tube and outlet nozzle

4.2.2.2.1.9 Removable Radial Shield (RRS)

The ~~radial~~ shield material is ~~assemblies are~~ made up of stainless steel rods ~~held~~ within thin walled ~~hexagonal~~ ducts. These assemblies are designed to be as flexible as possible in order not to contribute to the off-power restraint loads. A close-fitting support block is inserted inside the duct at the ACLP to provide axial restraint for the shield rods and to absorb seismic loads that are transmitted through the ACLP to the core former. Control of flow through the RRS is provided by a stack of orifice plates located inside the inlet nozzle.

4.2.2.2.1.10 Core Former Structure

The core former structure is composed of three substructures, the upper core former ring, a spacing cylinder, and the lower core former ring. The core former rings are comprised of profile milled segments assembled into continuous rings, as illustrated in Figure 4.2-46. The above core load plane former ring, called the lower core former ring, is mounted on a ledge machined in the inner diameter of the core barrel. The spacing cylinder, called the support ring, provides holddown for the lower core former ring and support for the top load plane former ring called the upper core former ring. The upper core former ring has six lugs that fit slots in the top of the core barrel to transmit seismic and other loads to the core barrel. A series of L-shaped keys are circumferentially slipped into the groove on the inside of the core barrel, between each of the six lugs, and trapped by means of a radially oriented dowell pin on either side of each slot. These L-shaped keys prevent vertical displacement of the core former rings. The upper core former ring is centered in the core barrel cavity by means of the six radial lugs. The lower core former ring is centered in the core barrel cavity by means of radial shims.

4.2.2.2.1.11 Maintainability

All the reactor internals except for the removable core ~~reactor~~ assemblies, are designed for a 30 year life with no scheduled maintenance. However, provision has been made to permit removal of the lower inlet modules to assure full plant life and malfunction recovery capability. Contributing factors which may require malfunction recovery capability include:

1. Potential damage to the reactor assembly receptacles, as a result of insertion and removal of reactor assemblies.

2. Potential wear or partial plugging of strainers and orifices, as a result of coolant induced changes.

58 4.2.2.2.1.12 Surveillance

Material surveillance coupons are contained within special ~~assemblies located in~~ removable radial shield ~~positions~~ assemblies and a fuel
51 transfer and storage assembly. In addition to these special assemblies,

4.2.2.3.1.3 <u>Horizontal Baffle (HB), Fuel Transfer & Storage Assembly (FT&SA), and Fixed Radial Shield (FRS)</u>

The HB, FT&SA, and FRS are internals structures and are not covered by mandatory Code rules, but the Owner's designee has required that the rules stated in 4.2.2.3.1 be applied to the design and analysis of these

INSERT 4.2-176a

New Section

4.2.2.3.1.4 Removable Radial Shield (RRS)

The RRS is a replaceable core component with a structure basically the same as those of the fuel and blanket assemblies. Therefore, the structural design criteria applicable for the RRS are also basically similar to those of the fuel and blanket assemblies, rather than those described in 4.2.2.3 for the permanent Reactor Vessel Internals components and structures. The RRS design criteria are described in 4.2.1.1.2.2.2.

Conclusions and Future Work

Assembly motion reactivity effects are conservatively predicted by current analytical procedures. The results shown indicate that reactivity related core restraint requirements of Section 4.2.2.1.2.8 are satisfied.

Core component contact loads and distortions are predictable using current analytical methods. Additional work is planned to verify dilation induced duct-to-duct contact predicted in NUBOW-3D with more detailed models.

Additional areas where further work is planned include:

1) Detailed analysis of core restraint performance beyond core 1.

2) The simulation of fuel management in the NUBOW-3D model.

3) Verification and improvement if necessary of the duct dilation induced duct-to-duct contact model. REPLACE WITH $\boxed{\text{INSERT 4.2-227}}$

~~4.2.2.4.4 Removable Radial Shielding (RRS)~~

~~The removable radial shielding is in a preliminary phase of design; thus, stress analysis taking into account the effects of environmental conditions has not yet been completed. Analysis will be conducted on the following considerations: thermal stresses and strains, refueling and handling stresses, strain limits for brittle material, and effects of irradiation induced swelling and creep on the core restraint system.~~

~~Preliminary thermal analysis has shown the maximum RRS temperature to be less than 920°F when cooled with bypass flow by natural circulation. Thus, it appears that bypass flow will provide adequate cooling.~~

~~More detailed analysis is now in progress in both the thermal and stress categories.~~

~~Shielding analysis, to date, has shown a solid volume fraction of at least 80% 316 stainless steel is required to meet ductility requirements on the fixed radial shielding and core barrel. This solid volume fraction is compatible with coolant channel volume requirements for adequate cooling.~~

4.2.2.5 Welding and Seizing of Reactor Internal Parts

The design considerations for welding and seizing of rotating or moving parts for reactor internals are presented in Table 4.2-64.

4.2.2.4.4 Removable Radial Shield Assembly (RRS)

The RRS configuration is shown in Figure 4.2-43. The structural analyses included two critical regions of the RRS, the Above Core Load Pad (ACLP) of the duct and the outlet nozzle. The selection of these two regions was based on structural analyses of the basically similar fuel and blanket assemblies which indicated that the assemblies inlet regions comprising the inlet nozzle, orifice assembly and transition had very high design margins, while operating under more severe environmental conditions than the RRS. The above mentioned structural analyses indicate that the ACLP is the critically loaded region.

In the ACLP analyses, maximum values from all loading sources were assumed to occur simultaneously at the single assembly subject to the worst environmental conditions. The various loading conditions considered were seismic, core restraint, and steady state and transient thermal loads. Three different sets of thermal-hydraulic conditions were considered from which the worst combinations of maximum duct midwall and cross-duct gradient temperatures was obtained. Stresses, strains and damage were determined for the ACLP resulting from the above loading sources and conditions.

An ANSYS finite element computer code was used for the RRS duct stress analyses. Based on the results of this analysis, inelastic and fracture toughness analyses of the ACLP were also performed.

The inelastic analyses were performed to determine strain and damage of the ACLP using the CHERN elastic-plastic-creep computer program. Two separate sets of analyses were performed, one representing BOL conditions, and one representing EOL conditions corresponding to a design life of 10 years. Twenty out of the 190 loading cycles of the unit histogram (see Figure 4.2-43A) were run for each of the two conditions to overcome the transient portions of the elastic-plastic-creep domain, and the remaining 170 cycles were conservatively duplicated by the 20th cycle. Both the calculated total maximum strain and the calculated maximum creep-fatigue damage when compared with the minimum allowable values yield positive design margins equal to 14.0 and 0.14, respectively.

A fracture toughness analysis of the duct ACLP was also performed, since the minimum elongation in the load pad at end of life is $\leq 3\%$, and brittle fracture is a potential failure mode. For this analysis, an initial flaw size and shape, and two orientations were assumed, consistent with the requirements defined in RDT E6-20T (now NE E6-20T). The analyses consisted of the following steps:

a) The critical stress intensity factor K_{IC} for BOL and EOL material conditions was derived from data on unirradiated sub-compact test specimens made from FFTF duct material.

b) The stress intensity factor for the assumed semi-elliptical flaw under tensile stress was determined.

c) A crack propagation analysis was performed to determine the crack growth under the applied duty cycle.

d) The maximum stress intensity factor K was determined for the final flaw size obtained from the crack propagation analysis.

e) Finally, the margin of safety was determined by comparing the calculated maximum EOL stress intensity factor K_{max} with the allowable design value which was obtained by applying a design margin of 1.5 on the calculated critical stress intensity factor K_{IC}. This evaluation yielded a positive margin of safety equal to 5.7.

ADD

TABLE 4.2-~~69~~ 70

TYPICAL NEUTRON ENVIRONMENT IN THE CRBR SHIELD ASSEMBLIES

RRSA Region	Elevation (inch)	Total Neutron Flux, E> 0.0 MeV (n/cm^2-sec)		Fast Neutron Flux, E >0.1 MeV (n/cm^2-sec)	
		Maximum	Minimum	Maximum	Minimum
Outlet Nozzle	-344.15	2.3×10^{11}	8.1×10^{10}	1.5×10^{10}	3.0×10^{9}
Sodium Filled Duct	-405.15	6.1×10^{13}	8.1×10^{12}	1.5×10^{13}	1.1×10^{12}
Above Core Load Pad	-411.15	1.4×10^{14}	7.0×10^{12}	4.4×10^{13}	1.4×10^{12}
Shield Rod (Core Mid Plane)	-437.15	1.2×10^{15}	2.4×10^{13}	5.1×10^{14}	5.7×10^{12}
Inlet Nozzle Transition	-469.15	1.3×10^{14}	5.5×10^{12}	3.4×10^{13}	1.1×10^{12}
Inlet Nozzle	-512.15	1.3×10^{11}	1.5×10^{11}	1.2×10^{10}	9.9×10^{9}

ADD

TABLE 4.2-~~70~~ 71

SHIELD ASSEMBLY INELASTIC STRAIN CRITERIA

Category	Normal and Upset Limits	Normal, Upset and Emergency Limits
Membrane Plastic Strain Limit	$\Sigma \left(\frac{\delta\varepsilon_m}{\varepsilon_u}\right) \leq 0.25$	$\Sigma \left(\frac{\delta\varepsilon_m}{\varepsilon_u}\right) \leq 0.35$
Total Inelastic Strain Limit	$\Sigma \left(\frac{\delta\varepsilon_t}{\varepsilon_{f/TF}}\right) \leq 0.5$	$\Sigma \left(\frac{\delta\varepsilon_t}{\varepsilon_{f/TF}}\right) \leq 0.7$
Creep-Fatigue Damage	---	Defined in Table 4.2-7

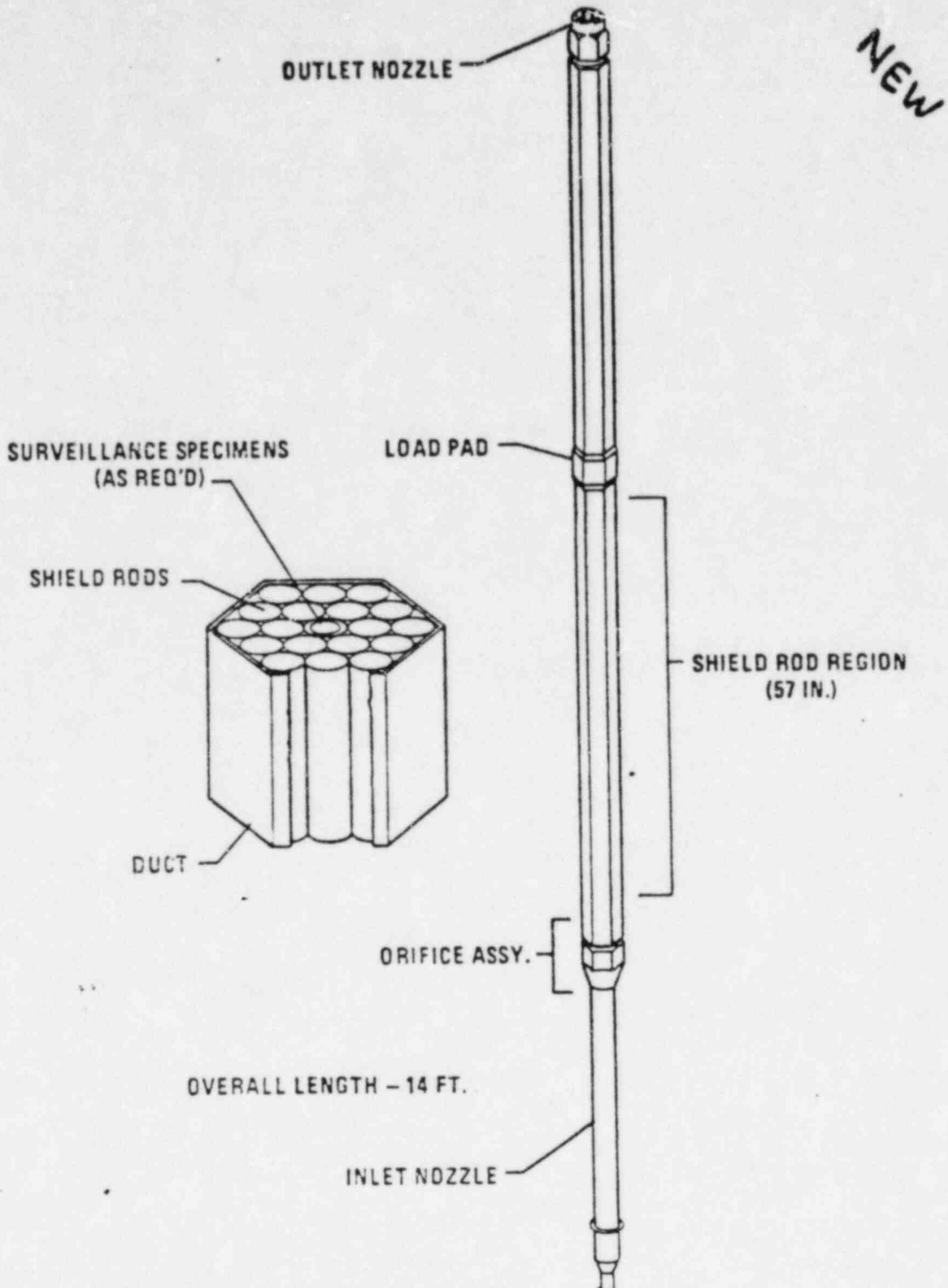
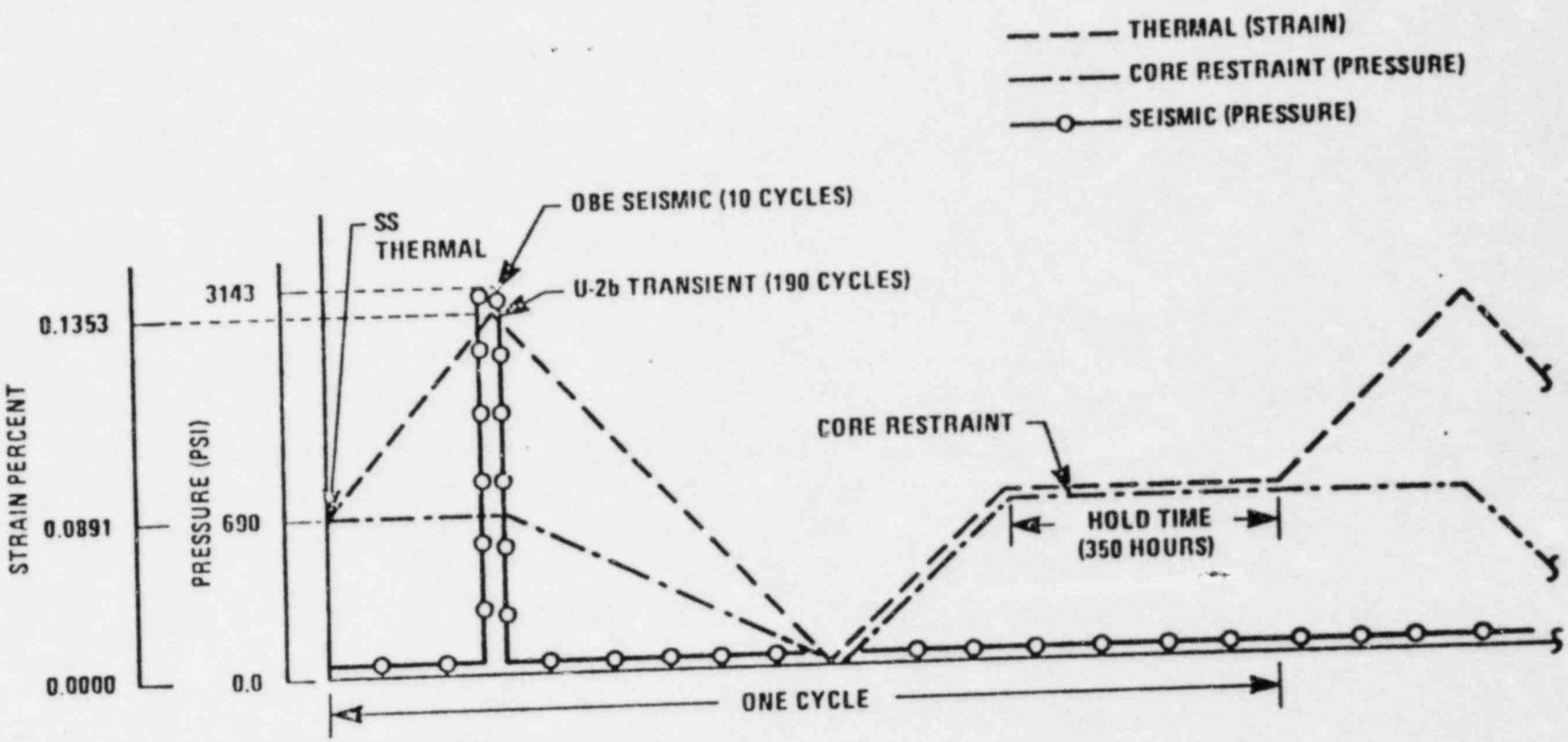


Figure 4.2-43. Removable Radial Shield Assembly

7954-4



NEW

Figure 4.2-43A. Loading Histogram Used in Inelastic Analyses of RRS ACLP

7954-1