MASTER

DESIGN EXPERIENCE -- CRBRP RADIATION SHIELDING

R. K. DISNEY
T. C. CHAN
F. G. GALLO
L. R. HEDGECOCK
C. A. McGINNIS
G. N. WRIGHTS

conf. 781117--7

NOVEMBER 1978

For Presentation at the US/USSR Joint
Fast Breeder Reactor Shielding Seminar

Obninsk, USSR

NOVEMBER 13-16, 1978

DISCLAIMER

DESIGN EXPERIENCE -- CRBRP RADIATION SHIELDING

R. K. DISNEY
T. C. CHAN
F. G. GALLO
L. R. HEDGECOCK
C. A. McGINNIS
G. N. WRIGHTS

WESTINGHOUSE ELECTRIC CORPORATION
Advanced Reactors Division
P.O. Box 158
Madison, Pennsylvania USA

ABSTRACT

The Clinch River Breeder Reactor Plant (CRBRP) is being designed as a fast breeder demonstration project in the U.S. Liquid Metal Fast Breeder Reactor (LMFBR) program. Radiation shielding design of the facility, evolving from the design experience gained in the USA Fast Flux Test Facility (FFTF) Project, consists of a comprehensive design approach to assure compliance with design and government regulatory requirements. Studies conducted during the CRBRP design process involved the aspects of radiation shielding dealing with protection of components, systems, and personnel from radiation exposure. Achievement of feasible designs, while considering the mechanical, structural, nuclear, and thermal performance of the component or system, has required judicious trade-offs in radiation shielding performance. Specific design problems which have been addressed are in-vessel radial shielding to protect permanent core support structures, flux monitor system shielding to isolate flux monitoring systems for extraneous background sources, reactor vessel support shielding to allow personnel access to the closure head during full power operation, and primary heat transport system pipe chaseway shielding to limit intermediate heat transport system sodium system coolant activation. The shielding design solutions to these problems defined a need for prototypic or benchmark experiments to provide assurance of the predicted shielding performance of selected design solutions and the verification of design methodology. An experimental program utilizing facilities at the Oak Ridge National Laboratory (ORNL) Tower Shielding Facility was conducted in parallel with the design phase of CRBRP. Design activities of CRBRP plant components and systems, which have the potential for radiation exposure of plant personnel during operation or maintenance, are controlled by a design review process related to radiation shielding. The program implements design objectives, design requirements, and cost/benefit guidelines to assure that radiation exposures will be "as low as reasonably achievable" (ALARA).

1. INTRODUCTION

The Clinch River Breeder Reactor Plant (CRBRP) is being designed as a U.S. demonstration of a liquid metal fast breeder plant for electrical power production. A major objective is to obtain data on the design, component fabrication, and construction of a licensed LMFBR on a utility grid. The CRBRP plant is a loop type sodium cooled system with a design thermal power of 975 Mwt. The plant consists of a reactor system with three independent primary and intermediate heat transport systems providing superheated steam to the turbine-generator system.

A major design objective for the reactor core is to provide flexibility in accommodating alternative fuel and blanket assembly configurations and fuel management schemes for optimization of plant performance parameters. The reactor core is an array of hexagonal fuel assemblies surrounded by radial blanket assemblies of similar dimensions. The reactor core fissile zone is approximately 0.91 meters (3 feet) high and is bounded by an upper and lower 0.36 meter (14 inch) axial blanket region. The reactor core (fuel and blanket) assemblies are surrounded by removable and fixed radial shielding. Removable radial shielding (RRS) assemblies have the same outer dimensional envelope as fuel and blanket assemblies, thereby providing the capability to remove and replace any assembly with the in-vessel and ex-vessel refueling machines during the 30 year life of the reactor plant.

A plan view and an elevation view of the reactor core are shown in Figures 1 and 2, respectively. The reference CRBRP reactor core design utilizes two enrichment zones of fuel assemblies (198 assemblies) surrounded by 150 radial blanket assemblies. A total of 324 removable radial shield (RRS) assemblies in four rows are required, as shown in Figure 1. The removable radial shield assemblies, which protect the core restraint system from neutron radiation damage, are supplemented by a fixed radial shield (FRS) to protect the core support system.

The reactor core, RRS, and FRS are supported by the core support structure which is supported within the reactor vessel, as shown in Figure 3. The reactor vessel is a cylindrical vessel approximately 6.2 meters (20 feet) in diameter and 17.3 meters (57 feet) high. The vessel closure head is an all-steel assembly utilizing a triple rotating plug configuration to provide in-vessel core assembly handling with a through-the-head non-articulated refueling machine. The entire reactor system is top-supported from the support ledge through the vessel flange and a vessel support system as depicted in Figure 4.

The CRBRP reactor system is near the center of the reactor containment building. Access to the closure head assembly and support ledge is provided through the head access area (HAA), which is open to the reactor containment building. Primary biological shielding external to the reactor vessel is provided by an ordinary concrete reactor cavity wall, as shown in Figure 4.

Key areas requiring extensive shielding design analysis and solutions are illustrated in Figure 4. As shown, the principal areas are: 1) in-vessel radial shielding, 2) closure head assembly and its penetrations; 3) reactor vessel support area and support ledge; and 4) primary heat transport system pipe chaseways. In addition, areas of the overall plant which required shielding design solutions include: 1) auxiliary piping penetrations, 2) heating and ventilating system penetrations; 3) radioactive cover gas processing and purification systems; and 4) fuel handling system equipment shielding.

Radiation shielding design of the CRBRP utilized the design experience gained in the FFTF program. This design approach was possible due to the similarity in physical size and design concept of the reactor system and the vessel support concept.

Design experience in the key shielding areas is discussed in the following sections.

2. REACTOR SYSTEM SHIELDING

The major objective of the in-vessel shielding of the CRBRP reactor system is to minimize the neutron fluence levels at permanent reactor structures. The design must assure that materials of construction for permanent components have an end-of-life ductility consistent with a threshold for brittle fracture criteria. The primary design constraint placed on the system, which required an optimization of the radial shielding, was that the core barrel diameter be minimized to provide an in-vessel transfer position for core assembly handling with a through-the-head refueling concept. This constraint was dictated by the constraint to use a reactor vessel diameter similar to the FFTF reactor vessel. Additional design constraints were: 1) austentic stainless steel or Inconel materials were required due to their compatibility with the sodium coolant; 2) replaceable shield assemblies must have the same dimensional envelope as core assemblies to facilitate handling in the refueling machines, 3) neutron attenuation provided by radial shielding must be minimized in order to maximize the neutron foreground of the reactor core at the ex-vessel flux monitoring system during reactor shutdown and refueling operations, and 4) the mass of replaceable shielding assemblies must be minimized to limit seismic loads (lateral) on the core assemblies. Due to the severe constraints imposed on the shield design, an extensive program to define irradiated material neutron fluence/ductility information was conducted in parallel with the CRBRP shield design phase.

Materials of construction for the CRBRP reactor system permanent structures were selected on the basis of previous performance in fast reactor and liquid sodium environments. The design criteria for materials selected is an end-of-life ductility requirement based on a threshold for brittle fracture criteria. The threshold of ductility chosen, 10% total elongation, is a level that assures ductile mode of failure and permits conventional structural analysis methods and criteria to be used in design. Design of the CRBRP in-vessel shielding is based on consideration of component neutron fluence residual ductility relationships, predicted two dimensional neutron flux distributions, and predicted component temperature conditions to provide component performance (end-of-life ductility and/or lifetime) consistent with design requirements. Preliminary in-vessel shield design was based on an interpretation of neutron radiation damage from fast-reactor irradiation-induced ductility changes based on a limited amount of test data obtained at USA fast reactor test facilities (i.e., Experimental Breeder Reactor II). Non-energy dependent neutron radiation damage analyses, used in preliminary design efforts, resulted in conservative shield designs that increased plant cost and required design solutions which did not satisfy all constraints. In order to relax shield requirements, a series of irradiation experiments were planned and conducted by Hanford Engineering Development Laboratory (HEDL). This program has increased the quality of test data and reduced data uncertainties. Measured material tensile properties provided needed irradiation effects data for the CRBRP out-of-core structural materials and provided damage data more applicable to the CRBRP nuclear and physical environment. Irradiations have been completed for 316SS, 304SS, 308L, and Alloy 718 materials. Test data in the total fluence range of 1×10^{21} - 1×10^{23} n/cm^2, as a function of fast flux fraction (E > 0.1 MeV) ranging from 0.4 to 0.9 MeV, have been obtained. Subsequent activities are to be directed toward the development of material property neutron energy dependent damage functions in order to further reduce the conservatism in the design criteria for LMFBR in-vessel shielding.

The primary mechanical constraint imposed on the in-vessel shielding design was due to the selection of the triple rotating plug closure head assembly (CHA) concept to provide for in-vessel core assembly handling with a through-the-closure head, non-articulated refueling machine. This design concept requires radial positioning of the in-vessel handling machine to a transfer position external to the core support structure (i.e., the core barrel). In order to minimize the reactor vessel and closure head diameters, the radial position of the transfer position was minimized. The CRBRP vessel diameter was selected to be essentially the same as the FFTF vessel in order to use vessel fabrication capabilities similar to the FFTF design experience. This design constraint on vessel diameter placed a severe requirement on the in-vessel shield design for the larger CRBRP reactor core (i.e., the inner diameter of the core barrel was minimized, minimizing the space available for radial shielding between the reactor core and core barrel).

The design configuration for the radial shield system is depicted in Figure 5. A total of 324 replaceable shield assemblies are included in the design with a 316SS fixed radial shield of 14.6 cm (5.75 inches) internal to the core support structure (core barrel). To accommodate alternative fuel management schemes and core assembly designs, a packed-rod, hexagonal duct replaceable shield assembly design was selected to provide the ability to vary the neutron attenuation characteristics of the replaceable shield. The replaceable shield assembly configuration is a hexagonal duct filled with neutron shield rods. Shield rods extend vertically from the bottom of the lower axial blanket to the approximate top of the upper axial blanket. An exception to this arrangement is at 14 replaceable shield locations in the outer row -- seven at each of two locations -- where additional shield rods are used above the axial blanket, in the vicinity of the core former structure support ring welds. Selection of the shield rod material has been limited to 316SS or Inconel 600.

The axial shield system design provides a 51 cm (20 inch) long fuel, blanket, and control assembly lower axial shield block (316SS) to assure that the design requirements of the core support structure lower inlet modules are met. No upper axial shield material is required since the upper internal structure fluence level is below 1.0×10^{21} n/cm^2, a fluence level at which no measurable change in material properties will occur from that of the unirradiated condition.

Shielding design problems encountered in defining the in-vessel shielding configuration performance were: 1) prediction of neutron streaming in the clearance gaps required in the design of the fixed radial shield, 2) prediction of neutron streaming in the coolant channels of the axial shielding provided in each core assembly, 3) prediction of neutron streaming in the fission gas plenum of each core assembly, 4) definition of mechanical interfaces in in-vessel components to minimize neutron streaming, and 5) arrangement of interface gaps in components to protect longitudinal and circumferential weldments in permanent structures.

Trade-off studies conducted in the early design phases of the in-vessel shielding included the definition of the amount of replaceable shielding versus fixed shielding. Design requirements placed on fixed shielding are less stringent than the core support structure requirements due to the use of a non-welded concept and the FRS not being a load bearing component. The need to minimize the number of replaceable assemblies was further constrained by: 1) the radial positioning limits, which restrict the in-vessel handling machine capability to remove replaceable shield assemblies, 2) the ability of the coolant flow distribution system in the core support structure to provide sufficient assembly coolant flow, and 3) the capability to design an assembly holddown in the upper internals structure. These design constraints were considered in trade-off studies and resulted in a four-row replaceable shield assembly configuration. This arrangement provided sufficient neutron attenuation to meet FRS neutron fluence design requirements.

Neutron flux distributions, used to define fluence levels at in-vessel components, were derived from one and two-dimensional diffusion theory and discrete ordinates transport theory solutions. The design methodology utilized multigroup (40-60 neutron groups) methods and nuclear data developed for the U.S. LMFBR program. Verification of methods and nuclear data has resulted from extensive analysis of shielding experiments conducted at facilities at ORNL. The relevance of the CRBRP design effort and the ORNL shielding experiment program is discussed in later sections. Companion papers at this seminar discuss the design methods and experiments in greater detail.

3. REACTOR ENCLOSURE SYSTEM SHIELDING

The CRBRP reactor enclosure shielding consists of the closure head assembly, reactor vessel support area (RVSA) and support ledge, and the reactor cavity wall. Key shielding design problems in these areas were: 1) the design of component penetrations and component interfaces in the closure head assembly, 2) reactor vessel support area and support ledge shielding, and 3) ex-vessel flux monitor shielding.

The principal characteristics of the RVSA and support ledge of the CRBRP are shown schematically in Figure 6. The reactor cavity, consisting of a 3 meter

(10 foot) annular cavity external to the reactor vessel, is required for the installation and routing of the heat transport system piping of the loop-type reactor and the ancilliary equipment in the cavity (e.g., remote in-service inspection and ex-vessel start-up, wide, and power range flux monitoring equipment). The reactor cavity is bordered on the top by the vessel support ledge and the reactor vessel support system.

A radiation zoning specification of <25.0 mrem/hr in the head access area (HAA) is defined as the design limit based on consideration of personnel access requirements and design objectives for plant personnel radiation exposure. From FFTF shielding design experience, the RVSA streaming problem in the CRBRP was recognized at an earlier stage, and the shielding requirements were minimized by arrangement of the annular gaps and utilization of the structural mass, such as the vessel flange. Key shielding elements in the RVSA shielding solution include a canned B_4C radiological shield at the lower elevation of the support ledge, which interfaces to a close tolerance with the reactor vessel outside diameter, a carbon steel shield collar in the thermal insulation module to reduce the gap at the vessel flange elevation, and a concrete shield ring between the support ledge embedment plate and the reactor vessel flange, to minimize the effects of radiation streaming into the HAA. The selection of a configuration and materials of construction was constrained by the high temperatures at the B_4C and reactor vessel interface and by the requirement to provide interface gaps which allow sufficient support ledge coolant flow.

Two dimensional discrete ordinates transport techniques were used to define the radiation streaming in the RVSA. Analyses have been performed by utilizing multigroup cross section sets of 40-60 neutron groups and 15-25 gamma groups. Forward biased quadrature sets containing 100-166 angles were utilized in the analyses to represent neutron streaming through the gaps between the reactor vessel and support ledge.

Closure Head Assembly (CHA) Shielding

As shown in Figure 7, the closure head assembly is a large diameter (6.1 meters, 20 feet) carbon steel assembly with multiple penetrations for head mounted components. The CHA is assembled from an eccentric triple rotating plug configuration with penetrations for components. The principal types of penetrations are: 1) refueling components, 2) control rod drive mechanisms, and 3) upper internals jacking mechanisms.

Radiation source terms considered in the CHA shielding design included: 1) the neutron and gamma streaming up the stepped annuli of the CHA penetrations or component interfaces, 2) the presence of radioactive cover gas below and in the CHA penetrations, and 3) neutron and gamma penetration through the CHA bulk shielding. The sodium filled dip seals, which form the sealing barrier for reactor cover gas in the CHA rotating plug annuli, were recognized as the major shield design problem. The design requirement on the operation of CRBRP with failed fuel is continuing operation at a failure level of the fuel rods producing 1% of the power. The radioactive fission product gases result in a radiation source requiring $\sim$0.3 meters of steel shielding to reduce radiation levels to levels consistent with the radiation zoning of the head access area (HAA). Dip seal arrangement trade-off studies resulted in the only acceptable design solution, a seal located in the closure head, as shown in Figure 8. This location of the dip seal results in radiation levels in the HAA, which are the principal contributor to HAA dose rate levels.

Component penetrations and component interface design solutions in the CHA were achieved by using strict control on interface gap sizes, using offset gaps, using the mass of the component as shielding, and by controlling cover gas penetration of annuli with down purges of recycled cover gas. Control of cover gas leakage through component penetration seals was considered in detail, and the contribution of leaked radioactive gases to the radiation level in the HAA and Operating Floor (OF) was included in radiation exposure estimates.

Design of the CHA also considered the upper axial biological shield system of the CRBRP. In order to use FFTF design experience, the CHA dimensions were selected at an early stage in the design. Early design concepts included a series of stainless steel thermal shields and Inconel or stainless steel radiological shield plates attached to the lower surface of the closure head. The closure head is of similar dimensions as the FFTF closure head (0.56 meters, 22 inches) and is a carbon steel material. Design analyses of the upper axial shield system for CRBRP resulted in the substitution of carbon steel radiological shield plates for the Inconel or stainless steel radiological shields. This design change resulted in a cost reduction due to use of a lower priced material and reduced fabrication costs. Due to the uncertainties in the neutron transport through large thicknesses of sodium and carbon steel, an upper axial shield experiment was designed and conducted at facilities of the ORNL. This experiment provided design assurance of the performance of the CRBRP upper axial shield system.

Ex-Vessel Flux Monitor System Shielding

The ex-vessel source range flux monitor (SRFM) system for monitoring the CRBRP reactor core during shutdown and refueling conditions is physically located in the reactor cavity external to the guard vessel. Design requirements imposed by the monitor system are: 1) the foreground neutron flux for the reactor core must be great enough to monitor changes in reactor core subcriticality, 2) neutron flux background due to extraneous sources (e.g., fuel assemblies in transfer within the reactor vessel) must be less than ten (10) percent of the foreground, and 3) gamma dose rates due to primary sodium coolant and structure activation must be less than 100 Rads/hour. The need for a strong foreground neutron count rate with the reactor core in a fully shutdown state required for optimization of the radial shielding of the reactor core to meet these requirements while meeting in-vessel shielding requirements on neutron fluence at components.

The shielding design at the source range flux monitor consists of a graphite moderator block of 51 cm (20 inches) by 63 cm (25 inches) surrounded by lead and B_4C background shields. The moderator block B_4C shield arrangement for reducing the neutron background due to fuel-in-transfer or the storage of fuel assemblies in the fuel transfer and storage assembly, is supplemented by B_4C reactor cavity shields. The gamma background at the SRFM is reduced to acceptable levels by surrounding the moderator block with lead to reduce reactor vessel, guard vessel, and vessel sodium coolant gamma levels, and by the utilization of a high purity aluminum alloy as the structural material for the SRFM to minimize its neutron activation gamma background.

4. BALANCE-OF-PLANT SHIELDING

Design of radiation shielding for the heat transport systems and auxiliary sodium systems is primarily provided by the arrangement of the ordinary concrete structural walls of the reactor containment building and by careful design of cell wall penetrations.

Shielding design problems requiring considerable design effort were: 1) heat transport system pipe chaseway shielding to meet requirements on intermediate heat transport system sodium coolant activation, and 2) auxiliary system piping and ducting penetrations through plant cell walls to meet cell radiation zoning requirements.

Heat Transport System Shielding

One of the most difficult problems in CRBRP shielding design analysis is that of determining radiation streaming through heat transport system pipe chaseways. Design requirements imposed on the CRBRP piping arrangement and neutron shielding configuration were that the neutron attenuation afforded by the pipe routing and neutron shielding must maintain the intermediate sodium activation below 60 pCi/cm^3 and reduce streaming neutron flux to a minimal level before reaching the delayed neutron monitors (DNM's) in the heat transport cells. The design solution selected is illustrated in Figure 9. Neutron shield walls were designed to assure the desired attenuation factor of $\sim 10^8$.

The HTS piping penetration problem of neutron streaming has been analyzed with a series of multigroup two-dimensional discrete ordinates transport solutions with various methods of coupling at each 90° bend. Coupling techniques include: 1) isotropic leakage source from the surface of one cylinder to a disk source entering the second leg, or 2) radial boundary source from the first leg transformed to an axial boundary source for the second leg. The development and use of a Monte Carlo transport method using albedo scatter data is planned to verify the design configuration. Due to the complexity in the design solution, a prototypic experiment for the HTS pipe chaseway was conducted in facilities at ORNL. This experiment was a full scale mock-up using piping mock-ups in concrete penetrations and cells. Pipe simulations used sodium carbonate as a sodium coolant simulation. Results of the experiment confirmed the CRBRP design solution.

An associated problem encountered in the CRBRP HTS shielding design is the photoneutron production in the concrete cell wall. Photoneutrons are generated due to the Na^{24} gamma source in the primary coolant pipes interacting with the deuterium in hydrogenous materials. More than 80% of the DNM neutron background is attributed to the concrete wall photoneutrons. Therefore, non-hydrogenous materials were specified as the DNM neutron background shielding material.

Auxiliary System Shielding

Design problems in radiation streaming were encountered in the heating and ventilation ducting design for system cells and in piping penetrations in auxiliary system cells. Acceptable design configurations have been obtained for piping or duct penetrations up to 76 cm (30 inches) in diameter routed through cell walls. Principal design problems with piping penetrations was the high temperature of the sodium coolant and the need for large thicknesses of insulation to protect the concrete walls.

Analyses of auxiliary piping and ducting penetrations involved either Monte Carlo radiation transport or single scatter point kernel techniques.

5. SHIELDING EXPERIMENT PROGRAM

During the design phase of CRBRP, specific design solutions were identified which required experimental verification of the design methodology, nuclear data, or selected shielding design solution. In these specific cases, experiment designs and experiment plans were defined in a cooperative effort with the reactor designer and the ORNL. Experiments were designed to be prototypic or simulations of CRBRP configuration and radiation environment. Experimental measurements and configuration parameters (e.g., material selection, material thicknesses) were scoped to assure sufficient data for verification of the method, data, or configuration dimensions. In addition to the prototypic experiments for CRBRP, a series of benchmark type experiments was defined and conducted at ORNL to provide data for verification of methods and/or nuclear data. The relevance of the prototypic and benchmark experiments to the CRBRP design is summarized in Tables 1 and 2. A companion paper defining the LMFBR experimental program in greater detail is included in this seminar.

6. SYSTEMS APPROACH TO STAFF RADIATION PROTECTION

An increasing number of USA nuclear facilities have developed systematic approaches to limiting radiation exposure of their staff of professional radiation workers. The CRBRP radiation protection program was incorporated into the overall plant design description in the early design stages. The programmatic goal for CRBRP is to limit the radiation exposure of the operating staff so that the size of the staff is established by the work requirements and not radiation exposure limits. The design requirements necessary to achieve this goal are sufficient to meet the overriding USA regulatory requirement that radiation exposures be ALARA.

The principal elements of the CRBRP radiation protection design approach are summarized below:

- An annual radiation exposure (man-rem/year) for the entire professional radiation worker staff was estimated. This estimate was based on a pragmatic appraisal of radiation exposures of the staff of USA LWR reactor plants which were operating within the CRBRP plant availability requirements and the anticipated size of the CRBRP staff.

- The allocation of this total radiation exposure by function and system within CRBRP was based on both USA Light Water Reactor (LWR) and fast breeder reactor experience.

- The radiation zoning (i.e., radiation dose rate in accessible cells) of the CRBRP nuclear island was established to limit the staff radiation exposures in normally accessible cells to approximately 15% of the total staff exposure. The equipment arrangement, plant layout, and equipment access requirements were considered in defining plant zoning requirements.

- Radiation source terms were developed for the purpose of shield design on an "upper limit" basis. These source terms include margin to account for design and construction uncertainties and were used for the design of the plant shielding. An analogous set of "best estimate" source terms was developed for estimating staff radiation exposure under both operating and maintenance conditions. The best estimate source terms incorporate sodium reactor experience (e.g., BOR-60) in source term modeling wherever possible.

- A cost/benefit formula has been developed for evaluating design changes which impact staff radiation exposure. This formula applies to changes which either increase or decrease radiation exposure. Cost/benefit evaluations are considered in conjunction with overall plant radiation protection requirements to determine the desirability of each proposed design change.

Each CRBRP system is periodically reviewed to determine the status of the radiation protection design development for that system. This review is conducted by a multi-discipline group of project design personnel. This group has expertise in radiation analysis, shielding design, safety, licensing, and plant maintenance. The review team periodically collects plant radiation exposure information and provides the system design with recommendations on areas where the radiation protection aspects of the design can be improved.

In addition, several different systems are reviewed on a semi-annual basis by an independent team of Health Physicists with extensive experience in all aspects of radiation protection at operating nuclear plants. The Health Physics team reviews system and component design, maintenance procedures, and radiation exposure data. Recommendations to further reduce radiation exposure based on ALARA experience at operating nuclear power plants are provided by the team.

It can be seen that the CRBRP radiation protection program is an interactive program requiring exchanges of information between project system designers and both project and outside radiation protection specialists. The scope and extent of this interchange are further illustrated in Figure 10.

The radiation protection program will be completed only after each system designer has appropriately documented the fact that the system meets the radiation protection requirements of the plant and is ALARA. This is demonstrated in part by meeting the annual man-rem allocation requirements. If a system cannot meet these requirements, then a formal change to the requirements must be proposed and justified by the designer. Increased allocation to one system can generally be compensated for by reducing the allocation to a system that has been successful in significantly undershooting their allocation. A reserve of radiation exposure for unanticipated (special) maintenance is also available for changes as the design matures.

The overall plant radiation protection will be considered completed when the total annual radiation exposure requirements are met and documented as required by the U.S. Nuclear Regulatory Commission.

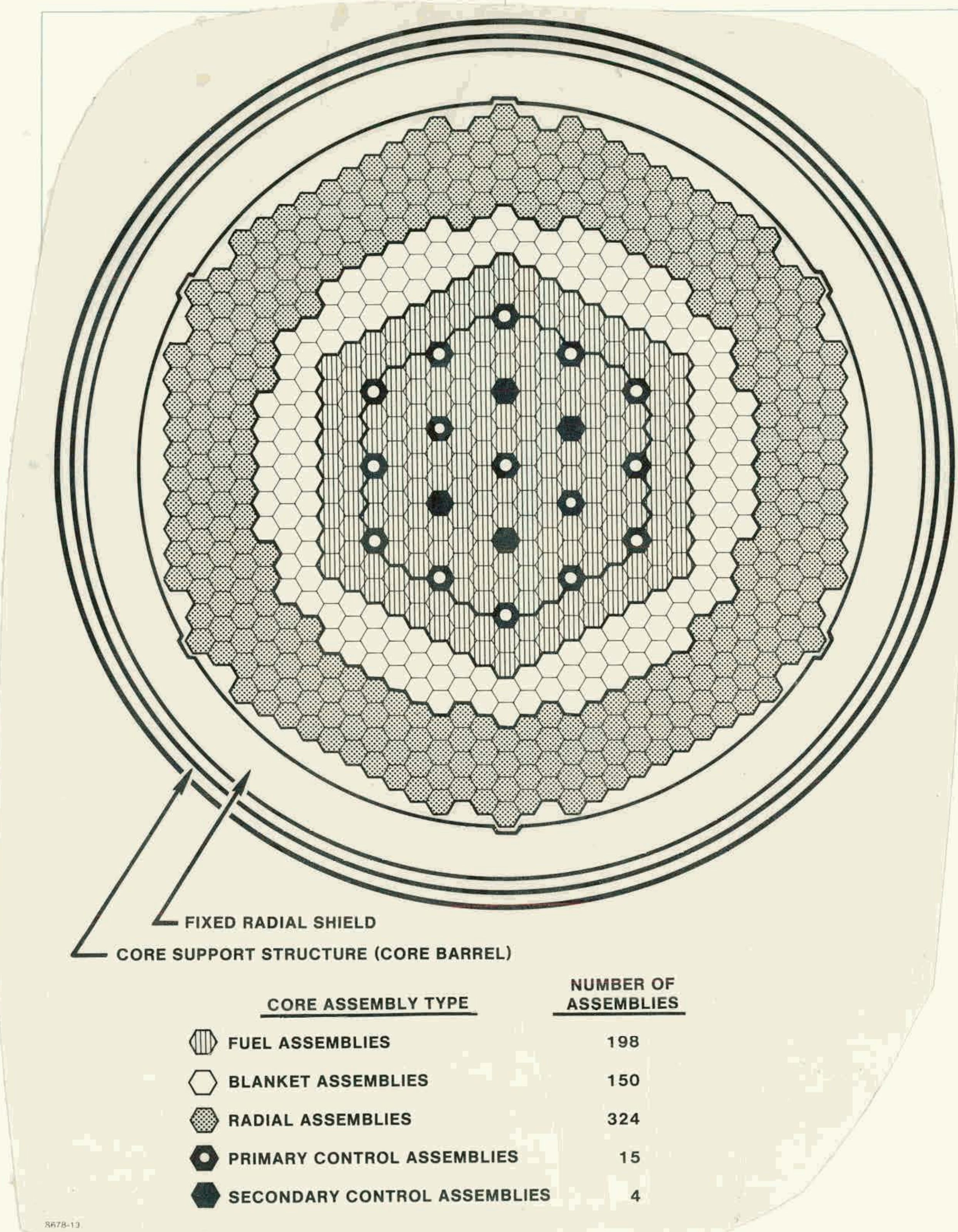

Figure 1. Plan View of CRBRP Reactor Core Layout

F-45812

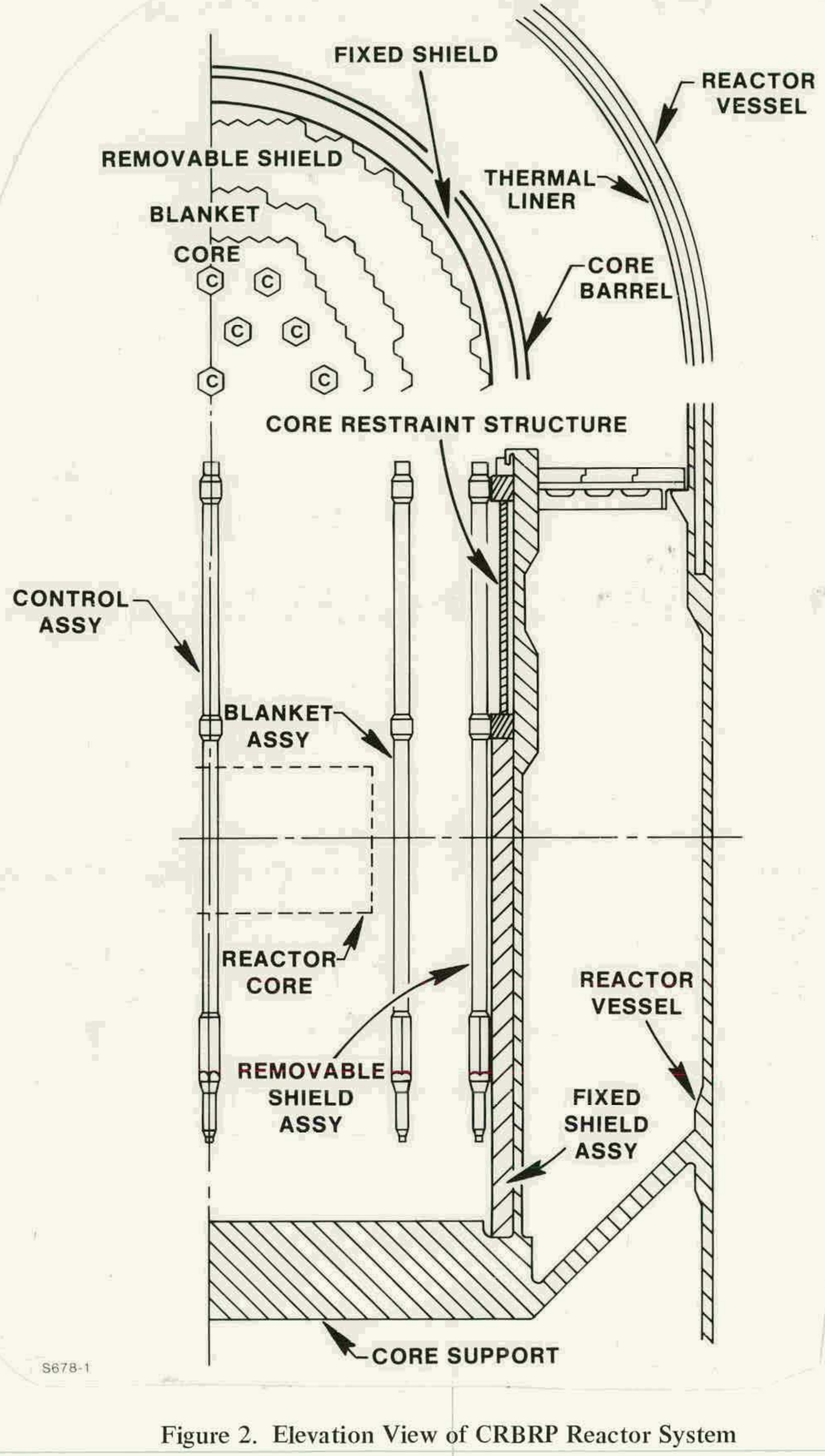

Figure 2. Elevation View of CRBRP Reactor System

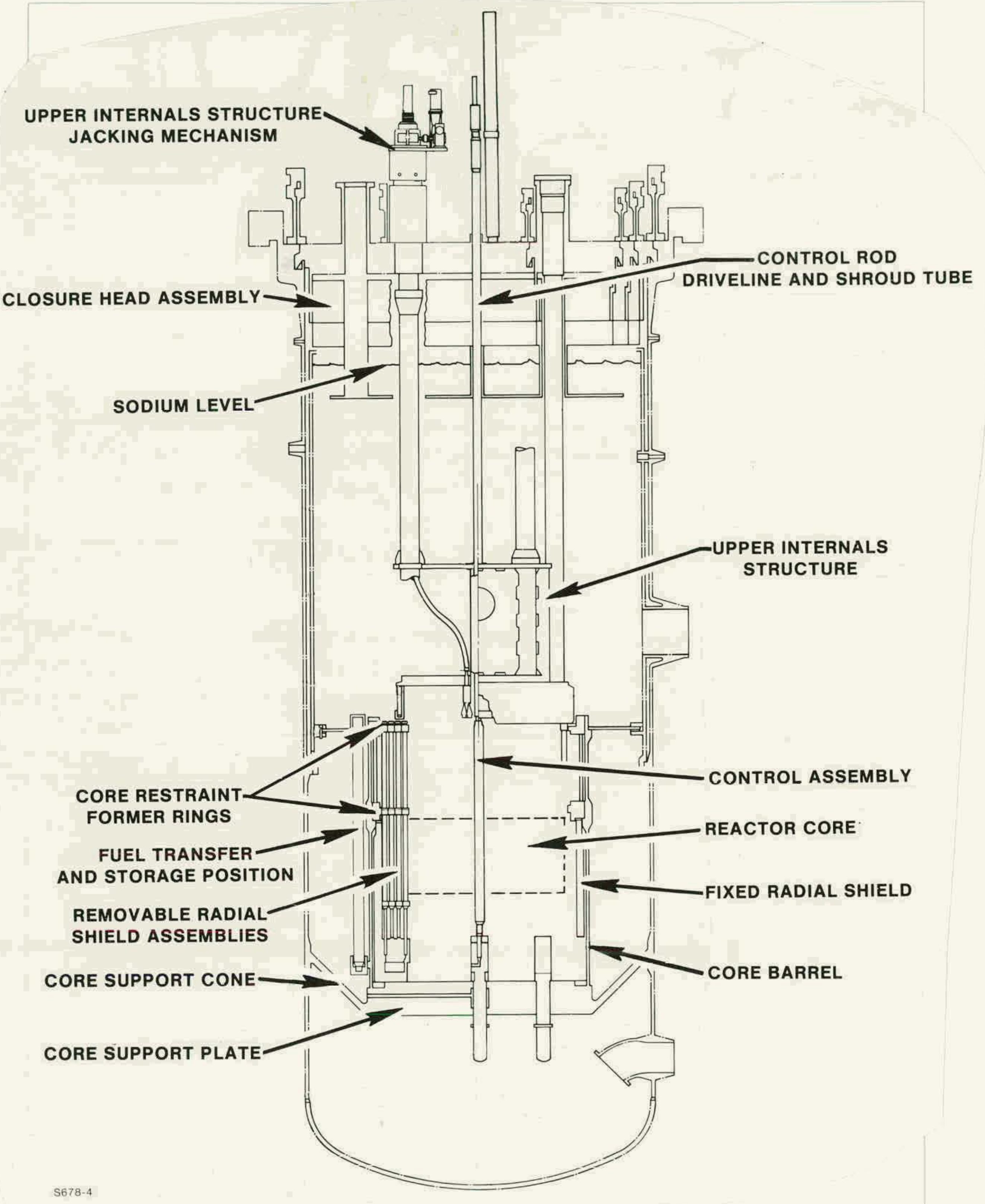

Figure 3. Schematic View of Reactor System

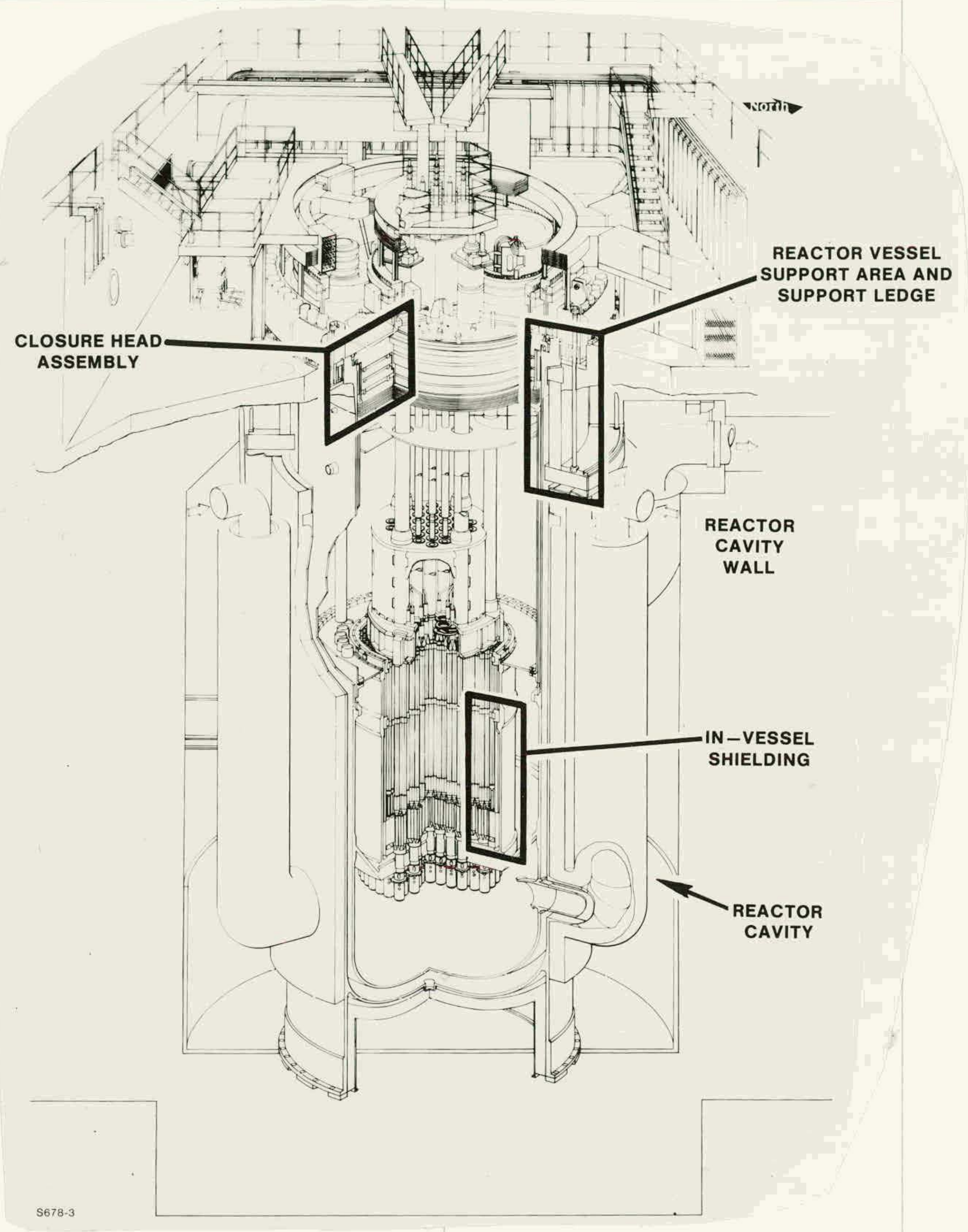

Figure 4. Key Shielding Areas in CRBRP Reactor and Reactor Enclosure System

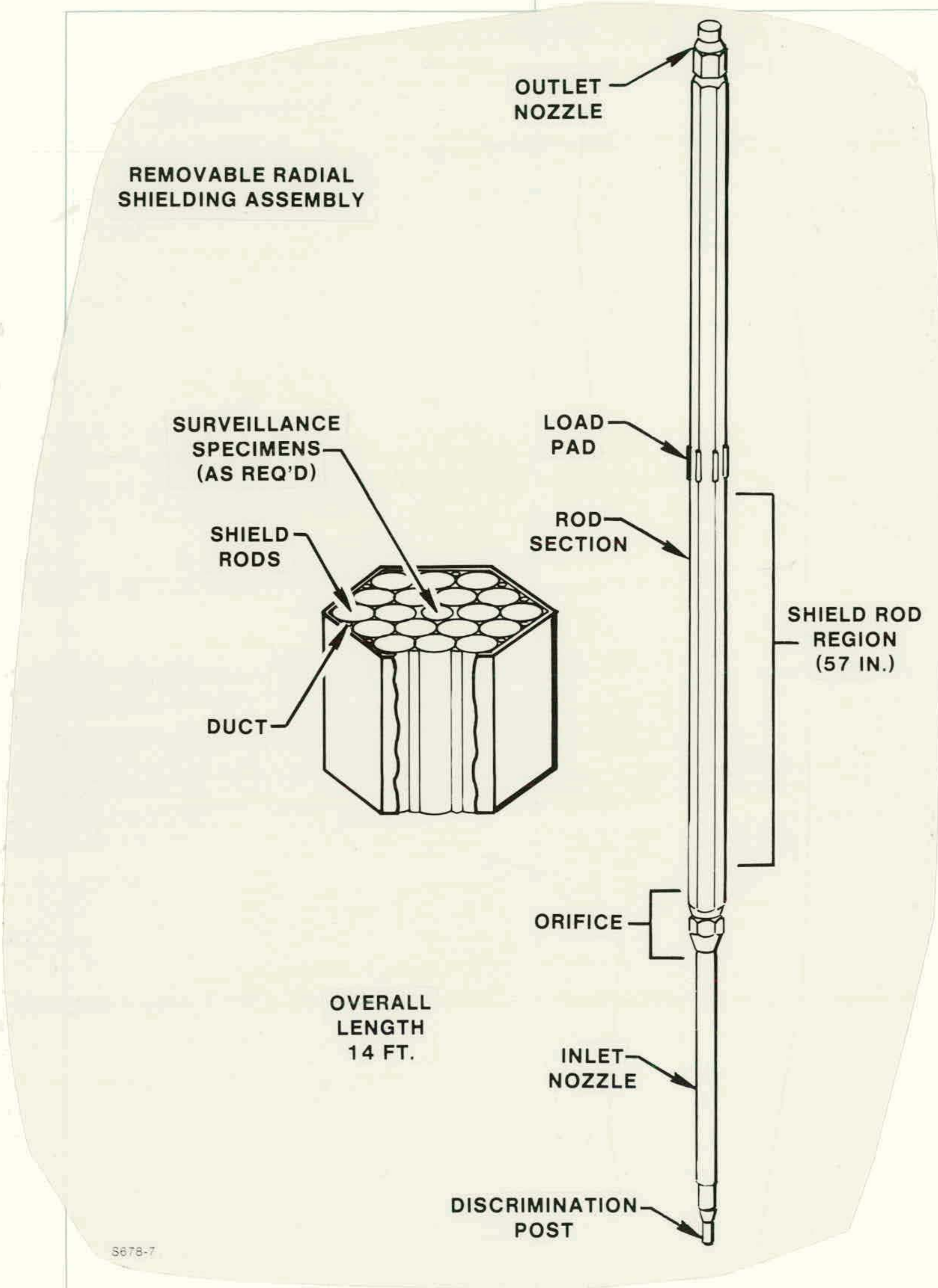

Figure 5. Removable Radial Shielding Components

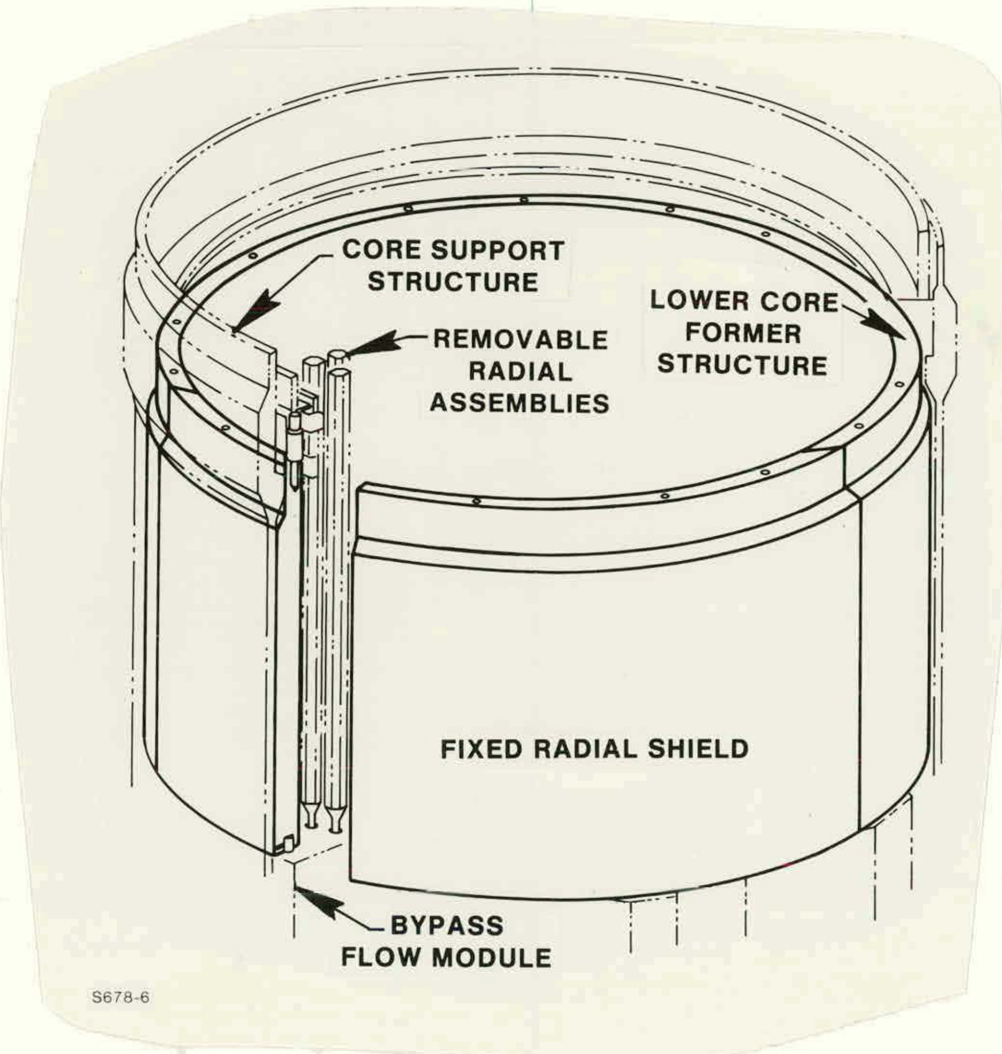

Figure 6. Fixed Radial Shield and Interfacing Components

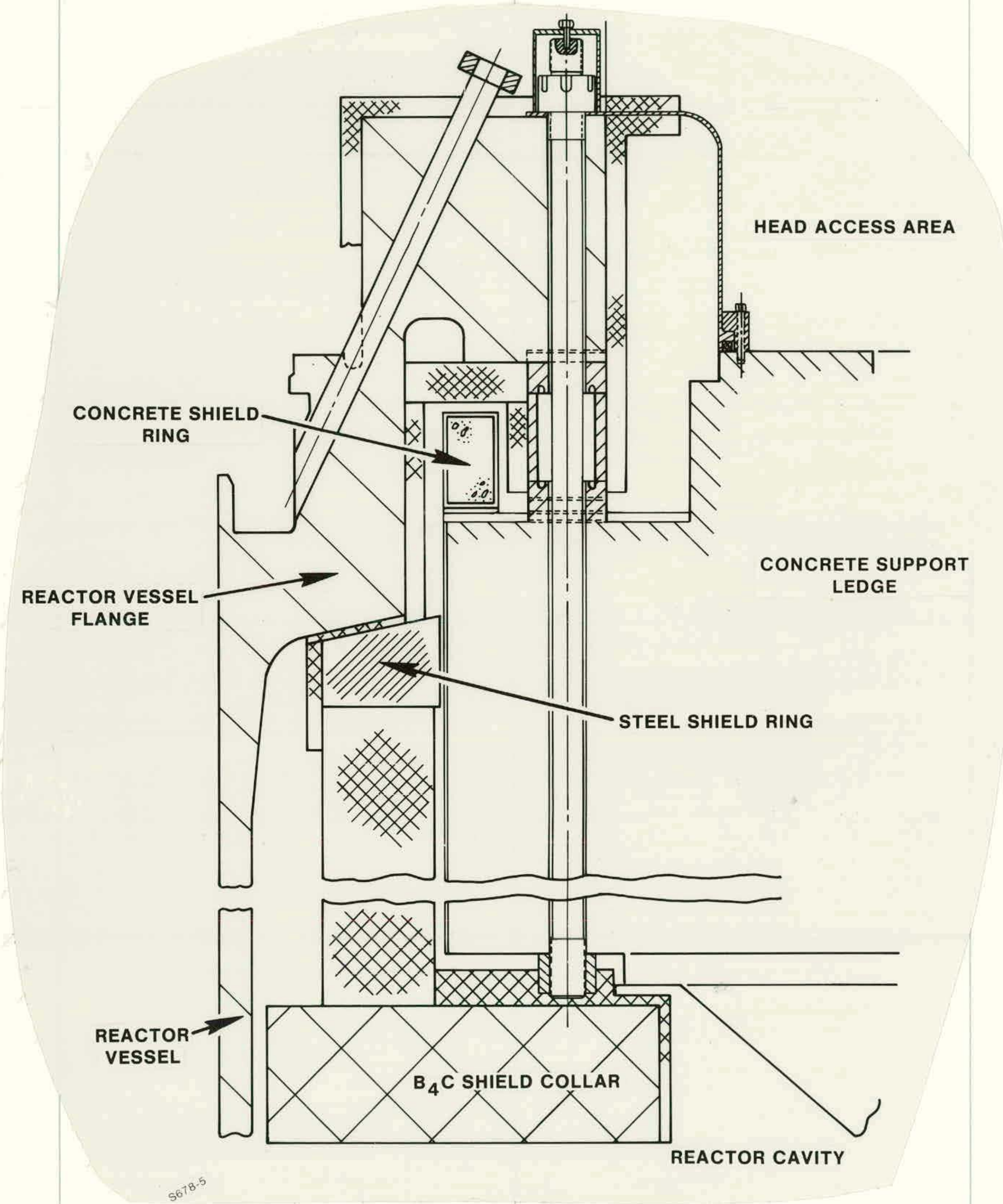

Figure 7. Reactor Vessel Support Area Shielding Arrangement

F-45812

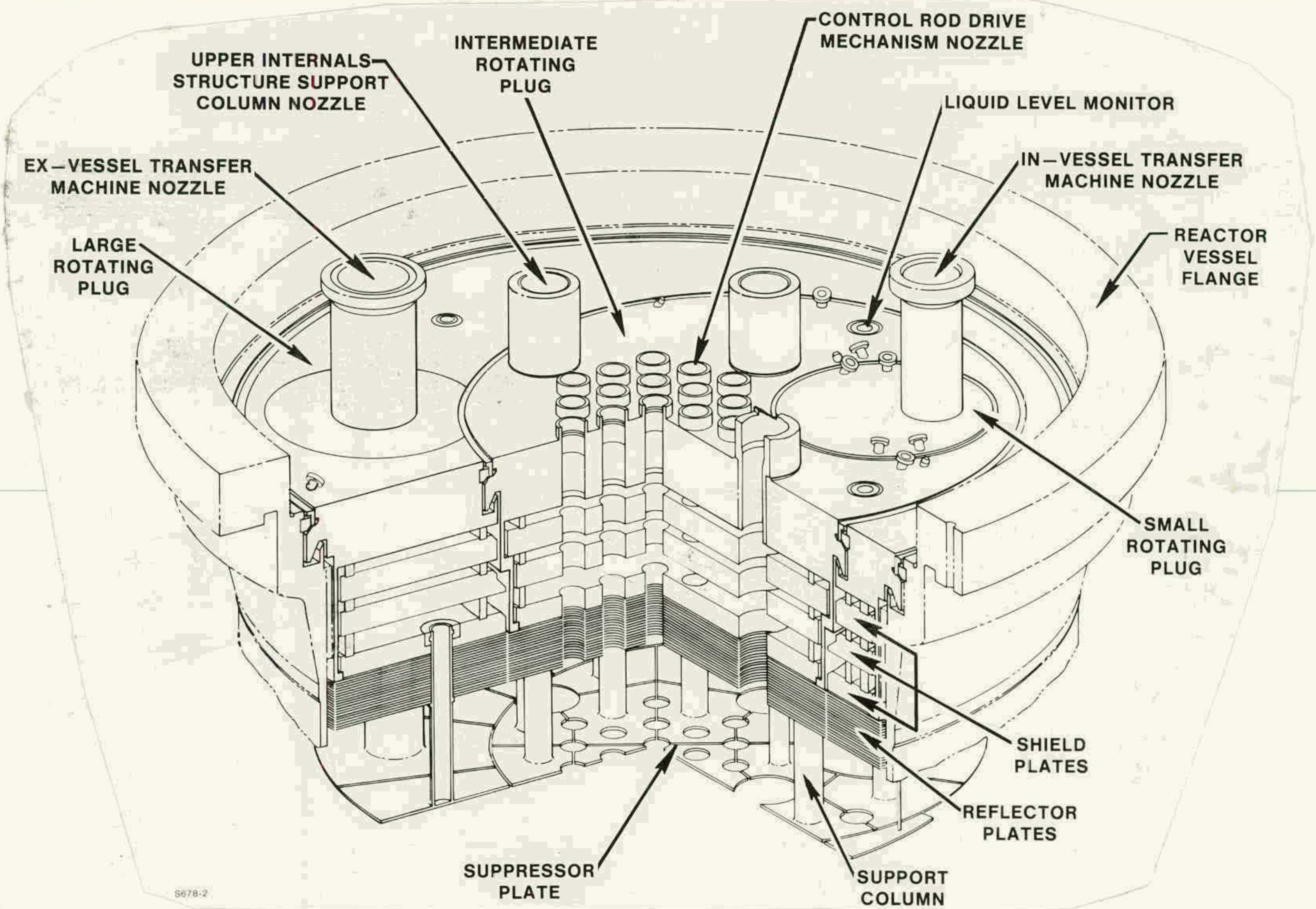

Figure 8. Closure Head Assembly Configuration

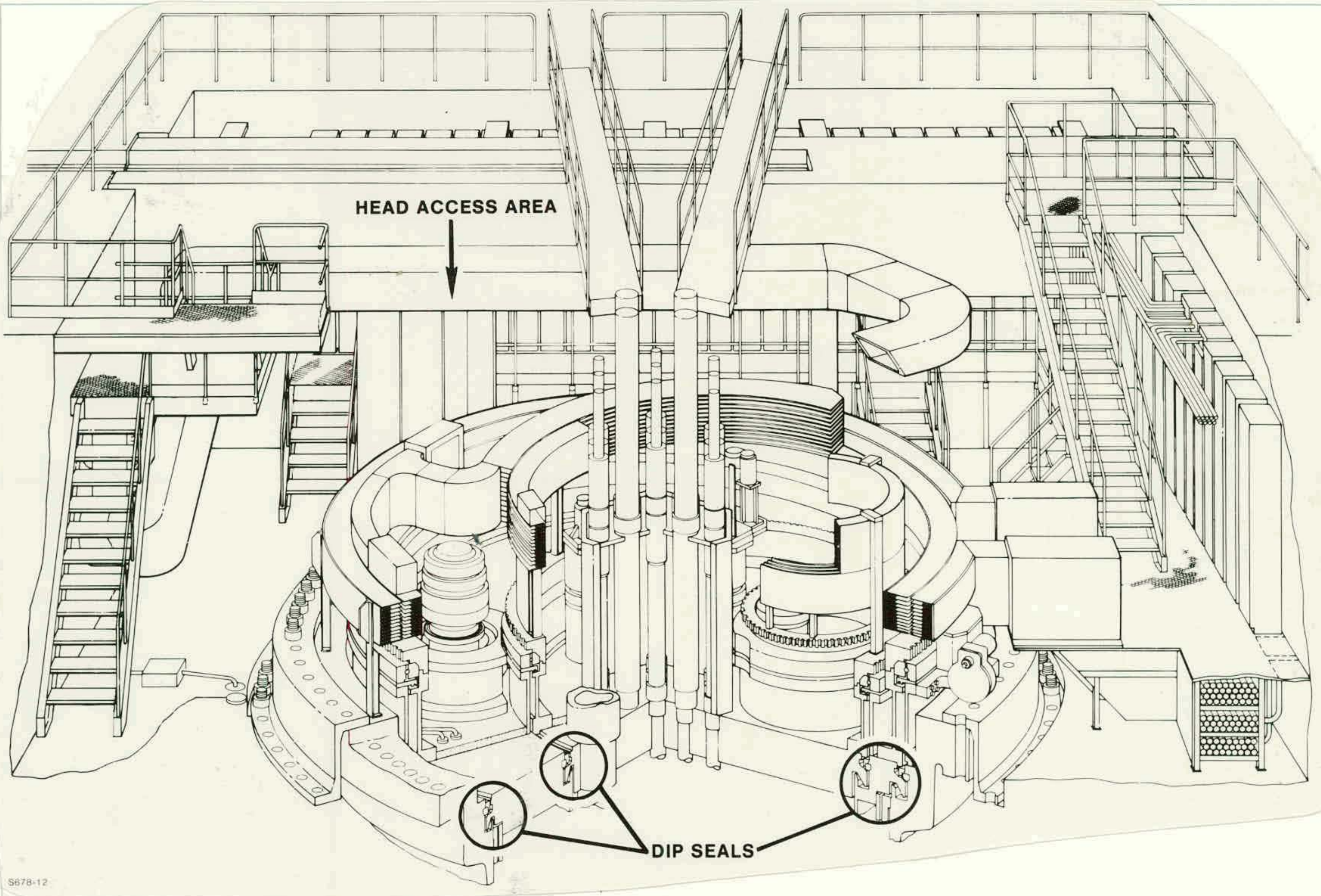

Figure 9. Closure Head Assembly Dip Seal Arrangement and Head Access Area

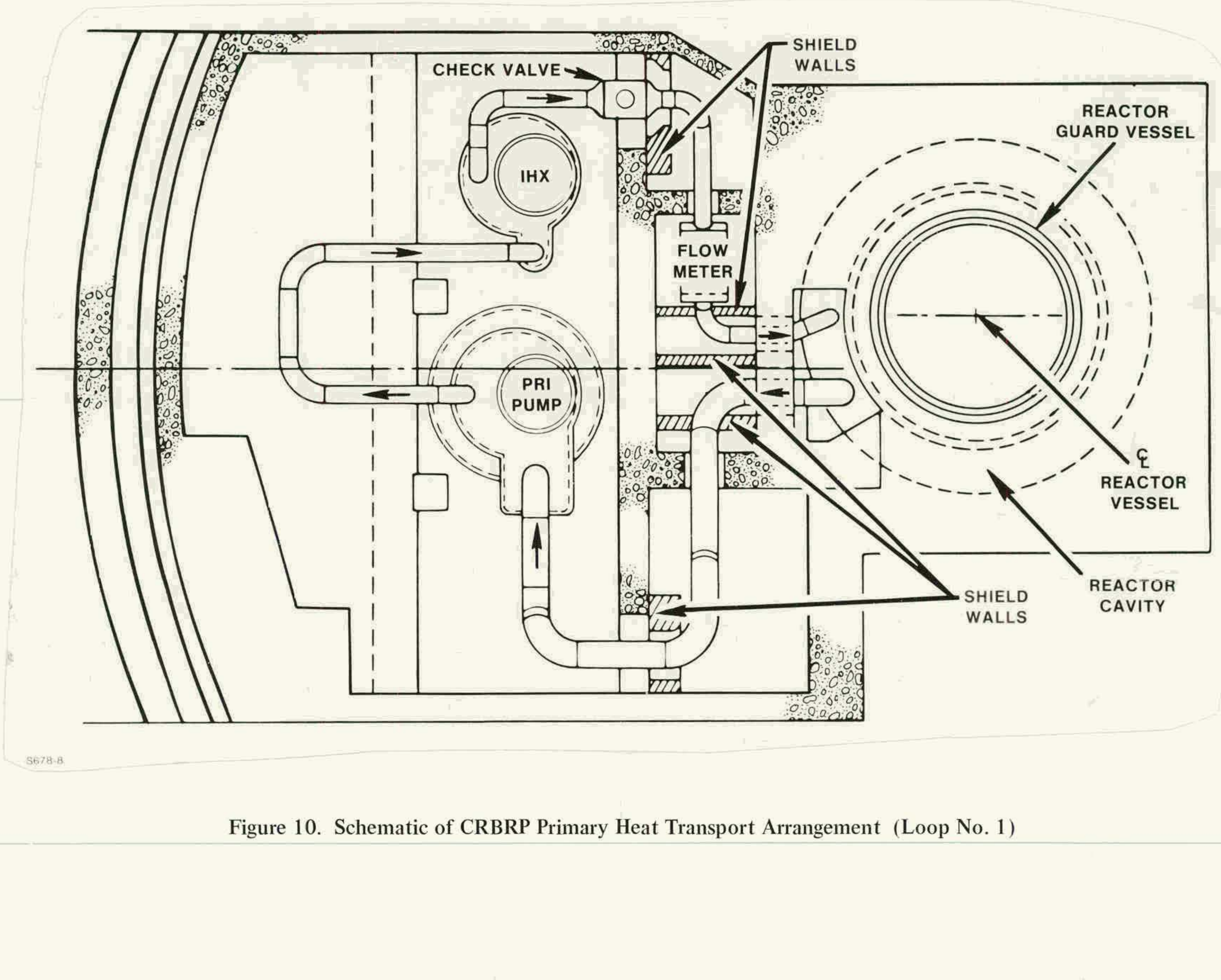

Figure 10. Schematic of CRBRP Primary Heat Transport Arrangement (Loop No. 1)

F-45812

Page No.

Experiment	Reactor System	Configuration	Spectrum Modifier	Purpose
Pipe Chaseway (Phase I/II)	FFTF	Simulation Of Primary Heat Transport Penetrations And Routing	Yes	Neutron Streaming In Piping Penetrations And Chaseway
Radial Shield/Stored Fuel	FFTF	Simulation Of Radial Shield And In-Vessel Stored Fuel	Yes	Neutron Attenuation And Fission Rate In Stored Fuel
Radial Shield/Stored Fuel/Flux Monitor	LMFBR	Simulation Of Radial Shield/Stored Fuel/And Ex-Vessel Flux Monitor	Yes, Radial Blanket Simulation	Neutron Attenuation/Fissions In Stored Fuel/Flux Monitor Response At Ex-Vessel Position
Radial Shield/Upper Axial Shield	LMFBR	Simulation Of In-Vessel Shielding Using Packed Rod Arrays Of Canned B_4C Or Steel	Yes, Radial Blanket Simulation	Neutron Attenuation Of Conceptual Shields
Lower Axial Shield	LMFBR	Simulation Of Lower Axial Shield With Control Rod Penetrations	Yes, Axial Blanket Simulation	Neutron Attenuation And Streaming In Control Rod Channels
Radiation Heating	CRBRP	Simulation Of Radial Shielding With Carbon Steel And/Or SS 304 Slabs — Three Experiments Including Gamma Dosimetry Development With TLD's And Ion Chamber	Yes, Radial Blanket Simulation	Neutron Attenuation And Gamma Energy Deposition In Shields
Pipe Chaseway	CRBRP	Simulation Of Heat Transport Piping And Chaseway With Two 90° Bends And 36" Pipe Simulation	Yes, Cavity Neutron Specturm	Neutron Attenuation In Design Configuration
Concrete Rebar	CRBRP	Simulation Of Reinforcing Steel In Concrete Wall	Yes, Cavity Neutron Spectrum	Neutron Attenuation And Secondary Gamma Attenuation With And Without Reinforcing Steel Bars
Upper Axial Shield	CRBRP	Simulation Of Upper Axial Shield Configuration With Up To 15' Of Na And Up To 30' Of Carbon Steel	Yes, 18" Of SS 304	Neutron Attenuation Of Laminar Arrangement Of Sodium Followed By Carbon Steel

S678-10

Table 1. Summary of LMFBR Prototypic Shielding Experiments

F-45812

Page No.

Experiment	Reactor System	Configuration	Spectrum Modifier	Purpose
Annular Slit Streaming	FFTF	Carbon Steel Slabs – 0, 1, or 2 Offset Annular Gaps, 1/4" To 3/4" Gap Dimensions, 44" Thickness	Yes, Na Pool Simulation	Characterize Neutron Streaming In Annular Gaps In Carbon Steel Shields
Carbon Steel	General	Carbon Steel Slabs – 1/2" To 36" Thicknesses, 5' Square	No	Neutron Attenuation
Sodium	General	Na Slabs – 2.5' To 15' Thickness, 11' Diameter	No	Neutron Attenuation
Stainless Steel	General	SS 304 Slabs – 18" Thickness, 5' Square	No	Neutron Attenuation
Inconel	General	Inconel 600-2.5" and 5.1" Thickness, 5' Square	No	Neutron Attenuation And Secondary Gamma Production And Attenuation
Carbon Steel/Borated Polyethylene	FFTF	Carbon Steel/B-CH_2 Slabs-10" To 20"/1" To 4", 5' Square	Yes, Na Pool Simulation	Neutron Attenuation And Secondary Gamma Production And Attenuation
Stainless Steel	CRBRP	SS 304 Slabs – 2.5" To 15" Thickness, 5' Square	Yes, Radial Blanket Simulation	Neutron Attenuation For Direct Comparison To Inconel
Inconel	CRBRP	Inconel 600 Slabs – 2.5" To 15" Thickness, 5' Square	Yes, Radial Blanket Simulation	Neutron Attenuation For Direct Comparison To SS 304

S678-11

Table 2. Summary of LMFBR Shielding Benchmark Experiments

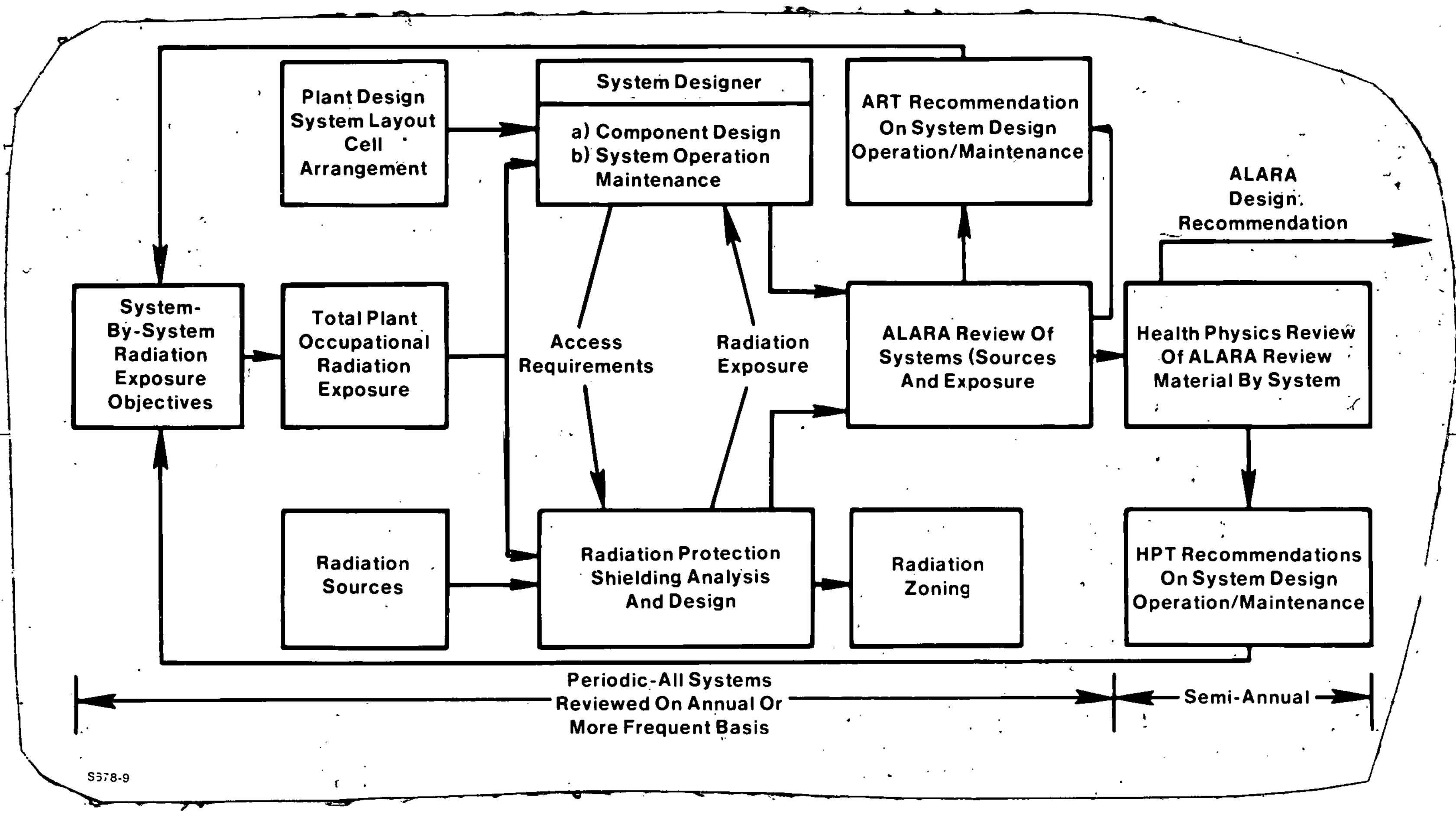

Figure 11. CRBRP Radiation Protection/Shielding Design Approach to Achieve ALARA Objectives