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HTGR GENERIC TECHNOLOGY PROGRAM SAFETY, SYSTEMS AND COMPONENT DESIGN AND DEVELOPMENT

**QUARTERLY PROGRESS REPORT
FOR THE PERIOD ENDING
SEPTEMBER 30, 1978**

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HTGR GENERIC TECHNOLOGY PROGRAM QUARTERLY REPORT SERIES

GA-A14922, for the period ending March 31, 1978
GA-A15055, for the period ending June 30, 1978

ABSTRACT

This publication continues the two quarterly report series on HTGR Component and Systems Development and HTGR Safety. The tasks previously covered (safety and reliability, structural materials, thermal barrier, and reactor internals) are now part of the HTGR Generic Technology Program, PA 65, which also includes heat exchanger engineering and mechanical and electrical engineering reported here. The structural materials work is now reported in a separate quarterly.

The primary purpose of the HTGR Generic Technology Program is to develop, in a time frame consistent with application to specific HTGR plants, the HTGR technology base and design definition of nuclear components and systems which are common to HTGR reactor types. The program areas included under the HTGR Generic Technology Program are listed below. The work areas reported in this quarterly are identified by an asterisk.

189 No.	Program Area	Work Area Title
00790	1	Safety and Reliability*
N/A	2	Systems Engineering
00791	3	Structural Engineering*
00793	4	Heat Exchanger Equipment*
00799	5	Mechanical Engineering*
N/A	(Incl. in 5)	Electrical Engineering*
00792	7	Materials Technology
00555	8	Reactor Operating Experience (PBEOL and FSV Surveillance)
00974	9	Reactor Core Engineering
00551	10	Fuel Development and Engineering
00552	11	Graphite Development
00549	12	Fission Product and Coolant Chemistry

189 No.	Program Area	Work Area Title
N/A	13	Fresh Fuel Process Development
N/A	14	Program Management Support
N/A	(Incl. in 14)	CEA Program Support
00760	16	Medium-Enriched Fuel Development

The work documented here includes HTGR safety and safety-related studies to better understand and more accurately define safety characteristics and safety margins of GCRs and under postulated accident conditions. Also included are the design, analysis, and testing of the PCRV, liner, penetrations, thermal barrier, reactor internals, steam generator, core auxiliary heat exchanger, and rotating machinery. Design studies and analyses plus experimental procedures and results are discussed and, where appropriate, the data are presented in tables, graphs, and photographs. More detailed descriptions of experimental work are presented in topical reports.

INTRODUCTION

This report covers the work performed by General Atomic Company under four program areas of the High-Temperature Gas-Cooled Reactor (HTGR) Generic Technology Program sponsored by the Department of Energy under Contract EY-76-C-03-0167, Project Agreement No. 65. This Project Agreement calls for development of the HTGR technology base and design definition of nuclear components and systems which are common to HTGR reactor types.

Characteristics of large HTGR designs include:

1. A single-phase gas coolant allowing generation of high-temperature, high-pressure steam with consequent high-efficiency energy conversion and low thermal discharge.
2. A prestressed concrete reactor vessel (PCRV) offering advantages in field construction, primary system integrity, and stressed member inspectability.
3. Graphite core material assuring high-temperature structural strength, large temperature safety margins, and good neutron economy.
4. Thorium fuel cycle (with high or medium enriched uranium) which allows good utilization of nuclear resources and minimum demands on separative work.

These basic features are incorporated into the 330-MW(e) prototype Fort St. Vrain reactor which is currently undergoing startup testing.

Work under the program areas of HTGR safety and reliability, structural engineering, heat exchanger engineering, and mechanical and electrical engineering is included in this report.

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1. SAFETY AND RELIABILITY (DV&S)
189a NO. 00790; PROGRAM AREA 1

1.1. SUMMARY

The main objectives of this work area are to ensure that HTGR generic design features meet applicable safety and licensing criteria and to develop safety reliability and availability data for HTGR components and systems. In FY-78 generic activities have mainly focused on Safety Analysis, Accident Initiation and Progression Analysis (AIPA), Safety Research, Safety and Reliability Computer Code Development, Nuclear Steam Supply (NSS) Systems Analysis, and NSS Systems Analysis Methods Development.

Outlines of Licensing Topical Reports (LTRs) for several safety-related topics, which will be submitted to the NRC for the purpose of initiating preapplication licensing reviews, were produced this quarter. These topics include interpretation of 10CFR100 and development of General Design Criteria, positions on Regulatory Guides, application of AIPA methodology in Chapter 15 of the PSSAR, and the development of a maximum hypothetical fission product release (MHFPR) model.

The AIPA status report on Phase II risk assessment studies was issued. A risk assessment of accident sequences initiated by major fires was also completed, showing that the probability per reactor year of a fire leading to core heatup is about 1×10^{-5} .

Programming was completed for incorporating a new fission product release model for failed fuel particles into the SORS computer code. Comparison with temperature ramp experiments shows excellent agreement.

The FY-78 static tests of iodine plateout on mild steel were completed. Checkout of test equipment and procedures for dynamic helium flow tests of iodine and cesium plateout is in progress.

Further development of the RATSAM computer program to evaluate long-term primary coolant flows and temperatures during core heatup was completed. This enables RATSAM to provide PCRV input conditions to the plateout code PADLOC for determination of condensable fission product fractions released to the containment during a longer time period of the core heatup event.

The computer program TRAIN for calculating transient pressures and temperatures associated with primary coolant system depressurization was combined with the containment atmosphere response code CNTB. Incorporating TRAIN, which is verified with FSV reactor test data, makes calculations of PCRV depressurization through a multicomponent system into the containment possible.

The SPLAT computer code for analyzing a hypothetical design basis depressurization accident (DBDA) with a steam generator leak was verified by comparison with FLASH code analysis.

The development of methods to predict suppression of natural convection in vertical ducts [such as the core auxiliary heat exchanger (CAHE) cavity] continued. Appropriate differential and algebraic equations for the suppression of natural convection in a vertical slot were formulated and are currently being verified.

Several further developments of the turbulent flow code MINT were completed. These included the capability to study natural and mixed convection, calculation of flow patterns in a plenum next to a tube bundle by including the effects of tube bundle drag, and a flow assembler code for MINT solutions of simple geometries which will allow more complicated geometries to be analyzed.

1.2. DESIGN APPLICATIONS AND LICENSING SUPPORT

The objectives of this task are to provide design support for generic systems and components in the area of safety analysis. By and large such analysis is employed in resolving current licensing issues. Furthermore, siting and licensing criteria and positions on Regulatory Guides for HTGRs are developed under this task.

Effort this quarter was associated with the development of an LTR that addresses the application of the General Design Criteria of Appendix A to 10CFR50 and the Siting Criteria of 10CFR100 to HTGRs. The milestone for the submission of a skeletal draft report (an outline with a brief descriptive narration) was met. The outline material is being readied for submittal to the NRC for their review.

Preparation of the LTR itself is under way. The General Design Criteria in Appendix A to 10CFR50 have been reviewed to determine which are general enough to be directly applicable to HTGRs, which must be modified for proper applicability to HTGRs, and what additions are needed to address characteristics that are unique to HTGRs. Writing of the text was initiated.

1.3. AIPA METHODS DEVELOPMENT

The primary objectives of the AIPA study are to provide guidance for safety research and development programs for HTGRs by means of probabilistic risk assessment (PRA) and to obtain risk information necessary to license the HTGR on its own merits. In FY-78 the main work efforts included preparation and issue of a status report on Phase II of the HTGR risk assessment study (Ref. 1-1) and safety studies on the effects of major plant fires.

The Phase II status report (Ref. 1-1) has been completed, patent cleared, and issued. A comprehensive assessment of public risks due to HTGR accidents is presented in the report, as well as an update of the

extensive peer review carried out by both U.S. and European safety experts on earlier volumes in the AIPA status report series. The Phase II results confirm the high degree of inherent safety of HTGRs indicated by the earlier results.

A risk assessment of accident sequences initiated by major plant fires was completed and quantitative results were obtained. The study was carried out in four major phases including:

1. Data collection and interpretation.
2. Identification of important fire locations.
3. Event tree/fault tree analysis.
4. Consequence and risk assessment.

The risk assessment methodology developed previously by the AIPA study team was refined in several ways in consideration of the unique characteristics of major fires. New methods were developed for screening the plant layout for the identification of potentially important locations for major fires and for estimating the probability that specific sets of plant systems would be damaged by the fire. The initiating event selected for event tree analysis was the occurrence of a fire of any size in the cable spreading area immediately below the control room. The probability per reactor year of such a fire leading to core heatup was found to be about 1×10^{-5} , which is about 30% of that assessed for all other initiating events in Ref. 1-1.

1.4. HTGR SAFETY RESEARCH

The objective of the HTGR Safety Research Task is to examine and improve models used to describe HTGR behavior during accident conditions. The current areas of R&D were determined by the AIPA studies. These areas

include (1) core material redistribution, (2) PCRV plateout, and (3) containment atmosphere response.

1.4.1. Core Material Redistribution

The report on the tests and analysis of fission product release from failed fuel particles (Ref. 1-2) has completed GA internal review and will be published shortly, satisfying a major task milestone.

1.4.1.1. SORS Code Modification

The fuel particle fission product release model utilized in the SORS code (Ref. 1-3) to predict release from HTGR cores during postulated temperature excursions was modified to incorporate the kernel release model described in Ref. 1-2. The revised kernel release model applies to both fissile (UC_2) and fertile (ThO_2) HTGR fuel particle kernels and is believed to be more accurate since it is based on observations of both isothermal and temperature rise reference fuel particle annealing experiments.

The SORS code is now capable of analyzing the release of multiple elements using the new model. Currently, the release of xenon, iodine (similar to xenon), krypton, and tellurium may be assessed in accordance with the data provided by Ref. 1-2. This capability encompasses all of the nuclides important in licensing siting calculations and the most important nuclides as determined in AIPA Phase I (Ref. 1-4) and Phase II (Ref. 1-1) studies (i.e., Xe-133, I-131, etc.). An optimized numerical solution technique was applied to achieve converged SORS code solutions with minimal computer time.

Figure 1-1 indicates the accuracy of the new kernel release model. The release of Xe-135 calculated by SORS is found to be in excellent agreement with the measured release during the temperature rise experiment. Figure 1-2 shows the change in the calculated Kr-85 core release for the core heatup temperature transient described in the AIPA Phase II report

Fig. 1-1. Comparison of measured Xe-135 release versus calculated release using the new kernel release model (Ref. 1-2) as incorporated into SORS

Fig. 1-2. Comparison of predicted Kr-85 core release during core heatup of a 3000-MW(t) HTGR using the old (Ref. 1-3) and new (Ref. 1-2) release models

(Ref. 1-1) when the new kernel release model is used rather than the old. There is very little change in core release for the fuel failure model assumed. Figure 1-3 indicates the change in Xe-133 and I-131 calculated core release during a postulated core heatup when first the old and then the new kernel release model is assumed. It is seen that iodine is released sooner using the new model than was predicted in Ref. 1-1.

Some additional improvements to the SORS code have been accomplished. The new kernel release model was incorporated into the SORSG version of SORS. Both SORSD nuclides, which are assumed not to be held up in the core graphite, and SORSG nuclides, which are assumed to be held up in the graphite, can be analyzed by the consolidated current version of SORS. This obviates the need for running two codes, SORSD and SORSG, for a particular problem. Hereafter the combined version will be known simply as SORS. Elements for which applicable kernel release data for the new model have not been evaluated are currently simulated as krypton.

1.4.2. PCRV Plateout

1.4.2.1. Plateout Tests

The static tests on iodine behavior on mild steel were completed, demonstrating a rapid rate of formation of iron iodides, predominantly FeI_2 , under evacuated conditions. The tests were conducted over a wide range of hypothetical core heatup accident conditions; the detailed results were discussed in the previous quarterly (Ref. 1-5). The mild steel specimen in these experiments was a 5.08-cm-wide roll of 12.7-cm-long metal foil, providing a geometric area of 129 cm^2 . Under static conditions, it was found that the rate of reaction of iodine with the Fe constituent of mild steel was extremely rapid and was a strong function of iodine partial pressures.

Recently, 12 experiments were repeated with smaller 1.27 cm x 1.27 cm mild steel foil coupons, providing a factor of 40 decrease in the available geometric surface area. Preliminary calculations show no significant

Fig. 1-3. Comparison of predicted core release during core heatup of a 3000-MW(t) HTGR using the old (Ref. 1-3) and new (Ref. 1-2) kernel release models in the SORS computer code

decrease in the iodine reaction rate at specimen temperatures of 600 to 1000 K. The tests at 1100 K show a factor of 4 to 5 decrease in the reaction rate.

Equipment to conduct dynamic tests of iodine in flowing helium has been assembled. Preliminary debugging of the various operational parameters is in progress. The experiments with iodine in flowing dry helium will begin in the near future. Later, primary coolant impurities will be added to the helium to study the effect on plateout. Similar tests will be conducted for cesium. Preparation of a test plan has been initiated.

1.4.2.2. RATSAM Analysis

Further development of the RATSAM computer code to evaluate long-term primary coolant flows and temperatures during core heatup was completed. This will enable RATSAM to provide PCRV input conditions to the plateout computer code, PADLOC, for determination of the fraction of condensable fission products released to the containment throughout the meaningful time period of the core heatup event.

The implicit integration technique in the RATSAM code has been modified to incorporate a predictor-corrector method which iterates to a converged solution. A means of further decreasing the code running time was achieved by developing a capability to transfer to a simplified geometric model from a larger more detailed model at any restart point in the simulation. This feature allows the use of a 24-node detailed model during the early hours of the transient when the PCRV is pressurized, natural convection is the dominant heat transfer mechanism, and plateout is not important. The transfer to the simpler 11-node model can be made when the PCRV is depressurized to enable faster running of the problem without requiring reanalysis of the problem with the smaller model. The less detailed model is comparable to the PADLOC model and should be able to predict flows to the accuracy justified by the current state of plateout knowledge, particularly since plateout is a function of laminar flow to

the 1/3 power. Use of the predictor-corrector solution technique with the less detailed model will enhance the solution accuracy and running time.

As a result of the FY-78 effort, several areas of RATSAM modeling were identified which would enhance the plateout analysis. These include physical models for radiation heat transfer, redistribution of decay heat sources, and failure of liner cooling and thermal barrier. These are recommended for future safety research.

1.4.3. Containment Atmosphere Response (CAR) Program

Work during the last quarter was performed in the area of small jets (Ref. 1-6), concerning their global effects on the containment atmosphere response. Examples of small jets discharging into the containment atmosphere are the PCRV release through the safety relief valve and multiple exhaust ducts and accidental ruptured lines in the helium purification system.

Existing computer programs at GA, e.g., CONTEMPT (Ref. 1-7), CNTB (Ref. 1-8), and RATSAM (Refs. 1-9 and 1-10), are unable to describe fluid flow resistance in a PCRV depressurization system consisting of multiple components (such as in the above two examples). However, in a recent GA report (Ref. 1-11) a computational model for the prediction of the controlled depressurization of the FSV primary coolant system to the environment was presented. The computational model and its associated computer program (TRAIN) have been analytically and experimentally verified by FSV reactor test data. The TRAIN computer program (Ref. 1-11) has therefore been linked to a reduced version (CNTBO) of the CNTB computer programs (Ref. 1-8), which permits one to follow the transient state of pressures and temperatures prevailing in the containment atmosphere. The CNTBO computer program was successfully linked in the past to the RATSAM computer program (Refs. 1-9 and 1-10) to account for the counter pressure buildup in the containment.

The flow chart for the linked version of the TRAIN and CNTBO computer programs is shown in Fig. 1-4 (the steady-state routine CTNBT is part of the main program). The gaseous discharge to the containment atmosphere is optional; i.e., constant counter pressure to the environment and variable counter pressure to the containment.

Table 1-1 lists the 10 different types of components found in the FSV depressurization system which are modeled in the subroutine CHAIN (Fig. 1-4). The type order in Table 1-1 corresponds to the computational order in subroutine CHAIN. The purified helium compressor is not included in the depressurization system for FSV. For large HTGRs the compressors are located outside of the containment building.

The linked version of the TRAIN and CNTBO computer programs is operational and able to predict the global response of the containment atmosphere and the primary coolant system during the transient state following discharge by small jets.

1.5 COMPUTER CODE DEVELOPMENT

Development and updating of computer codes for safety and reliability analysis are pursued on a continuing basis. The work includes development of new codes and modifications to improve modeling based on the recommendations of the AIPA study, NRC, national laboratories, and others, as well as documentation and user's descriptions of present code versions.

In FY-78 the main task was to develop a method for analyzing a hypothetical DBDA with a steam generator steam leak. This has been achieved by developing a secondary system model for the RATSAM code by coupling the FLASH code to RATSAM. This resulted in the SPLAT code as discussed in the previous quarterly (Ref. 1-5).

During this quarter the validity of the new code was tested. This was accomplished by running several transients with SPLAT and comparing

ROUTINE

CHAIN

CTNBI

MAIN

MOODY

PIPE

MAIN FUNCTIONS

PRESSURE DROP IN COMPONENTS, EXCEPT PIPES

CONTAINMENT ATMOSPHERE STEADY STATE (OPTIONAL)

INPUT AND OUTPUT; CONTAINMENT ATMOSPHERE TRANSIENT STATE (OPTIONAL)

FRICTION FACTOR

PRESSURE DROP IN PIPES

Fig. 1-4. Flow chart for the routines in the linked version of the TRAIN and CNTBO computer programs

TABLE 1-1
COMPONENT TYPES FOUND IN THE FSV DEPRESSURIZATION SYSTEM

Type	Description
1	Pipe
2	Valve
3	High-temperature filter adsorber (HTFA)
4	Dryer bed
5	Helium purification filter
6	Low-temperature exchanger (LTX), first pass
7	Low-temperature exchanger (LTX), second pass
8	Low-temperature adsorber (LTA)
9	Flow element
10	Orifice

the results with FLASH 4 (Ref. 1-12) analysis. A typical transient simulated the following hypothetical accident scenario in a 3000-MW(t) HTGR:

1. Loss of main loop cooling.
2. Reactor trip.
3. Circulator trip.
4. Steam generator economizer tube rupture.
5. Steam generator loop isolated.
6. Steam generator dump.
7. Main helium shutoff valves closed.

The tube rupture was modeled as a hole or split in the tube wall. The SPLAT predictions of steam generator pressure and flow response during the transient were in good agreement with detailed FLASH-4 results.

1.6 NSS SYSTEMS ANALYSIS

The scope of the NSS systems analysis is to support the safety evaluation of the HTGR Generic Technology Program in the areas of thermal, fluid flow, acoustic, and dynamic analysis. In FY-78 this task included primary loop natural convection analysis during loop shutdown and acoustics and vibrational analysis related to establishing safety and design margins for primary loop components.

1.6.1. Primary Loop Natural Convection

The objective of the primary loop natural convection subtask is to evaluate natural convection and the suppression of natural convection in the primary coolant loop during loop shutdown or in stagnant or low flow

regions of the primary coolant during normal reactor operation. These latter areas include the CAHE loop and various horizontal and vertical PCRV penetrations.

The effort during the third and fourth quarters has been directed toward developing methods for predicting the suppression of natural convection in horizontal and vertical ducts. In the third quarter results from the laminar flow code VISFLO were generated for low Grashof number ($\sim 10^6$) and compared with experimental results on flow suppression. In this quarter analytical developments that are based on an integral approach were initiated. The integral approach has been used successfully to predict natural convection in open thermosyphons (Refs. 1-13 and 1-14). With this approach analytical methods applicable to high Grashoff numbers and ducts with tube bundles can be analyzed.

The first effort using the integral approach has been applied to predicting the suppression of natural convection in a vertical slot, as illustrated in Fig. 1-5. Adapting the procedures given in Refs. 1-13 and 1-14, the following approximate integral relations that govern the flow illustrated in Fig. 1-5 were developed:

$$\text{Continuity: } \int_0^1 \hat{u} \, d\hat{y} = 1$$

$$\text{Momentum: } \frac{d}{d\hat{z}} \int_0^1 \hat{u}^2 \, d\hat{y} = \frac{Gr \, \ell/b}{Re^2} \int_0^1 T \, d\hat{y} + \frac{\ell/b}{Re} \left. \frac{\partial \hat{u}}{\partial \hat{y}} \right|_{\hat{y}=1}$$

$$\text{Energy equation for entire flow field: } \int_0^1 \hat{u} \, \hat{T} \, d\hat{y} = 0$$

$$\text{Energy equation for upflow: } \frac{d}{d\hat{z}} \int_0^{\hat{\delta}} \hat{u} \, \hat{T} \, d\hat{y} = \frac{\ell/b}{RePr} \left. \frac{\partial \hat{T}}{\partial \hat{y}} \right|_{\hat{y}=\hat{\delta}}$$

Fig. 1-5. Analytical model for natural convection in a vertical slot

where
$$Gr = \frac{g\beta(T_H - T_L)b^3}{\nu^2}$$

$$Re = \frac{\rho u_o b}{\mu} \quad Pr = \frac{\mu c_p}{k}$$

By assuming the following velocity and temperature profiles with respect to the y-direction

$$\hat{u} = -s\hat{y} + \frac{s\hat{y}^2}{\hat{\delta}} \quad 0 < \hat{y} < \hat{\delta}$$

$$\hat{u} = -\frac{s}{1 - \hat{\delta}} \left[\hat{\delta} - (1 + \hat{\delta})\hat{y} + \hat{y}^2 \right] \quad \hat{\delta} < \hat{y} < 1$$

$$\hat{T} = \hat{T}_o - \hat{T}_o \left(\frac{\hat{y}}{\hat{\zeta}} \right)^2 \left[3 - 2\frac{\hat{y}}{\hat{\zeta}} \right] \quad 0 < \hat{y} < \hat{\zeta}$$

$$\hat{T} = 0 \quad \hat{\zeta} < \hat{y} < 1$$

and performing the noted integration, the following ordinary differential equations and algebraic equations were also developed:

Momentum:
$$f_1(\hat{\delta}) \frac{d\hat{\delta}}{d\hat{z}} = \frac{Gr}{Re} \frac{\ell/b}{2} \frac{\hat{T}_o \hat{\zeta}}{2} + \frac{\ell/b}{Re} \frac{2s}{1 - \hat{\delta}}$$

Upflow energy:
$$\frac{d}{d\hat{z}} \left[f_2(\hat{T}_o, s, \hat{\delta}) \right] = \frac{\ell/b}{RePr} \frac{6\hat{T}_o (\hat{\delta}/\hat{\zeta})^2}{\hat{\delta}} \left(1 - \frac{\hat{\delta}}{\hat{\zeta}} \right)$$

Continuity:
$$s = \frac{6}{1 - 2\hat{\delta}}$$

Overall energy:
$$f_3(\hat{\delta}, \hat{\zeta}) = 0$$

where f_1 , f_2 , and f_3 are merely complex algebraic relations. These four equations are sufficient to solve for the four unknown parameters $\hat{\delta}(\hat{z})$, $\hat{\zeta}(\hat{z})$, $s(\hat{z})$, and $\hat{T}_o(\hat{z})$ (see Fig. 1-5), and thus evaluate the flow field.

The above differential and algebraic equations are currently being verified; they will then be programmed to solve numerically. A preliminary set of integral equations has also been developed for natural convection in a vertical cylinder having a tube bundle, such as with the HTGR CAHE cavity.

1.7. NSS SYSTEMS ANALYSIS METHODS DEVELOPMENT

Development and updating of HTGR systems analysis computer codes are performed in support of the generic effort in the thermal and fluid mechanics and dynamics area, including general thermal and fluid flow codes for isothermal and nonisothermal, laminar, and turbulent flow applications. The main activity in FY-78 was to produce turbulent flow code capability to determine local heat transfer due to hot streaks.

1.7.1. Turbulent Flow Code Development

Substantial progress was made in two areas. Checkout and development of the MINT turbulent flow code is discussed in Section 1.7.1.1. Section 1.7.1.2 presents the status of the flow assembler code. The assembler code will allow relatively complicated flow geometries to be analyzed by assembling MINT solutions of simple geometries.

1.7.1.1. Computer Program MINT

Buoyancy force terms have been added to the momentum transport equations so that natural and mixed convection can be studied with MINT. Pressure and velocity boundary conditions have been coupled in order to approximate the effect of a finite flow resistance just behind the boundary. If, for example, the flow pattern in a plenum next to a tube bundle is to be calculated, the effects of tube bundle drag on the adjacent flow can be efficiently simulated with the new boundary condition. These two additions to the MINT code are currently being checked out.

Checkout of other MINT options, including in-core versus out-of-core solutions, continues with good results. The computer costs of executing

various problems were also estimated. A three-dimensional calculation of turbulent recirculating flow is expected to cost between \$300 and \$1000. Since this sum cannot often be justified as part of a development effort, turbulent recirculating models have thus far been confined to two dimensions. Along this line, a two-dimensional axisymmetric model of an HTGR core upper plenum was executed with MINT. The results will be presented in a MINT year-end summary report, which is now in preparation.

1.7.1.2. Turbulent Flow Assembler

The four steps in the process of calculating turbulent flows in complex regions are:

1. Partition a complex flow region into contiguous simple elements.
2. Calculate the flow behavior within each element with the MINT code for nominal boundary conditions and also for a number of perturbed boundary conditions.
3. Reduce the functional dependence of the flow behavior on the boundary conditions to a set of linear quadratic equations.
4. Join the element solutions by matching variables at common boundaries, and solve the complete set of linear equations simultaneously.

A computer routine to solve the complete set of linear equations for all the elements (step 4) was completed. The MINT code is currently being modified to solve perturbed solutions (step 2) quickly and efficiently once a basic solution has been found. The programming for a separate routine to accomplish step 3 is also in progress.

1.7.2. Miscellaneous Code Improvements

The on-line plotting capabilities developed in the second quarter for the TAC2D thermal analysis code were extended to produce hard copy finished plots. The input to the TAC3D thermal analysis code was streamlined and a final heat balance was added.

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2. STRUCTURAL ENGINEERING
189a NO. 00791; PROGRAM AREA 3

2.1. SUMMARY

Preliminary design drawings for the new circumferential prestressing system load monitor were completed and a test program plan was prepared. Computer code preparations to enable three-dimensional analysis of PCRV long-term behavior are in the checkout and debugging phase.

Preliminary resolution of NRC concerns regarding design bases for penetrations and closures has been completed. This work will be the basis of licensing submittals to NRC for specific designs. Static shear tests of liner cooling tubes and anchor studs have been completed and test setup and checkout operations are proceeding for the fatigue portion of those tests.

Analyses of load-bearing ceramic pad test results showed deficiencies, and design changes have been initiated to improve both the strength and fabricability of the pads. General Atomic has agreed to take the lead in preparing four Project Work Statements for cooperative thermal barrier development work under the Umbrella Agreement. The thermal barrier long-term resiliency test reached the 10,000 hour mark.

A preliminary evaluation of fifteen alternative core support designs has identified three for further development and evaluation. Graphite thermal stress fatigue tests have been delayed due to equipment problems, and the fracture mechanics test program has gotten under way after extensive replanning due to higher than expected test cost proposals.

Verification of the core seismic computer program MCOCO continued but was hampered by difficulties with allocated time on the computer and

high running costs. Several test reports on core seismic model tests were issued. A modified version of the MODSAP computer manual was also issued.

2.2. PRESTRESSED CONCRETE REACTOR VESSEL (PCRV)

2.2.1. Design

2.2.1.1. Load Monitor System

Preliminary design drawings of the new circumferential prestressing system load monitor were completed and are scheduled to be released by the end of September 1978.

2.2.2. Design Verification and Support

2.2.2.1. Relaxation Tests of 12.7-mm (0.5-in.) Stabilized Strand

The test analysis report on relaxation of prestressing strand is in the final review stages and will be issued in the near future. The report indicates that Lo-Lax strand produced by Florida Wire and Cable Company is suitable for use in the PCRV linear and circumferential prestressing systems. Details were reported in the previous quarterly report (Ref. 2-1).

2.2.2.2. Circumferential Prestressing System Load Monitor Test

A test program plan for the circumferential prestressing system (CPS) load monitor tests was prepared and is expected to be issued by the end of the quarter. The purpose of the test is to verify the accuracy, reliability, and serviceability of the new CPS load monitor design. The load monitor tests are a continuation of a long-term CPS qualification program which was started in 1968. Testing has been interrupted since 1975 due to funding limitations, which are expected to continue through FY-79.

2.2.2.3. Long-Term Structural Behavior of the Offset Core PCRV

For the study of the long-term PCRV structural behavior using the THREED computer program, it is necessary to establish a concrete material model. This material model should be of the rate type formulation and compatible with the main body of the THREED program. A standard linear solid model has been adopted to represent the viscoelastic material behavior of an aging concrete. This rheological model consists of a Maxwell and a Kelvin element in series. To determine the material parameters which characterize the creep behavior of the PCRV concrete, the nonlinear regression method was used for curve fitting of the available creep test data. The creep data are derived primarily from a study of the PCRV concrete properties conducted by the University of California for GA. The coding of the new concrete material subroutine is complete and incorporated into the THREED computer program. Debugging of the coding is in progress.

Parallel efforts have been directed to preparation of a three-dimensional finite element model of the offset core PCRV which will be used for the creep analysis. Because of the asymmetric PCRV layout, a 180° model of the full height of the vessel is required. It is planned to continue this work in FY-79.

2.3. LINERS

2.3.1. Design

2.3.1.1. Design Basis for ASME Class 1 Penetrations and Closures

In 1975 and early 1976, GA made presentations and submitted a report (Ref. 2-2) to the NRC which contended that failure of ASME Class 1 penetrations and closures is sufficiently improbable that it need not be considered as a design basis event for the HTGR. The report demonstrates that there is sufficient similarity between HTGR penetrations and closures

and LWR pressure vessels that extremely high levels of safety and reliability apply to both. In 1977, NRC in the Interim Safety Evaluation Report for GASSAR-6 reported that the proposed change is "potentially acceptable" but requested resolution of several items. A study program has resolved, in general terms, the following items:

1. The inservice inspection of the interface between the HTGR liner and the penetration closure is done to the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Division 2. A comprehensive volumetric inservice inspection program is planned for the penetration pressure-retaining welds. The program provides periodic verification of the integrity of this area.
2. The largest pipe passing through the penetration or closure is assumed to fail and the need for flow restriction is determined.
3. Design information and data verification are provided for the penetration-concrete interface and the shear anchorage system.
4. It is demonstrated that the integrity of the penetrations and closures would be maintained under the emergency and faulted conditions which would exist upon occurrence of one of the postulated design basis accidents.

It is planned to refine these results in terms of specific liner design applications in FY-79.

2.3.2. Design Verification and Support

2.3.2.1. Shear/Fatigue Tests of Liner Cooling Tubes and Anchor Studs

The static shear tests of the anchor studs and the cooling tubes have been completed. The ultimate shear displacement at failure will be used

for shear deflection design limits. The average initial stiffness characteristics of the anchor studs are almost identical to those derived from data published by the Nelson Stud Company.

Setup and checkout operations for the fatigue portion of the tests are expected to be completed by the end of September and the fatigue tests are to be performed early in FY-79.

2.3.2.2. Loss-of-All-Liner-Cooling Test Planning

A skeletal test plan was developed in which the overall objective is to gain sufficient data to demonstrate satisfactory behavior of the PCR/V thermal barrier, liner, and concrete during postulated elevated temperature events. The tests are to verify the adequacy of the PCR/V cooling water system vital/non-vital split, to demonstrate that only selected portions of the liner cooling are necessary during a DBDA, to demonstrate adequate safety margins for postulated thermal barrier failures, and to determine the failure mode and temperature distribution of the top plenum thermal barrier, liner, and PCR/V concrete during a postulated unrestricted core heatup event.

2.4. THERMAL BARRIER

2.4.1. Design

2.4.1.1. Redesign of Load Bearing Ceramic Assembly

The experimental and analytical findings reported in Section 2.4.2.1 clearly indicate design deficiencies in the load bearing ceramic blocks. Efforts have therefore been applied to improve the subject design.

The new assembly design uses a stack-up of six simple pads and three dowels (Fig. 2-1) instead of three, more complex pads employed in the old design. This new configuration was obtained by partitioning the previous

Fig. 2-1. Core support pad ceramic insulating assembly

assembly into thinner sections and into concentric cylindrical annuli in order to reduce thermal stresses and improve component fabricability.

A preliminary thermal stress analysis of this revised core support ceramic pad design was performed in the current reporting period. Both steady-state and transient thermal conditions were addressed. Based on this preliminary analysis, and recognizing that the structural criteria for ceramic materials remain relatively immature, the study indicates that the proposed ceramic blocks are structurally adequate. However, before the design can be verified, it will be necessary to improve the strength property data base of the ceramic materials being used and to advance the definition of the structural design criteria. It is also necessary to perform a detailed dynamic analysis of the ceramic assembly. It is proposed to pursue these efforts in FY-79.

2.4.1.2. Design Development Advancement

Alternate Material Candidate Evaluation. A summary was completed which documents the study conducted during FY-78 in the continuing evaluation of candidate metallic materials employed in the conventional Class A and Class B thermal barriers. The Class A material (carbon steel) has an extensive data base up to temperatures of about 400°C (750°F). Efforts have begun to investigate more closely the creep characteristics in the accident condition regime above this temperature range.

The reference Class B metallic material is Hastelloy X. Several issues relating to Class B materials have developed and continue to be studied. The two receiving the most attention concern the uncertainties of irradiation behavior and carburization. These issues have been noted in previous quarterly reports during FY-78. Alternate Class B materials continuing to receive attention are Hastelloy XR, cast Inconel 713LC, and Incoloy 800H. It is proposed to continue this work during FY-79.

2.4.2. Design Verification and Support

2.4.2.1. Structural Ceramic Studies

Thermal structural evaluation has been completed for the full-size ceramic pads tested at Southern Research Institute (SoRI). As previously reported, 12 of the 24 pads tested survived the 90,720 kg (200,000 lb) axial loading in combination with a steady-state thermal gradient. Three types of pads were tested (upper, middle, and lower). The least problems occurred with the middle pads, while various degrees of cracking occurred in the upper and lower pads. Analytically computed stresses and Weibull statistical analysis confirmed the basic inadequacy of the original upper and lower pad designs.

The upper pad was controlled by thermally induced stresses, while the lower pad was controlled by mechanically induced stresses. However, evidence of residual stresses was found in most of the failed upper pads. This was largely the result of inadequate outgassing of the material during the manufacturing firing cycle. For these reasons, pad redesign should aim to: (1) ease the outgassing process (thereby minimizing residual stresses and at the same time producing parts with more uniform densities, and (2) reduce the induced secondary (i.e., thermal) stresses.

The proposed modified design is shown in Fig. 2-1 and briefly described in Section 2.4.1.1. It is expected that this design, together with more stringent quality control standards and nondestructive evaluation methods, will result in a ceramic assembly for which a high confidence of pad integrity can be established.

2.4.2.2. Chatter Tests

The Boeing Engineering and Construction Company test report was received by GA on August 15, 1978. The data are being analyzed in preparation for the Phase I final test analysis report.

2.4.2.3. Proposals for DOE/FRG Umbrella Agreement

Following the progress reported in the previous quarterly report (Ref. 2-1), a letter was received from P. Rittenhouse (representing DOE at the KFA, Jülich) which reaffirms previous interest expressed by the German industrial partners for cooperation in the field of thermal barrier component development. The letter also requests GA to take the initiative in drafting Project Work Statements (PWSs) for the following tasks:

- CD-1 Thermal Barrier Design Application of Candidate Metallic Materials
- CD-2 Thermal Barrier Design Application of Candidate Ceramic Materials
- CD-3 Development of Common or Compatible Design Criteria for Metallic Materials
- CD-4 Development of Common or Compatible Design Criteria for Ceramic Materials

General Atomic has agreed to take the above action.

2.4.2.4. Vibration Assessment Studies

Preliminary planning of testing and analytical studies continued through the reporting period. Detailed analysis and planning for the next test phase also continued through the current reporting period.

2.4.2.5. Long-Term Resiliency Test

The scheduled milestone of 5000 hours (10,000 hours total) has been completed. Selected insulation samples are presently being removed and measured. The test autoclave is undergoing normal maintenance in preparation for startup with several new material samples. The next phase of the program will include additional low-density Saffil plus new materials such as Durablanket H, Fiberfrax H, and Kaowool 2600. Completion of refurbishment and scheduled startup are expected by early October.

2.5. REACTOR INTERNALS

2.5.1. Design

2.5.1.1. Revisions to ASME Boiler and Pressure Vessel Code

Graphite structural criteria are being written into the ASME Boiler and Pressure Vessel Code, Division 2, Subsection CE. Structural criteria studies are being carried out as a part of this program for inclusion in the Code. Eighty percent of the text of the initial draft of Subsection CE has been revised. A delay has occurred in the scheduled preparation of the remaining 20% of the text, caused by the rescheduling of more urgent tasks. The revised draft will be submitted to the joint ACI/ASME Code Committee for review during the first quarter of FY-79.

2.5.1.2. Oxidation-Resistant Core Support Structure Study

The final report on this study is currently being reviewed. A total of fifteen alternative proposals were investigated. Of these, the report recommends that the following three concepts be developed further in sufficient detail to fully evaluate their feasibility:

1. A design in which the upper portion of the core support block is produced in seven individual pieces which are manufactured from a graphite which is much more oxidation resistant than PGX (e.g., H-451 or Stackpole 2020).
2. The introduction of a purer (lower ash and iron content) version of PGX graphite or a totally new graphite.
3. A design in which each individual fuel column is supported by its own individual post so that smaller sized, lower oxidation rate graphite can be used.

In addition, the possibility of changing critical system parameters such as core outlet temperature and primary coolant moisture level limits should be considered as an integral part of these design changes.

It is anticipated that this report will be issued during the first quarter of FY-79.

2.5.2. Design Verification and Support

2.5.2.1. Graphite Thermal Stress Limit and Fatigue Tests

Specimens of core support and permanent side reflector graphites are being tested under conditions inducing large thermal gradients to develop design allowable values for secondary thermal stresses in both single-cycle and fatigue conditions. The design function of the core support and permanent side reflector requires that these components maintain a high performance capability throughout the life of the reactor. These thermal stress limit and fatigue tests of the various graphites used will enable the development of analytical methods and failure criteria which will result in a safe, reliable design.

Southern Research Institute has experienced a program delay of approximately 3 months caused by the need to relocate some of the high-power test equipment in an isolated area of the plant.

The remaining low-power (ATJ) and high-power (PGX, 2020, and ATJ) tests will now be completed during the first quarter of FY-79, with the thermal fatigue cycle testing of PGX possibly not being completed until the second quarter of FY-79.

2.5.2.2. Thermal Mixing Tests

Tests are being conducted on a range of flow channel configurations in the bottom reflector blocks and core support blocks with the objective

of improving mixing of the coolant streams from the different columns of a fuel region, at the least possible penalty in increased pressure drop.

The final phase of these tests is proceeding with approximately 90% of the testing being complete. The quality of the data has been increased as a result of installation of an improved simulated temperature sensor which provides predictable multiple heated column results.

Completion of the tests and issue of the final report will be accomplished during the first quarter of FY-79.

2.5.2.3. Fracture Mechanics Test

Southern Research Institute submitted a reduced scope cost proposal in line with the FY-78 funds available and was awarded a contract for the remainder of FY-78. Initial test organization and planning have been under way, and the test material has been delivered to SoRI.

Specimen preparation and environmental chamber fabrication will be initiated during the first quarter of FY-79, with testing expected to begin in the second quarter of FY-79.

2.6. STRUCTURAL MECHANICS

2.6.1. Design Verification and Support

The overall objective of this task is to develop, verify, and maintain state-of-the-art analysis techniques in the areas of structural mechanics, structural dynamics, and seismic analysis. In FY-78 the activities under this task are divided into two work areas: core seismic program and general methods development.

2.6.1.1. Core Seismic Program

The core seismic program is an ongoing effort which aims to develop the necessary analytical and test technology to evaluate the seismic accuracy of the HTGR core and core support structures. This program interfaces with the Fuel Element Design task, which covers seismic strength tests and stress analysis. These activities are reported elsewhere.

Model Testing. This subtask deals with scaled model tests designed to provide basic characteristic data on individual core element behavior and to give information on the interaction of groups of elements including two- and three-dimensional core arrays. The test data are used for correlation and verification of the analytical codes. These tests have been described in previous progress reports and their preparation was completed prior to FY-78. The test reports are now in various stages of preparation and review.

This quarter three test reports were issued:

1. One-Fifth-Scale and Full-Scale Fuel Element Rocking Tests (Ref. 2-3).
2. One-Fifth-Scale and One-Half-Scale Two-Dimensional Seismic Tests (Ref. 2-4).
3. One-Half-Scale, Two-Dimensional, Two-Axis Seismic Tests (Ref. 2-5).

Analytical Development. Substantial efforts were further expended this quarter to find ways to reduce the computer costs associated with running the core seismic computer programs. This is a severe problem, in particular for codes like CRUNCH-2D and MCOCO. For this reason verification of these codes has not been completed according to the planned work scope of comparing analytical results with test data by simulating time history earthquake and sinusoidal sweep test conditions. The cost of such

computer runs has turned out to far exceed the allocated budget for FY-78 and the time permitted on the UNIVAC 1110 computer system.

A study was conducted to determine the effects of reduced interblock stiffnesses using CRUNCH-1D. This study indicated that reducing the stiffness values significantly reduced the cost of core seismic analysis without distorting the core behavior characteristics. However, the resultant loads would have to be increased to reflect the actual spring stiffness values. This and other cost reductions described in the previous quarterly (Ref. 2-1) alone are not enough to produce reasonable running time and costs. To achieve this, an array processor for use with the computer hardware is required, which is estimated to cut the running time on the computer by an average factor of 10. Hopefully this piece of equipment, which can also be utilized by other computer programs with similar problems, can be purchased in the near future.

The MCOCO correlation with the one-fifth-scale full-array test is still in progress. Sinusoidal dwell simulations at several frequencies about the resonance point have been completed and comparisons between test and analysis are being made. Sine sweep input is also being incorporated in the code, although the cost of running even a small sweep range is extremely high.

The full-array model in a typically deflected mode for a sine dwell is shown in Fig. 2-2. The earlier correlation for MCOCO showed discrepancies between analysis and test in permanent reflector block rotation and led to the incorporation of a reflector block link. This has reduced the permanent reflector block rotations to an acceptable level. Refinement of the full-array model has been in progress but is limited by the enormous running times and cost as discussed above.

Core Seismic Licensing Topical Report (LTR). A skeletal draft LTR for the core seismic program was written for submission to the NRC. This document covers the development of core seismic computer codes, model

Fig. 2-2. One-fifth-scale core deflections from sine-dwell excitation at 3.6 Hz and 0.5 g

testing, and verification of codes with test data. The LTR is written in conjunction with the licensing pre-application review task initiated with the NRC to cover several important design and safety issues.

2.6.1.2. General Methods Development

Methods development for the analysis of gas-cooled reactor components, including the PCRVR, core elements, and core support structures, as well as soil/structural interaction, has been a major effort in prior years. Presently, computer code updates are performed to keep abreast of new techniques in this area, and modifications are made to support DOE licensing requirements and to implement the most cost effective methods for analysis, as well as code documentation and maintenance.

Cathode Ray Tube (CRT) Hardware and Software Development. This sub-task deals with the development of software for interactive CRT demand processing terminals to improve data preparation and displays of results. During the current reporting period, a dual interactive buffer was installed in the Tektronix 4014-1 terminals. This buffer permits 512 characters or 200 vectors to be displayed in refresh, rather than storage, mode. Software was developed to facilitate the implementation of this new feature into Fortran plotting programs. Two initial applications include animated displays of structural models and interactive conversation with users using a screen workspace area while developing a stored display.

Code Development. The revised MODSAP computer manual was issued as GA-A14006 (Rev) (Ref. 2-6).

2.7. REFERENCES

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3. HEAT EXCHANGER EQUIPMENT 189a NO. 00793; PROGRAM AREA 4

3.1. SUMMARY

Conceptual design work on the core auxiliary heat exchanger (CAHE) continued with emphasis on further development and optimization of the bayonet tube configuration. Additionally, work was started on a CAHE design specification which will be used to direct the design.

In the design verification and support (DV&S) area, a preliminary review version of the overall heat exchanger DV&S Plan was completed. Also a Test Plan for heat exchanger pressure boundary in-service inspection (ISI) was completed in draft form for review. In methods development, modifications were incorporated in both NUSIZE and SUPERHEAT computer programs with preliminary user's manuals scheduled for completion in September 1978. A user's manual for CSTRES is also scheduled for completion for review in September 1978.

3.2. DESIGN

3.2.1. Core Auxiliary Heat Exchanger Design and Analysis

The fourth quarter effort included (1) an update of the thermal sizing for new systems conditions, (2) revisions of the general arrangement drawing, (3) preparation of a design specification, and (4) preparation of long-range work plans.

The resizing of the CAHE for updated systems conditions resulted in a candidate redesign of the bayonet tube configuration. The redesign was precipitated by several concerns stemming from continuing design development,

such as general concerns regarding ISI capability, design adequacy of the combined peripheral seal/lateral seismic support, and excessive tube support grid pressure drop. [The grid design was discussed in detail in the previous quarterly report (Ref. 3-1)].

In the resizing analysis it was found that by packing the tubes on the closest pitch possible (as limited by minimum tubesheet ligaments and radiography limits for NDE of the internal-bore tube-to-tubesheet welds), the heat transfer coefficients increased, decreasing the overall heat transfer tube length to the point where only one tube support grid was required.

In order to convey the hot helium into the tube bundle with the closely spaced tubes, it was necessary for the gas to enter the tube bundle at the top of the bayonet tubes and exit the tube bundle at a location immediately above the tubesheet at the outer periphery of the bundle. This resulted in a smaller pressure drop at the tube bundle inlet and also eliminated the problem of having to insulate the tubesheet against the hot inlet helium. Cooled helium emerging from the heat exchanger is collected in a vertical duct through which it is conveyed to the auxiliary circulator. The revised CAHE arrangement and flow paths are illustrated in Fig. 3-1.

Additional advantages with this configuration are that the shorter tube bundle reduces seismic loadings, improves static and dynamic stability, and discourages natural convection of gas within the bundle because of reduced elevation heads.

It is intended that the single tube support grid would be located at the top of the tube bundle. Extensions will be provided on the end caps of the tubes which will be inserted into mating holes in the support plate. This will provide more free flow area in the tube support grid since the grid will not be required to encircle each tube, but only to hold the "pins" extending from the end caps.

- NOTES
1. ALL DIMENSIONS ARE NOMINAL AND COLD
 2. LINER THERMAL BARRIER DIMENSIONS DO NOT INCLUDE FASTENER. FASTENERS PROTRUDE 1.00" MAX BEYOND DIMENSIONS SHOWN
 3. DIMENSIONS WITHIN BRACKETS [xxxx x] ARE IN MILLIMETERS.
 4. MIN DIMENSION REQUIRED FOR CAHE OPERATION

27	PISTON RING SEAL	SB 409 GR1 SOLTREATED
26	INSULTG SHROUD CAP 60"ODx5"TK	
25	INSULATING SHROUD 60"ODx5"TK	
24	GRID ASSY 50"OD	
23	OUTLET DISTRIBUTION CONE 50"TK	
22	INLET DISTRIBUTION CONE 50"TK	
21	SHROUD LINER 58"ODx.25"TK	SB 409 GR1 SOLTREATED
20	TUBE JUNCTION 1.50"ODx4.5"LG	SA 334 GR1
19	TUBE INNER .625"ODx.065"WALL	SA 334 GR1
18	TUBE INNER .875"ODx.065"WALL	SA 334 GR1
17	TUBE CAP 1.375"OD 4"LG	SA 336 F22
16	TUBE OUTER 1.375"ODx.14"WALL	SA 213 T22
15	OUTLET PIPE 2.400x.50TKx8.5"LG	SA 516 GR 70
14	SECONDARY HEMIHEAD 52"ODx.50TK	SA 516 GR 70
13	SECONDARY FLANGE 56"ODx.50TKx6"TK	SA 508 CL1
12	SECONDARY TUBESHEET 56"ODx6"TK	SA 508 CL1
11	HEX NUTS	SA 194 B16
10	STUDS 2.5"DIA x 20"LG	SA 193 B16
9	10 SCH 30 GRAYLOC HUB	SA 508 CL 2
8	24 SCH 60 GRAYLOC HUB	SA 508 CL 2
7	PIPE JUNCTION 36"ODx22"ODx14"LG	
6	10" WELDOLET	SA 509 CL 2
5	INLET PIPE 36"ODx1.5TKx62"LG	SA 302 GR D
4	PRIMARY HEMIHEAD 52"ODx2TK	SA 302 GR D
3	PRIMARY FLANGE 74"ODx50TKx2TK	SA 508 CL 2
2	PRIMARY TUBESHEET 30"ODx5TK	SA 508 CL 2
1	SHROUD ASSY 51"ODx.5TKx10FT LG	SB 409 GR1 SOLTREATED
	CAHE ASSY	

Fig. 3-1. Revised CAHE arrangement and flow paths

Future work will concentrate on development of further design detail of the gas entrance and exit configurations, detailed pressure drop analysis, and mechanical design features.

A design specification and statement of work were started and will be used to direct the design. They will also be used in bid packages to solicit proposals from prospective suppliers for design and fabrication of the CAHE in FY-79.

3.3. DESIGN VERIFICATION AND SUPPORT (DV&S)

3.3.1. DV&S Program Plan

A draft of the overall DV&S Plan was prepared for review. It is scheduled for issue by the end of September 1978.

3.3.2. DV&S Test Plan for Heat Exchanger Pressure Boundary In-Service Inspection

A draft of the Design Verification and Support Test Plan for Heat Exchanger Pressure Boundary In-Service Inspection was prepared for review in early September. This plan is scheduled for issue by the end of September.

Primary coolant HTGR heat exchangers generally require special penetrations through the pressure vessel for secondary coolant supply and return lines. The penetrations are generally in two parts. One is an integral part of the heat exchanger and the other is an integral part of the pressure vessel. These parts are joined together to complete the primary coolant pressure boundary at the time the heat exchanger is installed. The design of the penetration must therefore meet the criteria of both the heat exchanger and the pressure vessel, usually resulting in complex shaped forgings, thermal sleeves, numerous welds, and a variety of materials. The penetrations must be designed to permit ready ISI per

the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Division 2.

The purpose of the DV&S program will be to determine the feasibility of ISI on typical heat exchanger penetrations and establish design constraints, criteria, and/or modifications required to meet the ISI rules, including development of the required criteria, equipment, techniques, and design modifications to permit ISI in accordance with ASME requirements.

The program will initially consist of six tasks, briefly outlined as follows:

1. Bimetallic welds. Tests will be performed to determine the inspectability of bimetallic welds and welds made on safe-ends for the material and thickness parameters.
2. Backing strips. Tests will be performed to determine the inspectability of welds made with permanent backing strips for the material, environment, and joint thickness parameters.
3. Access. Tests will be performed to determine the minimum access required for manual and remote inspection for welds in both the horizontal and vertical positions.
4. Special contours, shapes, and sizes. Tests will be performed to determine the inspectability of various shapes and contours which, due to their geometric configurations, represent potential ISI problems.
5. Instrument environment. A review will be made of ISI sensor breakdown problems related to high radiation levels, and an economic tradeoff study will be carried out covering current surface temperature limitations to prevent detector damage versus the potential savings achievable by developing sensors capable of operating at higher temperatures.

6. European requirements. A study of the feasibility of meeting the current European ISI requirements will be included.

As the program proceeds, the need for additional tasks as well as modifications and/or elimination of existing tasks is anticipated. These changes (potentially resulting from code revisions, new NRC requirements, design changes, and/or test results) will be included in the program through modification of the Test Plan and any necessary supporting documentation.

3.3.3. Heat Exchanger Methods Development

An existing version of the SUPERHEAT computer program has been selected as the standard production version and has been established on a Univac 1110 catalog file. Although no major modifications were made, minor revisions and corrections were incorporated. A user's input and description manual is being prepared for draft review by the end of September.

The NUSIZE program has been revised with modifications which correct the option 1 calculations, simplify the amount of input, delete unused subroutines and unused portions of subroutines, correct the output for the straight tube axial flow option, and add a plotting capability. A preliminary user's manual is being written and will be distributed for review before the end of the quarter. The manual will be published in the first month of the next quarter.

A test version of the CSTRES computer program has been created on a Univac catalog file. The input format was changed to the standard form and some modifications and corrections to the program were also made. A user's input and description manual is being prepared for review by the end of September.

3.4. REFERENCE

- 3-1. "HTGR Generic Technology Program: Safety, Systems and Component Design and Development, Quarterly Progress Report for the Period Ending June 30, 1978," DOE Report GA-A15055, General Atomic Company, June 1978.

4. MECHANICAL AND ELECTRICAL COMPONENTS 189a No. 00799; PROGRAM AREA 5

4.1. SUMMARY

In studies of the overall main helium circulator arrangement with an external driver, the selection of a main high-pressure shaft seal was considered, covering both sliding and clearance face seals. A sliding seal of the type used in pressurized water reactor (PWR) cooling pumps was selected. Work was also started on the preparation of a functional specification for the main circulator. A valve-plate-mounted fluid nozzle was studied as a shutoff valve actuation augmentor. A valve position indicator was also investigated.

The aerodynamic design of the circulator compressor was revised to fit new cycle conditions. A comparison of designs with vaned and vaneless diffusers showed that the vaneless diffuser was acceptable for the circulator requirements. A complete performance map was generated for the compressor with a vaneless diffuser. The effect of reducing inlet diameter to produce higher inflow velocities for more sensitive flow measurement was studied. It was found that an unacceptable penalty in compressor performance resulted from this approach.

4.2. MAIN HELIUM CIRCULATOR

4.2.1. Design

4.2.1.1. Conceptual Overall Design

Various arrangements for the high-pressure shaft seal were explored. The basic options are clearance seals or sliding face seals. The selected design is a rubbing face seal which can be replaced without major

disassembly. A mechanical face seal of this type is commonly used in PWR cooling water pumps and is within the range of commercial application. In addition, a mechanical seal is essentially a zero leakage device and will simplify the circulator service system.

Progress was made on an overall functional specification for the main helium circulator. A proposed outline was issued for comments and work has started on the main body of the text.

4.2.1.2. Shutoff Valve for Centrifugal Compressor

Studies on the flapper valve with a fluid nozzle attached to the valve plate have been completed (Fig. 4-1). This type of fluid assist actuator would provide a constant force independent of valve position.

A removable LVDT which would indicate valve plate position is shown in Fig. 4-2. The LVDT could be replaced by pulling it out through the guide pipe. The design is such that the valve can be installed and removed without disturbing the position-indicating mechanism.

4.2.1.3. Aerodynamic Design Studies

The design of the centrifugal compressor for the HTGR main circulator was revised for the new cycle conditions given in Ref. 4-1. The entire range of speeds covered in the previous studies, 1600 to 2400 rpm, was covered in a preliminary calculation, and a speed of 2200 rpm was selected. The performance of the compressor for this speed was calculated using Ref. 4-2, and the tip diameter was adjusted to give the correct exit pressure at the calculated efficiency. Details of the new operating conditions and the dimensions and performance for the revised compressor design are given in Table 4-1.

A comparative study was carried out to evaluate the advantages and penalties involved in the use of a vaneless diffuser for the main circulator. To make the comparison, a vaned diffuser was designed for the compressor.

Fig. 4-1. Flapper valve with fluid nozzle attached to valve plate

Fig. 4-2. Valve plate position-indicating mechanism

TABLE 4-1
COMPRESSOR FOR MAIN HELIUM CIRCULATOR, 900-MW(e) HTGR POWER PLANT

Cycle Requirements

Helium pressure at circulator inlet, MPa (psia)	7.125(1033.44)
Helium temperature at circulator inlet, °C (°F)	313.9 (597)
Required pressure rise, kPa (psi)	116.2 (16.86)
Required exit pressure, MPa (psia)	7.238(1050)
Loop mass flow rate, kg/s (lb/s)	292.8 (645.9)
2% allowance for leakage, kg/s (lb/s)	5.85 (12.9)
Circulator mass flow rate, kg/s (lb/s)	298.7 (658.8)

Compressor dimensions

Tip diameter, mm (in.)	1,712.97 (67.44)
Tip width, mm (in.)	122.94 (4.84)
Shroud diameter, mm (in.)	1,034 (40.71)
Hub diameter, mm (in.)	475.49 (18.72)
Axial length, mm (in.)	304.8 (12)
Blade exit angle (from tangent), degrees	60
Blade inlet shroud angle, degrees	34.55
Blade inlet hub angle, degrees	60.39

Compressor performance

Overall efficiency (including disk friction and exit loss), %	79.16
Input shaft power, kW (hp)	7,594 (10,180)
Input shaft torque, N-m (lb-ft)	40,287 (24,303)

The diffuser was of conventional design with vanes of "circular arc" shape designed as in Ref. 4-3. The number of passages was set by the requirement of having a "throat aspect ratio" of roughly 0.75. The exit radius ratio of the diffuser was made equal to that of the vaneless diffuser (2.2) and the leading edge of the vanes was set at a radius ratio of 1.14. With these radius ratios, the exit angle of the vanes was varied until the combination of vane opening ratio and vane curved length gave the equivalent of a linear diffuser with a total included angle of approximately 7° . The vaned diffuser axial height was set at 0.98, the impeller tip width, as was that of the vaneless diffuser.

The off-design performance of the two versions of the compressor was calculated using program "PREDM" (Ref. 4-2). Only one speed line, that for 100% of design speed, was calculated, with a flow rate range from 60% to 180% of design flow rate. The resulting performance curves are shown in Fig. 4-3 as compressor efficiency versus the inlet flow coefficient. Figure 4-3 clearly shows that although a marginal improvement in efficiency is obtained with a vaned diffuser over part of the flow range (about 1.0 percentage point at the design flow), higher efficiency is maintained over more of the flow range with a vaneless diffuser. The selection of a vaneless diffuser for the HTGR main circulator centrifugal compressor therefore appears to be correct in view of the requirements of the circulator.

The complete performance map of the compressor with a vaneless diffuser was generated using Ref. 4-2. The results of the calculation were stored on a magnetic drum file which can be used by systems analysis programs directly. The calculation produces 24 parameters for each operating point including direct, nondimensional, and normalized values of the parameters. A typical plot of the nondimensional parameters is shown in Fig. 4-4, which presents the pressure rise coefficient ψ and compressor efficiency η as functions of the flow coefficient ϕ . ψ and ϕ are defined as:

Fig. 4-3. Comparison of vaneless and vaned diffusers, off-design performance at 2200 rpm

Fig. 4-4. Main circulator off-design performance

$$\psi = g \frac{\Delta P}{\rho U_{tip}^2}$$

$$\phi = \frac{Q}{A_{tip} U_{tip}}$$

where A_{tip} = rotor frontal area, m^2 (ft^2)

$g = 1.0 \text{ kg-m-s}^{-2}\text{-N}^{-1}$ ($32.2 \text{ lb}_m\text{-ft-s}^{-2}\text{-lb}^{-1}$)

ΔP = pressure rise, Pa (lb_f/ft^2)

U_{tip} = rotor tip speed, m/s (ft/s)

Q = volumetric flow rate, m^3/s (ft^3/s)

ρ = gas density, kg/m^3 (lb_m/ft^3)

The results are given for speeds from 20% to 120% of design speed. It will be noted that over the range of speed from 40% to 120% of design speed, the curves lie close enough together that the performance of the compressor could be represented by a single curve of ψ versus ϕ . This, of course, is the reason for using nondimensional parameters. The deviation of the curves at lower speed is caused by Reynolds number effects not included in the formulation of ψ and ϕ .

A further study was carried out to explore the possibility of increasing the sensitivity of the flow measuring equipment by increasing the meridional velocity of the gas entering the compressor. It was determined that a reduction in inlet diameter from the design value of 1016 mm (40 in.) to a value of 762 mm (30 in.) would be required. The design of the compressor was therefore modified to obtain inlet diameters of 952.5, 889, 825.5, and 762 mm (37.5, 35, 32.5, and 30 in.), keeping meridional velocity at the rotor exit equal to that at the rotor inlet for each diameter. The performance of each of the modified compressors was analyzed using Ref. 4-2. The results showed that the efficiency would be reduced steadily with decreasing inlet diameter and would be 60.4% at an inlet diameter of

762 mm (30 in.). This penalty was considered too great to justify this approach to improvement of flow measurement.

4.2.2. Design Verification and Support

4.2.2.1. Main Helium Circulator Tests

Circulator Service System Tests. With the steam-driven prototype circulator in working order, the bearing water system, including the shutdown bearing water system, was checked and found to give satisfactory performance. During this period several problems with malfunctioning helium valves and a leaking brake and seal manifold had to be solved by disassembly and repair.

Full-Speed Tests. Full-speed tests with a helium/nitrogen mixture in the steam loop were conducted and a maximum speed of 7140 rpm was reached. The test vessel was filled with helium at atmospheric pressure. A compressor performance map was taken and circulator coastdown tests were performed.

Unbalance Tests. After completion of the full-speed tests, the circulator was modified and an unbalance of 3600 mm-g (5 in.-oz), and later 7200 mm-g (10 in.-oz), was added to the turbine disk. The circulator was started up and a speed of 1500 rpm with full bearing water and buffer helium flow was established. While the speed was held constant, the bearing water flow was gradually reduced from full flow of $14.02 \times 10^{-3} \text{ m}^3/\text{s}$ (185 gal/min) to $5.68 \times 10^{-3} \text{ m}^3/\text{s}$ (75 gal/min). At certain predetermined bearing water flow points, the shaft displacement was recorded and a photograph was taken of the displacement wave pattern as displayed on the oscilloscope.

This procedure was repeated for 3000, 3500, 4000, 4500, 5000, 5500, and 6000 rpm for both unbalance conditions. The test was terminated and the bearing water flow increased whenever the shaft displacement began to

exceed the safe limits. The unbalance data were recorded and will be evaluated in the test analysis portion of the test report.

Shutdown Seal Leakage Tests. In order to perform this test, the seal interspace test line was modified. A sample volume of 3.8 liters (1 gallon) was added, plus three valves and a pressure gauge to bring this sample volume on-line with the seal interspace and to isolate this subsystem from the helium supply. The rate of pressure decay in this newly created subsystem, consisting of seal interspace, feedline, and sample cyclinder, was measured and recorded for different seal interspace pressures and various seal actuation pressures. Since the sealing surfaces were purposely not silver plated, it is expected that a comparison with Fort St. Vrain shutdown seal leakage rates will show an increase in seal leakage. The test results will be evaluated in the test analysis portion of the test report.

4.3. REFERENCES

- 4-1. "Plant Technical Description - Reference 900," General Atomic unpublished data, May 22, 1978.
- 4-2. "Program PREDM: Multistage Centrifugal Compressor Performance Evaluation," Northern Research and Engineering Corporation (NREC) Report 900AXA-1 (program proprietary to NREC).
- 4-3. Eckert, B., Kroeners Taschenbuch der Maschinentechnik, Vol 2, Part II, Alfred Kroener Verlag, Stuttgart, 1956.

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GENERAL ATOMIC COMPANY
P. O. BOX 81608
SAN DIEGO, CALIFORNIA 92138