HTGR SAFETY EVALUATION DIVISION

DEPARTMENT OF APPLIED SCIENCE

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SUMMARY REPORT

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**HTGR Safety Evaluation Division
Brookhaven National Laboratory**

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Progress Report

Introduction

The HTGR Safety Evaluation Division at BNL was formed in March 1974. The purpose of the group was to provide research and analytical support for RSR on aspects of HTGR's related to public safety. A similar effort was initiated at LASL at the same time.

A substantial fraction of the effort during the first part of the year was devoted to the development of the HTGR Research Program Outline. This was done in collaboration with LASL. During part of the first year some of the BNL personnel also provided assistance to the RSR staff in Washington.

In addition to BNL personnel, the HTGR group is attempting to develop a wide national and international base of HTGR expertise for RSR by close collaboration with consultants and subcontractors at different universities and scientific establishments. Yearly meetings are being planned in which the consultants, subcontractors, BNL and LASL staff and RSR representatives will review, criticize and discuss programs for existing and future work on HTGR safety.

In an effort to provide as thorough a program as possible on HTGR Safety, the BNL work includes both specific activities involving specialized expertise and total group efforts utilizing an interdisciplinary approach to accident delineation in particular and HTGR public safety in general. The interdisciplinary group efforts involve long-term projects such as development of sequences and consequences of both multiple failure accidents

and design based accidents. In addition, the group is involved in reviews and evaluation of existing LTR's and safety related reports written by GA and ORNL.

Some of the group interdisciplinary projects initiated in the first year include development of the sequences and consequences of accidents in which one or more core support columns fail, and total loss of coolant accidents leading to possible sublimation of graphite. We have also begun a comparison of the German, United Kingdom and GA design based depressurization accidents. In the latter case for example, the project is being carried out, in part, with the collaboration of HTR people at KFA, Dragon, and Euratom.

A necessary part of the first years work was involved in collecting and evaluating existing literature pertinent to safety related HTGR problems. Although our evaluation studies are far from complete, and present conclusions may be significantly altered by information not yet reviewed, a number of preliminary evaluations are summarized below.

1. Fission Product Release and Transport

1.1 Fuel Failure Mechanisms (Preliminary Evaluation)

The current HTGR reference fuel design (Summit Nuclear Steam Station) consists of TRISO coated UC_2 fissile particles and BISO coated ThO_2 fertile particles. For the TRISO particle, the fuel kernel is surrounded by successive layers of low density pyrocarbon (buffer), high density pyrocarbon (inner PyC), silicon carbide (SiC) and high density pyrocarbon (outer PyC). The BISO particle coating consists of a low

density pyrocarbon (buffer) inner layer and a high density pyrocarbon outer layer. The overall diameters of the TRISO and BISO particles are nominally 540 μm and 820 μm , respectively. There will be approximately 40% TRISO and 60% BISO particles in a large HTGR. An important recent change in the manufacture of the fuel is the use of the weak acid resin technique (WAR). The process produces a glassy carbon inner layer which is supposed to reduce the amoeba effect.

Approximately 10^{11} coated particles are held in cylindrical rods (sticks) by a matrix consisting of carbonized pitch and graphite shim material. The fuel sticks are inserted into blind holes drilled in the hexagonal graphite fuel blocks and each channel is sealed with graphite pitch material. Thus, during normal operation, most of the fission products are retained by (1) coatings of intact TRISO and BISO particles and (2) intact graphite fuel block channels (for fission products which are released from defective particles). The BISO particle coatings allow a diffusive release of certain fission products under normal operating conditions.

The coatings are considered the primary barrier to fission product release in the HTGR concept. However, fuel particle coating failures have been observed and the failure mechanisms are presently under study. Fuel failure mechanisms which have been observed and reported are as follows:

- (1) Pressure vessel. Fission gas pressure builds up within the particle until the tensile strength of the individual coating is ex-

ceeded. Since fission gases are generated throughout the lifetime of the fuel, pressure vessel type failures occur with greater frequency toward the end of life.

(2) Kernel migration. The fuel kernel slowly moves from the cold side of the particle toward the hot side by one or more mechanisms until the integrity of the buffer and inner pyrocarbon layers have been destroyed. These mechanisms occur throughout the lifetime of the fuel and, as with pressure vessel failures, become pronounced near end of life. Kernel migration coefficient (time-temperature relationship) measurements have been made and are used to determine the fuel life cycle (4 years). However, extensive measurements have not been made on irradiated (to design burn-up) WAR processed reference fuel.

(3) Fission product interaction. This involves the diffusion of certain rare earth fission products (Ce, Nd) to the SiC layer of TRISO particles and subsequent interaction resulting in failure of the layer. The mechanism is temperature limiting; rare earth fission products begin to accumulate at the interface starting at $\approx 1300^{\circ}\text{C}$ with 100% failures occurring above $\approx 2000^{\circ}\text{C}$.

(4) Fabrication defects. Following the manufacturing process, contamination caused by defective or missing coatings, is present. General Atomic estimates that its quality control procedures can assure that $< 0.2\%$ of the particles are in this category. Thus, based upon 10^{11} total particles in an 1160 MWe HTGR, $\sim 2 \times 10^8$ failed particles will pass inspection and be present in the core from startup.

Some of the more recently discovered modes of failure which

are still not thoroughly understood are as follows:

(1) Loss of coatings by matrix adhesion. Under certain circumstances which create large differences in thermal expansion, the matrix adhesive forces on the particle coating are high enough to crack and tear away sections of the outer pyrocarbon layer.

(2) Liquification of fuel kernel. There is evidence to show that under certain circumstances, fuel kernels have liquified. This is believed to be associated with the conversion of the carbide particle to an oxide by a reaction involving oxygen impurity (from fission reactions).

(3) Interaction with Pd. The formation of Pd fission product from decay of Pu leads to an interaction and attack by Pd on the SiC layer. This reaction may have serious consequences if large amounts of Pu are added to future fuels.

Current approaches to the mechanical design of coatings have used the principles of pressure vessel design. The materials involved are relatively brittle and non-yielding. The SiC layer, for example, depends upon the outer pyrocarbon layer for restraint in order that the SiC layer does not exceed its design allowable stress of 30,000 psi after full burnup (75% FIMA). A possible weakness of the stress model may be its failure to include the effects of neutron induced PyC creep.

The thermal performance of the fuel is influenced by kernel migration and fission product attack of the SiC layer. Both of these effects can limit fuel performance and are affected by transient temperature excursions as well as undetected hot channels. In the hypothetical event of a

loss of coolant accident coupled with a delayed start of the emergency core cooling system, core temperatures can rapidly increase to a level ($> 1600^{\circ}\text{C}$) where large numbers of fuel particles may fail. In addition to assessing the hazards resulting from the quantities of fission products released, the behavior of the individual fuel kernels needs to be understood at these high temperatures. For example, as UC_2 or ThO_2 approach their melting points, will the kernels remain virtually intact or will they liquify, vaporize or form low melting compounds. Certain observations regarding kernel migration are still not well understood. For example, UC_2 kernel migration is slightly lower in irradiated particles than non-irradiated particles, suggesting a possible dependence upon radiation defects or fission product content. In addition, the ThO_2 amoeba effect does not occur unless the particle is irradiated. Both of these observations would suggest some relationship with irradiation or fission product content.

Uncertainties such as those discussed above are typical reasons why additional experimental verification programs would be required on any proposed new fuel concept to understand the possible failure mechanisms peculiar to that fuel design. In-pile testing of the "reference" fuel has only recently been started by GA with the first specimens scheduled to be examined in mid 1977. The FSAR for the Delmarva HTGR (Summit) is to be submitted by the end of 1977 for GA to remain on its schedule of going critical in 1980. Depending on the results of post-irradiation examination, it seems as though little time exists for re-evaluation of fuel if another design change is made. At the present time, we would conclude that a fuel design change would require further experimental verification.

A change in the reference design is a realistic possibility. General Atomic is presently working on (or sponsoring) the use of dopants such as Al or Si. Small additions of these dopants and oxygen are believed to form compounds with some of the fission products, reducing their mobility and subsequent release rates. If this work proves encouraging, GA may incorporate these dopants in the "reference" design. A decision will then have to be made as to the extent to which this new fuel will have to be tested. Another design change being considered is to replace the fuel stick concept with a technique whereby the "green" particles/matrix mixture will be placed in the fuel holes in the hexagonal block, and the entire block will be subjected to the curing operation. Capsules containing fuel fabricated in this manner are presently being irradiated and will be examined in time for a decision on the Delmarva HTGR. Post-irradiation examination of these capsules will be carefully reviewed.

Fission products can also be released to the primary coolant by fission of uranium or thorium present as contamination outside of the particle coatings and by release from defective particles (cracked or missing coatings). This means of release is controlled by fuel specifications and quality controls that limit the amount of contamination and failures allowed in as-manufactured fuel. The possibility that off-specification fuel could be present requires the formulation of a sound quality control program for fuel manufacturing. A continuing review of this program and the fuel product specifications associated with fuel quality assurance measures should be performed. An experimental and

analytical capability should be developed to permit the independent evaluation of quality control and assurance procedures, and assessment of the impact of changes in the reference design fuel. The fuel product specification documents of the present reference design fuel and any proposed changes in the fuel design should be acquired and reviewed.

2. Primary Coolant Impurities (Preliminary Evaluation)

The role primary coolant impurities play is varied and will influence many aspects of the HTGR system. During normal operation of the reactor, helium with low levels of impurities will be present and any degradation of materials should be slow. However, some effects will occur over extended periods of time making predictions difficult. These effects include the transport of fission products both within the core graphite itself and to various surfaces within the primary system. The low level of impurities normally expected may lead to degradation of structural components such as the graphite support block and support posts. Failure of these components may lead to various types of accidents. The graphite to be used for construction of these components has not been clearly defined or specified. Therefore, it is not at present possible to evaluate some of the problems which may arise. The potential problems include catalytic oxidation of the graphite due to impurities present leading to localized areas of graphite removal. This, in turn can lead to component failure with only fractional amounts of material removal. The problem requires determination of the effects of possible slow degradation of the support graphite as

a consequence of localized attack.

The catalytic effects of contaminants within the graphite on the reaction of oxidizing impurities with graphite is not clearly defined in terms of the size, shape, distribution and composition of the contaminants. The method of testing for these effects will depend on impurity levels and distribution in the graphite, and also will depend upon the coolant impurity levels which lie in a critical range in considering the effects of ions.

The level of impurities in the primary coolant may also lead to oxidation and carburization of metallic components in the system. Normally with decreasing concentrations of reactants in the primary system it may be expected that lower rates of attack of the metal will take place. However, these rates are highly dependent on the properties of oxidation products or layers on the metal surfaces which can change markedly with different levels of impurities and result in a marked increase in attack at low levels of impurities. This behavior has been observed.

The composition of the gas will determine the possible carburization of metallic components of the primary system. The carburization by the gas will be complex and depend on impurities in the system such as CH_4 and other organic molecules produced within the core or from degradation or leakage of oils and lubricants. Other effects might be attributable to the variations in the oxides found on metal surfaces which may also protect against carburization. The oxide formation and "annealing" on the metal surface will depend on the variations in the oxidation potential of the

primary coolant.

The formation of CH_4 and other species can occur radiolytically within the core and other impurities such as N_2 may also enter into reactions and produce unusual products, e.g. N_2 may enter into reactions with graphite, CH_4 , CO , O_2 , H_2O and H_2 to form compounds such as NO , NO_2 and HCN . The safety aspects of these compounds (some of which are poisonous) in a high temperature radiation environment and under accident conditions are being considered. An additional effect of these impurities could lead to mass transfer of carbon with the formation of fine particulate matter. The particles could be expected to form in the cooler areas of the reactor where fission products will also deposit. Over extended periods of time the particles with the deposits of fission products could build up and may only be released following a depressurization accident. Rapid grain boundary attack of metal components as a result of penetrations of impurities, e.g., S, Te, may lead to embrittlement at the surface or stress cracking.

An additional area of major importance is the effect of impurity levels on transport of fission products in both the core graphite and the primary system. Once again the levels of impurities may play a major role which has not been investigated to any great degree. Impurities penetrating the core graphite may reach chemical equilibrium with the graphite but these impurities can react with fission products in the core graphite and influence their rates of migration and release to the primary coolant. The influence of these effects over extended periods has not

been considered in any great detail and could lead to significant differences in fission product release under various accidental conditions. These effects of impurities will also be operative following any large steam or air leakage into the He.

The analysis of these problems is complicated by the lack of understanding of the behavior of impurities in the primary system and difficulties in monitoring impurity levels accurately. Both factors are important, as the composition measured in reactors or loops do not relate to any clear reaction sequences. The causes of the discrepancies require clarification as they influence primary coolant compositions and the effects of these compositions on other aspects of the reactor and its safety.

3. Rapid Graphite Oxidation

Accidents causing "rapid" graphite oxidation in an HTGR will involve air ingress following a depressurization accident and steam ingress following steam generator tube leaks. The steam-carbon reaction is endothermic and, although the likelihood of large quantities of steam entering the reactor core is greater than that for air, the reaction is not self-sustaining. Considerable damage can be caused by large steam ingress accidents: (1) the fuel can be hydrolyzed, (2) the graphite fuel blocks, support blocks and support posts can be oxidized, (3) the fission products can be redistributed, probably by formation of more volatile species, and (4) metal components can be oxidized. Probably the most significant safety related consequence involving steam ingress would be the de-

terioration of the graphite core support posts.

OXIDE3 predicts that following postulated large steam-ingress accidents, a large quantity of the water will make its way to the support posts. It has been shown that preferential oxidation of graphite by water is catalyzed by metallic impurities (worm-holing). Preferential deterioration of the binder in some graphite has also been observed. The main emphasis regarding the core support posts noted in the Fulton PSAR pertains to their stress integrity. It is believed that an averaging of properties here could be misleading and that the safety aspects of the core support posts are not transparently evident. It should be experimentally demonstrated that the reference core support post material can sustain a hypothetical number of steam transients (including others such as high temperature) without losing its mechanical integrity. Locations of interest are the upper and lower post seats, where deterioration could lead to binding which in turn would increase the bending stress applied to the post.

The aspect of permanency of the posts (40-year lifetime) under various planned and unanticipated transients is here limited to any specific grouping of three. Three such columns are utilized in the support of a refueling region and the possibility and consequences of the failure of one, two or three of these supports are being considered. At present, the letter of H. C. Applegate to J. G. Y. Chow of January 13, 1975 indicates the core support graphite has not yet been expressly specified. The main material properties specified (Spec. No. 900161) deal with room

temperature strength, graphitization, maximum sulfur content, and a documented history of post production.

While failures are at present considered hypothetically, it is expected that when one column does fail others will follow. A study of the origins of failure, in spite of assurances, should therefore attempt to minimize consequences. For example, where a particular undetected faulty furnace lot is a fault cause, the effect of randomizing the location of such a lot in a core should be studied along with the more serious effect of localizing three such faulty blocks under a single fuel region.

An air ingress accident, although less likely to occur, appears to be individually far more damaging than a steam ingress accident since the air-graphite reaction is exothermic and self-sustaining under certain circumstances. Small amounts of air may be present in the reactor due to any of the following: (1) inadequate outgassing following re-fueling, (2) accidental ingress from support systems, i.e. purification system, and (3) minor breaching of the containment. Large air ingress rates apparently only are postulated following a depressurization accident. We are considering whether such an accident may lead to a graphite fire.

Both air and steam ingress accidents could lead to the formation of explosive gas mixtures (H_2 , CO and air) in the reactor or containment building. This complex problem will be studied under a proposed contract with Penn State.

OXIDE3 has been submitted as GA-LTR-7 and has been developed to analyze the effects of a chemical reaction with the core of an oxidant

inleakage into the primary coolant in a specific transient. While giving a general idea of the consequences of an inleakage, it is recognized that the program input averages the core. Various physical and chemical properties of the core graphite have been implied from measurements on various grades of graphite. It is therefore expected that such results might introduce some amount of error when applied to non-linear systems. The effects of these deviations can be determined through a parameter sensitivity analysis of the code. We are considering a future version of the code which would incorporate a Monte Carlo approach of property definition. For the present instance, provided that the results of a transient calculation are conservative in their safety margins, non-linearity of the system would probably not be safety limiting.

Due to the importance of the possible oxidative attack upon the core support posts it is necessary to know if there is steam in the lower plenum. OXIDE3 does not concern itself directly with safety of the core support posts although it can be used to predict the atmospheric conditions present in the lower plenum resulting from various transients. Due to the importance of this calculation, the results of the OXIDE3 program should be verified by a separate and independent calculation. In this regard a literature review on steam-graphite oxidation is being conducted with a specific purpose. Specifically, the survey will include both the catalytic effects of impurities incumbent with local oxidation as well as the uniform rate equations for the graphite-oxygen reaction. The literature data are being tabulated to determine the spread of the existing experi-

mental oxidation rates and to determine the validity of the existing rate equations. The parameters being considered are rate, temperature, graphite type, impurities, density, pore size, BET surface area. The concentrations of H_2 , CO , H_2O and CO_2 in the environment are also being considered as variables.

4. Structural Evaluation (Preliminary Evaluation)

The interdisciplinary group effort in this task area is involved with the structural dynamics of the sequences following the failure of one or more core support posts. At this time we have not reached any decisions relevant to this group effort. The specific problems covered in this task area are reported in a following section.

5. Materials (Preliminary Evaluation)

The capability of the structural materials to maintain their properties and integrity throughout the reactor design life of normal operating and designed transient conditions is an absolute requirement. However, in safety analysis it would be valid to examine the behavior of the materials under extremely severe conditions when the reactor undergoes multiple failure accidents and the reactor may be in the faulted condition. Therefore, in this program, the review and assessment of material properties and behavior will encompass the entire range of normal operating conditions to the extreme severe conditions that could be conjectured by multiple failure accident scenarios.

5.1 Metal Components

Preliminary state of the art review of metal components used

in the primary coolant circuit of the HTGR indicated a serious lack of long term experience and insufficient base technology data on fatigue and creep to extrapolate to 40 years of service life. It has been identified that high cycle fatigue data are not available for safety analysis of some of the metal components. Failures due to high cycle fatigue originate at the surface of a metal component. In the impure helium of the HTGR environment, it has been shown^(1,2) that preferential grain boundary attack of the metals can take place. This type of attack causes surface defects where fatigue failures can initiate. Safety and failure analysis of vibrational stresses caused by the high helium flow velocity, rotating equipment and other cyclic stresses requires high cycle fatigue data to complete an evaluation. Our preliminary evaluation indicates that the generation of this type of data is not included in any of the presently proposed RRD basic technology programs.

Review of GA-LTR-1 - An Analysis of HTGR Core Cooling Capability and GA-LTR-5-Vol. III - Large HTGR Steam Generator Design Verification and Support Programs-Materials, has been underway. As other pertinent LTR's become available they will be included in the review and assessment of these documents. Review so far has been preliminary and no firm statements can be made at this time.

Review of GA-LTR-1 is coordinated with the heat transfer and structural evaluation group to evaluate the effect of the metal components reaching the critical safety limit. Numerous discussions have been held to determine the possible effects of multiple failure accidents on the

critical safety of metal components such as the flow control orifice at the inlet plenum, the PCRV metal liner, and the metal plates of the thermal barriers. However, no conclusions have been drawn from these preliminary discussions.

Review of GA-LTR-5-Vol. III indicates insufficient data and experience for Incoloy 800 and some of the nickel base alloys proposed for use in the steam generator circuit. Even for a very widely used alloy such as 2½Cr-1Mo ferritic steel, the effect of fatigue and creep interaction needs to be studied further. Incoloy 800 is extensively used and data from many more heats of this alloy should be studied. In view of the wide variation of properties from heat to heat found in Type 304⁽³⁾ stainless steel, similar studies should be made on Incoloy 800 which is also a stainless steel. Also, more long term data of testing times of greater than 10,000 hours and preferably some testing times up to 35,000 hours are needed to gain confidence on the behavior of this material. Furthermore, tests should be run in impure helium similar in composition to that of large HTGR's. It is noted that some of these data are being generated under the basic technology program of RRD for HTGR's.

In answer to inquiries, the ASME Boiler and Pressure Vessel Code, Sec. III through Code Cases 1331 and 1592, has set rules and guidelines for elevated temperature design and construction. Although we are not members of the ASME Committee we have attended meetings of the Task Groups and Working Groups deliberating the recommended procedures for elevated temperature design. It appears from the attendance at these

meetings that a great deal needs to be established in this area and much needs to be learned about fatigue-creep interaction. We will continue to attend these ASME Committee meetings to keep abreast of materials qualified in Code 1592 and stress limits set on the materials.

Our review of the steam generator construction indicates that two areas should be analyzed from the standpoint of safety. These two areas are:

(1) Mechanical and corrosion properties of the transition weld between the Incoloy 800 stainless steel and $2\frac{1}{2}$ Cr-1Mo ferritic steel in the steam generator coil.

(2) The material and fabrication specifications of the $2\frac{1}{2}$ Cr-1Mo steam generator support flange.

We requested information for these two areas in our letter of May 17, 1974 to GA. To date, we have received reports on corrosion inspection of a loop containing the transition weld but no mechanical test data. With regard to area (2) on the steam generator flange, we were told that specifications are not yet available. Without this information it is obvious that safety evaluation of these areas will be delayed.

5.2 Graphite Components

Various types of graphites are used in the core and core support structure. H-451 grade, a Great Lakes Carbon Corporation near isotropic graphite appears to be a prime candidate for the core graphite. However, other graphite manufacturing companies have grades similar to H-451 and these other graphites may be selected by GA. Similar vagueness prevails over

the use of other types of graphite used in the HTGR, such as core support block and core support posts. For the core support post, Union Carbide Company ATJ graphite, a molded graphite, was used in Fort St. Vrain. However, we were informed (letter of Howard C. Applegate, January 13, 1975) that ATJ is "not expressly specified for the core support posts". Several other types of graphite were mentioned as possible candidates including extruded type graphites. Because of the lack of a reference graphite material for the various applications, we are required to test several types of graphites.

Included in the above reference letter to us from GA was GA Specification R0161 9C0161, Issue B, "Core Support Floor and Gas Seal", dated October 30, 1974. This specification gives the requirements for the graphite materials of the support floor, support posts, seats, and peripheral gas seal to be used in a HTGR. This document specifies mechanical properties, graphitization firing temperature and sulfur content. However, it does not contain limits on impurities such as iron and other elements that might catalyze the oxidation of the graphite. Therefore, severe localized oxidation and perhaps weakening of the core support structure cannot be precluded in the safety evaluation of this area of the HTGR.

5.3 Prestressed Concrete Reactor Vessel (PCRV)

An assessment of concrete materials technology applicable to the PCRV is being performed. The purpose of the program is to determine if there is a sufficient body of reliable concrete property data available for

use in the design and safety evaluation of PCRV's. Gaps in the technology will be identified for future experimental programs.

There are at least three conditions present in PCRV's that make them unique when compared to the conditions present in other massive concrete structures. These are: (1) elevated temperature, (2) large temperature gradients, and (3) the presence of large amounts of radiation.

The normal operating temperature for most PCRV's is not very high. For example, a temperature of 130°F (54°C) is specified for the Fort St. Vrain reactor and the French EDF4B reactor⁽⁴⁾ operates at 167°F (75°C). The reason for possible concern is that even these relatively low temperatures can have an appreciable effect on the properties of concrete. This effect arises because of the presence of relatively labile water intimately involved in the hydrated cement structure. The amount of this water in the cement structure (portland cement hydrates) is a function of the ambient humidity and exerts a strong influence on properties such as drying shrinkage, irreversible creep, reversible creep, and length changes associated with humidity changes. These hydrates cannot be regarded as inert; their chemical stability vis-a-vis water plays a controlling role in many phenomena⁽⁵⁾.

There have been two excellent reviews on the application of prestressed concrete to reactor pressure vessels^(6,7). These reviews were published in 1964 and in 1968 and they summarize very concisely the state of the art of concrete technology as applied to PCRV's. The later review is more detailed than the 1964 study and contrasts the designs of a number of

existing PCRV's. Both reviews emphasize design limitations of concrete, and areas where further work is needed. Tan⁽⁷⁾ summarizes the position of PCRV designers: "Because of the lack of information on long-term behavior of concrete under high temperature and irradiation, designers of PCRV's have limited the maximum temperature, temperature drop, and radiation dosage to within the limits known to have little effect on the overall properties of concrete."

Several areas relating to concrete properties for which little information was available were listed by Browne⁽⁸⁾ as follows:

(1) The effect of temperature on sustained failure of concrete in a multiaxial stress state.

(2) The effect of moisture migration, accentuated by temperature, on mass concrete creep.

(3) The correlation of strains, stresses, and moisture migration within the vessel structure compared with predicted values in design model testing.

(4) The influence of coolant leakage on concrete properties.

(5) The influence of present and possibly future higher levels of radiation on all concrete properties.

It is not only for purposes of vessel design that the properties of concrete at elevated temperatures must be known. It is also required for a valid safety analysis that the behavior of PCRV concrete to design temperatures and to temperatures in considerable excess of design be known. The purpose of these data would be to assess the effects of

highly improbable accidents, such as a total loss of coolant or failure of the pressure relief systems.

Quite detailed discussions of the concrete design properties, their application, the various assumptions made, and the limitations involved are presented by Tan⁽⁷⁾ and in the Fort St. Vrain Reactor FSAR Section 5.4⁽⁹⁾ and Appendix E⁽¹⁰⁾. The Fort St. Vrain report also describes in detail the experimental program involved in developing a suitable PCRV concrete and the evaluation of its properties as functions of hydration, design temperature, and time (reactor lifetime).

It is doubtful whether a single concrete can be considered typical of the types used for PCRV's. However, one general characteristic of concretes specified for a number of reactors is that they are all relatively high strength compositions⁽⁴⁾. It is difficult to directly compare the different concretes because testing is done at different degrees of hydration, different age, and on specimens of different size and shape. The type of cement used is also different. It is possible by various empirical interpolations and extrapolations to "adjust" the various concrete strengths to that of a standard condition so that they may be compared to each other. The standard conditions chosen are a 4-in-diam x 8-in-long cylinder, cured in the sealed condition for 28 days. It is assumed that a cement like Type II portland cement is used and that the curing conditions are similar. Generally the cylinder strength is ~80% of the cube strength⁽¹¹⁾. The strength vs. cure time relationship for a concrete made from Type II cement can vary considerably. The curves

developed in the Fort St. Vrain program are used (Figure E-16.2). In Table 1 the specified concrete strength for a number of PCRV's is shown⁽⁴⁾ along with the normalized specified strength. No normalized specified (28 day) strength is under 5000 psi. The lowest normalized strength is that for Fort St. Vrain concrete (5200 psi); the highest is for the Bugey I vessel (7350 psi).

In actual practice, the concrete mixes produce much higher strength concrete so that the specified strength is actually a minimum strength. For example, for the Fort St. Vrain concrete, in order to insure that all concrete is greater than the specified design strength, the required average strength must be 7200 psi. For the Bugey I pressure vessel the same conditions require the average strength to be 8500 psi normalized⁽¹²⁾. The lower strength requirement for the Fort St. Vrain reactor concrete is probably a more conservative design than the French concrete in that the lower strength requirements should be easier to achieve.

The control of the strength of concrete during a project is based on testing cylinders cast during the course of construction. The achievement of the desired design strength is based on the results of testing these cylinders, not directly on the evaluation of the concrete in the structure itself. The Bureau of Reclamation in their work on dam construction has compared 28 day cylinder strength to actual cores of various sizes and ages⁽¹³⁾. The cores, which varied in size from 6 inches in diameter to 22 inches in diameter were tested at ages of 3 months to

Table 1
A Comparison of Specified Concrete Strengths for
Various Concrete Pressure Vessels

<u>Reactor</u>	<u>Specified Concrete Strength, psi^(a)</u>	<u>Normalized Concrete Strength, psi^(b)</u>
Fort St. Vrain	6000 at prestress	5200
Oldbury	4700 at 28 days	5875
Wylfa	4700 at 56 days	5400
EDF4E	5600 at 90 days	6150
Dungeness B	4700 at 28 days	5875
Bugey I	5880 at 28 days	7350
Fessenheim	5400 at 28 days	6750
Fulton	6510 at 60 days	6175

(a) Fort St. Vrain FSAR, Vol. IV, Appendix E, Table E.1-1.

(b) The normalized strength is the adjusted strength for a 4-in-diam x 8-in long cylinder at 28 days, sealed during curing.

10 years. For the conditions of interest in PCRV's, the strength at time of prestressing (average age of 6 to 12 months), the strengths of the most massive cores, 22 inches in diameter, were from 40 to >100% greater than the 28 day cylinder strength. Even when the difference in concrete age is accounted for, (the 28 day cylinder strength is generally 80 to 85% of the 180 day cylinder strength), the core strengths appear greater than the cylinder strengths. This occurs even though the concrete in dams contains 6 inches in diameter aggregate. Figure 202 in Reference 12 suggests that for cast cylinders the compressive strength of a 4-in-diam x 8-in-long specimen should be ~20% greater than the strength of a 22-in-diam cylinder when the maximum aggregate is 1/4 the diameter or less. Based on this, since the aggregate size of PCRV concrete is a maximum 1.5 inches (as compared to 6 inches for dams) and since the strength increases with decreasing aggregate size⁽¹⁴⁾, it is possible that the compressive strength of massive PCRV concrete can be greater than that indicated by the test cylinders. No information has been found on the results of testing cores from actual PCRV vessels or vessel models. However, it does appear that the use of test cylinders to monitor the strength of PCRV concrete is a valid and probably conservative estimate.

At the present stage of this study it would appear premature to make any recommendations, and the opinions expressed here so far are subject to change. Subsequent progress reports will review studies in other areas of PCRV concrete technology. At appropriate times topical reports will be issued.

5.4 Other Materials

Two other materials are of concern in the safety analysis of the reference design HTGR: core control rod materials and the thermal barrier.

The review and assessment of both of these materials have been initiated. The behavior of these materials when the safety limit has exceeded during a multiple accident are being evaluated. Considerably more study and evaluation will be required before any conclusions can be drawn on the effect of melting of the steel container for the borinated graphite control and migration of the boron poison out of the core. Likewise considerably more work will be required to gain realistic evaluation of the consequence of the thermal barrier failure in the outlet plenum causing possible flow blockage and perhaps failure of the PCRV liner and overheat of the PCRV concrete. However, eventually these effects will be evaluated from the safety standpoint by the HTGR Safety Evaluation Group.

6. Safety Instrumentation and Control Systems (Preliminary Evaluation)

Specific task areas in this sector include coolant impurity monitoring, failed fuel detection and location, thermometry, PCRV surveillance, noise analysis and dynamic testing, data acquisition and processing, neutron detectors, and post accident monitoring.

The temperature range of HTGR's is higher than that found in water reactors, the primary coolant is novel and has markedly different properties from the water of conventional reactors, and the kinetics time scale of HTGR's is remote from that typical of water reactors. It is not

surprising that instrumentation and controls concepts for HTGR's cover new ground. Gaining a secure depth of experience in such a broad field is difficult with a limited time resource.

The program at BNL has two initial foci. Impurity monitoring on a broad range is being addressed so that a wide variety of contaminants can be detected at minimal concentrations. Post accident monitoring is the second focal area. Since this task includes most of the others it has a useful breadth, and a perspective which is even broader.

Coolant contaminant monitoring studies are planned for Fort St. Vrain. Chemical and radioactive contaminants will be monitored to assay the reactor coolant instrumentation. This offers an unexcelled opportunity for RSR to obtain practical experience with an operational HTGR. In support of the Fort St. Vrain studies a small scale coolant test loop will be assembled at BNL.

7. Accident Delineation (Preliminary Evaluation)

Ideally the goal of accident delineation would be to identify all accident sequences that are consistent with physical laws, have significant likelihoods, and would have public safety implication. In principle, the accident delineator could start considering a set of events with public safety implications, for example: radioactive releases, and then work backwards on the basis of consistency with physical laws to describe a set of event sequences/accident sequences leading to those events. Alternatively, the accident delineator could start from a given event, for example, a pipe break, and then work forward on the basis of consistency

with physical laws to describe again a set of sequences of events leading to events with public safety implications. However, both methods, the strictly forward and the strictly backward, lead to unmanageable and unnecessary complications. In practice, using a considerable amount of insight, and trial and error, it is possible to identify sets of similar events. For example, small pipe breaks in the primary system, from which the accident delineator can work both forward, in consequence chains (event trees) and backward in cause chains (fault trees). This is roughly the strategy followed in the Rassmussen study.

The effort on accident delineation to date has been directed to formulate event trees starting from the depressurization event, the core support failure and the complete loss of coolant event. No attempt is made at this time to assess the probabilities of these initiating events. The selection was based on high consequence considerations and is of course provisional. It is expected that further analysis of the events will modify and expand the initial choice.

A tentative functional event tree for the depressurization event used in the DBDA has been formulated following the functions identified in the vendor's analysis. For this event tree, the vendor has developed analytical models to simulate the system behavior on the upper path. Work is now in progress to identify and procure, if available, models to describe the behavior of the system in the lower branches of the event tree.

The delineation of accidents starting from failure of parts of the core support structure has been initiated. The effort at this time centers

on identification of failure modes and prediction of potential motions of the core.

Preliminary work on the loss of cooling initiating event has led to the identification of the need for models to represent the heating of the core and the migration of fission products under no flow and high temperature conditions.

SPECIFIC ACTIVITIES

1. Fission Product Release and Transport

1.2 Fission Product Distribution (Core) (H. S. Isaacs, S. Aronson, R. G. Behrens)

The movement of fission products from the fuel to the helium coolant depends on the release from the fuel, diffusion in or through failed fuel coatings and across the fuel element graphite and the core graphite. An analysis of these transport rates has begun with a literature survey which is being made on cesium.

Fission products are generally considered in groups depending on their individual characteristics. Non-volatile fission products which are held within the fuel particle include zirconium, niobium and rare earths and are not usually considered to migrate from the fuel, but may move when extreme temperatures are considered. Some consideration has also been given to the effects of extreme temperature on the fuel and fission product behavior and further work will be required to detail the products normally considered to be non-volatile and immobile. The rare earth metals may also be considered in terms of their movement within the fuel and initiating failure of the fuel coating. These products may, however, be found in the fuel graphite as a consequence of manufacturing contamination. Other products which may require investigation include palladium, rhodium and ruthenium.

The mobile metallic fission products include strontium, barium, rubidium and cesium. Silver is usually included in this group although

little is known about its behavior. (Recent evidence found in the European HTR program indicates that Ag may pass through the SiC coating at temperatures in excess of about 1250°C). Judging from the vapor pressure of tellurium and from its relatively high concentrations, this fission product may also be included in this group. The movement of this element has also been emphasized as it may be similar to sulfur and produce embrittlement of nickel base alloys. These metals are considered to migrate relatively rapidly in graphite although the modes of diffusion are not known. The mode of transport includes bulk, grain boundary, surface and in-pore diffusion, the latter possibly being accelerated by helium movement in the pores in parallel with the coolant stream. This process will also be influenced by impurity concentrations in the coolant helium and may in addition change surface diffusion rates. The effects of common element concentration on the diffusion rates have been suggested and the effects of concentration have been observed. However, further clarification of the effects would be advantageous in extrapolation of these results. Sorption of these metallic fission products plays an important role in reducing the amount of transport to the coolant but also influences diffusion rates because of the dependence of diffusion rates on concentrations. Irradiation has also been said to influence the mechanisms of cesium diffusion but because of the wide scatter of results the reasons for this behavior are not clear. At this stage of the investigation it appears that extrapolations of the results for release of metallic fission products under accident conditions will require clarification.

The most mobile fission products are those that are volatile at operational temperatures. This includes the fission products iodine and the inert gases xenon and krypton. The primary barrier to these elements is the fuel coatings but once these have failed the migration through the porous graphite and into the helium is rapid.

Once the fission products reach the primary coolant/graphite interface the subsequent behavior depends on the amounts entering the helium and the quantities and positions of their deposition and buildup in the primary or purification systems.

1.3 Fission Product Distribution in Coolant Circuit (H. S. Isaacs)

The analysis of the distribution of fission products in the primary circuit is necessary because (1) of difficulties in maintenance of the reactor components when contaminated with radioactivity and (2) these fission products can be the first to leave the reactor containment in the event of an accident.

Little emphasis has been given to this important problem during the report period. However, it is apparent that there may be many areas where clarification of the adsorption characteristics of most fission products will assist in defining their behavior. Fluctuating levels of oxidizing primary coolant impurity concentration and composition will influence the nature of metal surfaces and could possibly change deposition and lift-off as the metal surfaces change. At present little information is available relating absorption and surface characteristics or changes of metal surfaces with alternating relatively oxidizing and reducing nature

of the helium.

Changes in impurity levels may also give rise to lift-off and particulate formation or aerosols. Many of the species involved in the transport mechanisms should also be determined. The lift-off characteristics under various accident sequences will play an important part in determining the quantities of fission products released.

1.4 Transport of Metallic Fission Products from HTGR Graphite
(S. Aronson, R. G. Behrens, H. S. Isaacs)

The distribution of fission products in the primary system must be known prior to any accident to predict the consequence of the accident. Cesium is the most mobile of the high-yield, long-lived isotopes and little is known about its behavior as a function of the normal and off-normal limits of impurities in the primary coolant.

Experiments will concentrate on adsorption and diffusion of cesium in the moderator graphite initially under high purity conditions and followed by measurements relating to impurity ingress.

Adsorption isotherms will be measured by a Knudsen effusion, radiotracer technique. The experimental apparatus for these studies will consist of a Varian all-metal, high-vacuum system with 200 liters/sec of ion pumping plus titanium sublimation pumping. A water-cooled, stainless-steel housing in which the experiments will be conducted is to be appended to the vacuum system. Cesium-graphite containing tracer quantities of ^{134}Cs or ^{137}Cs will be placed in a refractory metal Knudsen effusion cell which is to be situated within the appended housing. The crucible will

be heated to temperature (900-1400°C) by radiation from within the vacuum chamber. The furnace and Knudsen cell assembly are to be mounted on a removable flange for easy access to the cell and furnace assembly. The temperature of the effusion cell will be measured with a calibrated platinum-platinum, rhodium thermocouple.

The rate of cesium loss from the effusion cell will be measured by counting the gamma spectrum of isotopic cesium within the effusion cell using a Bicron sodium iodide, thallium-activated scintillation detector housed in a lead pig and associated counting electronics consisting of Canberra Industries 802-1 preamplifier, 816 spectroscopy amplifier, 830A single-channel analyzer, 1775 nuclear counter, and a scanner-printer.

The counting electronics has arrived, has been tested, and appears to be working satisfactorily. The scintillation detector will soon be tested using a standard cesium gamma source.

The Varian vacuum system has been ordered and is to arrive soon. The water-cooled housing has been designed and materials for its construction will soon be ordered. Thought has been given to the furnace design and we are currently awaiting information on appropriate low-voltage, high-current power supplies before we can proceed with design and construction.

We have started looking at the SORS codes in some detail in an attempt to perform independent calculations of metallic fission product release rates and fractional releases using the models and data presented in the SORS document GA-A12462 (GA-LTR-10) and in GA-A12634 (J. H. Norman, "Review of Vapor Pressures and Diffusion Coefficients of Certain HTGR Core

Materials and Fission Products for Use in Reactor Accident Calculations").

Other activities during this report period include compilation of a bibliography of pertinent fission product literature, and a literature search for articles on the thermodynamic properties of cesium, cesium oxide, cesium iodide, and tellurium, its oxides and halides.

The data base and models used for fission product release calculations in the SORS code, the vapor pressure and the diffusion coefficient data for metals in graphite environments have been reviewed.

The release of high-boiling fission products by vaporization has been adequately handled in SORS. Detailed reports of experimental procedure are not available to allow close and careful scrutiny of Freundlich adsorption behavior of cesium, barium, strontium, and rubidium used in SORS. There is indication that discrepancies may exist between the mass spectrometric results (used in SORS) and results of preliminary radiotracer studies carried out at GA. There appears to be at least an order of magnitude difference between the metal vapor pressures obtained by the two techniques. The transition from Freundlich to Langmuir adsorption observed for cesium is taken into account in the SORS code. This transition is also assumed to occur for barium, strontium, and rubidium though no experimental evidence exists to indicate this for these metals. Measured Freundlich isotherms are used to calculate vapor pressures for these metals down to a metal concentration of $0.078 \text{ micro-moles/m}^2$ (observed Freundlich-Langmuir cut-off for cesium) and vapor pressures below the transition point are estimated. This estimate leads to conservative vapor pressures at low

metal concentrations. It is clear that adsorption behavior around and below the Freundlich-Langmuir transition region should be confirmed and clarified and correct the SORS code.

The effect on vapor pressure of simultaneous adsorption of metals should also be incorporated into the SORS code. The effect may be significant on calculated release rates. It is believed that this has not been done because of a paucity of experimental data and an inadequate model to describe the phenomena.

Values of metal diffusion coefficients used in SORS to calculate fission product release via a rate-limiting diffusion process are estimated. The estimates are based on the terrace-ledge-kink model of surface structure which pictures metal diffusion along the graphite surface as a part of the overall metal vaporization mechanism. The model assumes that the activation energy is roughly 50% of the metal's enthalpy of vaporization. This is true if the vaporization coefficient for the metal adsorbed on graphite is unity, that is, the activation enthalpy of vaporization is equal to the thermodynamic enthalpy of vaporization. There is no evidence to indicate that the vaporization coefficient of cesium from graphite is unity. It may be significantly less. This means that $\Delta H_{\text{vap}}^* > \Delta H_{\text{vap}}$ and implies that the diffusion coefficients and thus release rates estimated in SORS may be too high perhaps by as much as an order of magnitude.

1.5 Fuel Technology and Quality Assurance (A. J. Romano)

Visits to GA and ORNL, meetings with several European visitors, and literature reviews of documents and reports pertaining to the HTGR

reference fuel have led to the conclusion that additional information should be obtained regarding fuel quality control and assurance procedures. A review of current and future relevant data to allow for a meaningful and reliable assessment of the effects of HTGR fuel fabrication and operational behavior on safety-related consequences is being made and a topical report will be issued. To perform this task it will be necessary to acquire documents such as the Fuel Product Specifications and Fuel Process Specifications for large HTGR's.

Fission products can be released to the primary coolant by fission of uranium or thorium present as contamination outside of the particle coatings. In addition, $\approx 0.2\%$ of the fuel will have missing or defective coatings⁽¹⁵⁾. Thus, based upon the estimated total of $\approx 10^{11}$ particles in a large HTGR core, approximately 2×10^8 particles would have defective coatings resulting from the fabrication process alone. These are methods of release controlled by fuel specifications and assurances.

Steady-state and transient solutions of heat transfer problems involving normal and off-normal conditions were made using the TAC 2D computer code. The emphasis of the study was to determine the effect of thermal parameters on fuel particle coating failures. Details are presented in task area 9.4.

2. Primary Coolant Impurities

Thermodynamics of Gaseous Impurities and Their Monitoring (H. S. Isaacs)

The control, monitoring and reactions of gaseous impurities in the primary system are of major importance when both high and low concentrations are present. The immediate consequences of large water leaks may be studied in detail as the impurity excursions would be present only for short periods of time and the temperature-time-environment system can be simulated for study. The influence of low concentrations are also important but reactions and changes take place over long periods of time, making the study of the effects less accessible to experimentation and requiring either a detailed understanding or extended periods of study. The presence of low concentrations of impurities can lead to degradation of materials, e.g. by oxidation or carburization of metals, hydrolysis of fuel and pitting of structural graphite components. The impurity levels could also influence the movement of fission products, carbon transfer and the formation of particles in the primary system. These effects will depend on the concentrations of impurities present in the helium. Hence it is of primary importance in any program aimed at the understanding and prediction of behavior of any reactions in the primary system of an HTGR, that the concentrations be closely controlled and accurately measured.

At the present stage of analysis of the effects of normal levels of impurities in helium systems both in reactors and experimental loops there seems to be some analytical difficulties. The amount of hydrogen

present is usually considered to be much higher than expected but the ratio of hydrogen to water also remains low even over extended periods in reactor systems. The gas must then be considered to remain both oxidizing to the graphite and oxidation and hydrolysis of fuel can occur at all times.

Calculations

The equilibrium concentrations of reactants and reaction products H_2O , CO , CO_2 , H_2 , CH_4 following an injection of water have been calculated with various amounts of either hydrogen or oxygen present prior to the equilibration with the water injected. The calculations assume both isothermal conditions, equilibrium and the presence of excess carbon.

The results of the calculations are shown in Figure 1 plotted with the standard free energy of formation of various oxides. When 1.0 atmosphere of water was reacted the resulting variations of oxygen potential, $RT \ln(O_2)$, where O_2 is the effective partial pressure of oxygen calculated) with temperature are shown by the curve labeled 1.0 Atmos. in Figure 1. A partial pressure of 1.0 atmosphere would correspond to a concentration of 1/50 Vpm in HTGR helium at 50 atmospheres. As can be seen from this curve, at high temperatures, above about $800^\circ C$ the oxygen potential follows the curve for the standard free energy of formation of CO . This indicated, and it was seen from the calculations, that virtually all the water was converted to CO and the presence of the hydrogen and other reactants did not influence the oxygen potential as set by the CO . In contrast, at temperatures below $600^\circ C$, CO_2 predominated and the oxygen potential approached that

Figure 1

Standard free energy of formation of some oxides and the equilibrium oxygen potentials for various pressures of water reacted with carbon as a function of temperature. The "upper limit" curve relates to the maximum oxygen potential for which the curves are linear with temperature. The "C.S.Z." and "Y.D.T." curves give the lower limits of 99% ionic conductivity of calcia stabilized zirconia and yttria doped thoria respectively.

expected for CO_2 as shown. Significant amounts of water were also present at the low temperatures. When 10^{-6} atmospheres of water reacted the results showed similar trends but the predominance of CO formation occurs at 350° and above as seen on the " 10^{-6} Atmos" curve in Figure 1. At temperatures above 350° (and above 850°C in the former case) the relation between oxygen potential and temperature are linear. The limits to this relation are shown by the "upper limit" curve in Figure 1.

As mentioned above the presence of reaction product hydrogen did not change the oxygen potential from that expected if only CO was effectively present. This behavior was also seen when the hydrogen pressure prior to water injection and equilibration was increased to over 10^{-2} atmospheres (200 Vpm H_2). However, when the hydrogen concentrations were increased to 1.0 atmosphere (20,000 Vpm) there was a reduction in the oxygen potential by less than 1.5 Kcalories. When oxygen as well as water was added the calculated oxygen potentials again indicated the reactions gave the predominance of CO under conditions below the "upper limit" on Figure 1.

Applications of the Calculations to Gas Monitoring

The above calculations indicate that the oxygen potential can be used to monitor any concentrations of oxygen derived from water, air, or other sources, in a helium system if equilibrium is reached. Under these conditions the oxygen potential is readily related to the total CO concentration which in turn gives the total oxygen concentration. The system is virtually independent of hydrogen concentrations expected but any concen-

trations above an initial partial pressure of 1.0 atmosphere (20,000 Vpm H_2) would only give an error of less than 50% in oxygen concentrations at any levels even below one Vpm total oxygen.

Oxygen potentials are readily measurable employing electrochemical systems based on solid oxygen electrolytes. Two electrolyte systems that have been extensively used are calcia stabilized zirconia (CSZ) and yttrium doped thoria (YST). The electrolytes may be used over wide ranges of oxygen potentials but do suffer from limitations due to the presence of electronic conduction at low oxygen potentials. The electronic conduction reduces the sensitivity of the measurements. The oxygen potentials at which electronic conduction reaches 1% of the total conduction are shown for CSZ and YST in Figure 1. These properties indicate that the zirconia may be used only at low temperature $\sim 500^{\circ}C$ if only low oxygen potentials were to be measured, as the range of logarithmic proportionality would be limited at this temperature as may be seen from the "upper limit" line in Figure 1. In addition, the behavior response of the instrument at $500^{\circ}C$ may not be acceptable because of the high resistance of the electrochemical cell at low temperatures. The use of yttria doped thoria at $700^{\circ}C$ shows none of these limitations and will be used to monitor the total oxygen present in the experimental helium systems.

This area of investigation is also related to tasks 5 and 6.

Helium Impurities Loop (A. J. Romano)

The objective of the proposed Helium Impurities Loop (HIL) will

be to provide a simple loop facility in which various HTGR environments can be reproduced. The loop will be used as part of the experimental effort of the program.

The preliminary design has incorporated the following:

(1) the latest thoughts and experiences gained in recent visits and exchanges, especially those of Dragon representatives, (2) the fact that many of the components will be fabricated from parts and materials which are readily available on site, and (3) overall simplicity which will facilitate its construction. The loop will be used for the following purposes:

(1) to expose materials for mechanical testing at design temperatures and in the presence of impurities, (2) to evaluate impurity monitoring techniques, (3) to perform graphite oxidation studies and provide samples for subcontract work, and (4) to perform corrosion studies of metals.

The design operating conditions will be:

Maximum temperature = 900°C (1652°F)

Maximum pressure = 25 psia

Flow rate = 600 CFH

Impurity levels = as desired

A tentative schematic flow diagram of the loop is shown in Figure 2.

The design is limited to the gas circulator available which can deliver 600 CFH at 25 psia. The loop consists of three circuits. The main circuit contains a vessel with a block of graphite, several test sections, the gas circulator and a water-cooled heat exchanger. The other two circuits will

Figure 2

be used to inject impurities and purify the helium, respectively.

Components for the loop are presently being acquired and/or assembled.

4. Structural Evaluation (M. Reich, P. Bezler, J. Fraenkel, T. Y. Chang, B. Koplik, J. R. Curreri, L. Laske, C. Goodzeit)

Work on HTGR structural evaluation has concentrated on the following areas:

1. Code development
2. Graphite failure criteria development
3. Core seismic evaluation studies

In addition to the above, a general review is being carried out involving the following:

1. Identification of structural failure modes that may occur.
2. Compilation of the constitutive properties of the HTGR structural materials.

3. Incorporation of the design criteria into analysis methods for response or failure analysis.

4. Simulation of loads for normal operating and accident conditions.
5. Establishment of a structural evaluation procedure so that each of the above efforts form an integral part of the evaluation process.

4.2 Core

4.2.4 Code Development

4.2.8

A three-dimensional finite element structural computer code applicable to elastic HTGR structures is being made operational on BNL's CDC 7600 computer. The code analysis method allows for the evaluation of the interaction between one-, two- and three-dimensional elements des-

cribing a heterogeneous structure. It is uniquely suited to evaluate a PCRVR comprised of rebars, tendons, liner and a gross concrete matrix acted on by random or distributed loads. When fully operational the code will be used to investigate the PCRVR behavior under various load conditions. A future addition to this code involves the inclusion of a routine that predicts concrete crack development and growth.

An elastic-plastic-creep computer code is also being developed to evaluate composite two-dimensional heterogeneous structures. It will permit the evaluation of plane and axisymmetric HTGR vessel models under extreme loading conditions. In final form the code will contain provisions to predict crack development and growth. When completed, the above methods will enable the assessment of steel creep relaxation effects, concrete cracking effects, and, in general, delineate the effects of accidental overloading of the PCRVR.

In addition, Non-Sap and Sap IV have been made operational on BNL's 7600 CDC machine.

Finally, an 1160-MW version of the GA's POKE code was made operational on BNL's 7600 CDC computer. Results of a sample problem run at BNL were identical to those supplied by GA.

4.2.1 Graphite Failure Criteria Development

4.2.3

In order to evaluate the structural performance of the HTGR core and core support components, strength criteria for different types of HTGR graphite were investigated. This work can be summarized as follows:

1. General Description

Graphite, in general, is an anisotropic, heterogeneous, and brittle material exhibiting moderate non-linearity in its stress-strain response. Like most brittle materials, graphite has a much higher compressive strength than tensile strength, and its stress-strain behavior can be altered rather significantly by the presence of hydrostatic stress. Essentially there are two types of graphites due to differences in the forming process, i.e. extruded and molded graphites. Both of the types described above are used in the HTGR designs.

The anisotropy of graphite is caused primarily by the forming process. In forming, the graphite particles tend to take a preferred orientation, either parallel to the direction of extrusion or perpendicular to the direction of molding. The direction of the preferred orientation is referred to as the "with-grain" direction, whereas the direction normal to the with-grain direction is referred to as the "against-grain" direction. Other factors affecting anisotropy of graphite are the inherent anisotropy of the graphite crystals, grain sizes, pore distributions, etc. As a result, there is an apparent difference in the material properties measured in the with-grain and against-grain directions. For practical purposes it is sufficiently accurate to consider graphite as a transversely isotropic material. The material axes are defined, as shown in Figure 3, such that the axis of rotational symmetry is parallel to the with-grain direction for extruded graphite and is normal to the with-grain direction

Extruded Graphite

Molded Graphite

Coordinate Systems for the Extruded and Molded Graphites

Figure 3

for molded graphite.

2. Strength Criterion

With respect to the material principal axis, the strength criterion for transversely isotropic graphites may be expressed as a function of the stress components,

$$\begin{aligned} &F_1(\sigma_1 + \sigma_2) + F_3\sigma_3 + F_{11}(\sigma_1^2 + \sigma_2^2 + 2\sigma_6^2) \\ &F_{33}\sigma_3^2 + F_{44}(\sigma_4^2 + \sigma_5^2) + 2F_{12}(\sigma_1\sigma_2 - \sigma_6^2) \\ &+ 2F_{13}(\sigma_1\sigma_3 + \sigma_2\sigma_3) = 1 \end{aligned} \quad (1)$$

where $\sigma_1, \sigma_2, \dots, \sigma_6$ are the contracted notations of the multi-axial stress components, and $F_1, F_3, F_{11}, \dots, F_{13}$ are the strength constants to be characterized from experiments results.

With respect to isotropic graphite currently being considered for the core structure of large HTGR systems, the strength criterion (i.e. Equation 1) simplifies to,

$$\begin{aligned} &F_1(\sigma_1 + \sigma_2 + \sigma_3) + F_{11}(\sigma_1^2 + \sigma_2^2 + \sigma_3^2) \\ &+ 2(F_{11} - F_{12})(\sigma_4^2 + \sigma_5^2 + \sigma_6^2) \\ &+ 2F_{12}(\sigma_1\sigma_2 + \sigma_2\sigma_3 + \sigma_3\sigma_1) = 1, \end{aligned} \quad (2)$$

an equation with only 3 independent constants, F_1, F_{11} , and F_{12} .

3. Experimental Characterization

As mentioned the strength constants are obtained by experimental means. The methods applied involve uniaxial, biaxial, tri-axial and torsion test specimens.

a. Uniaxial Test

In view of the anisotropic property of graphite,

two types of specimens should be prepared for determining the uniaxial tensile and compressive strengths. One is that the loading axis of the specimen is parallel to the with-grain direction of graphite and another type is that its loading axis is perpendicular to the with-grain direction.

Referring to coordinate system defined in Figure 3, let

R = Tensile strength of graphite along the X_1 -direction.

R' = Compressive strength of graphite along the X_1 -direction.

Z = Tensile strength of graphite along the X_3 -direction.

Z' = Compressive strength of graphite along the X_3 -direction.

Then from Equation (1), one can find

$$\begin{aligned} F_1 &= \frac{1}{R} - \frac{1}{R'} \\ F_3 &= \frac{1}{Z} - \frac{1}{Z'} \\ F_{11} &= \frac{1}{RR'} \\ F_{33} &= \frac{1}{ZZ'} \end{aligned} \tag{3}$$

b. Biaxial Test

Thin walled cylindrical specimens are used to determine the constant F_{13} . The specimen is prepared in such a way that the axis of the cylinder coincides with the axis of material symmetry. The cylinder is subjected to simultaneous action of internal pressure and axial force.

Let

$$\begin{aligned} \sigma_3 &= \sigma_B \text{ (Axial stress in the cylinder)} \\ \sigma_2 &= \alpha \sigma_B \text{ (Circumferential stress in the cylinder)} \end{aligned} \tag{4}$$

where α is a proportional constant; and all other stress components vanish

$$F_{13} = \frac{1}{2\alpha \sigma_B^2} [1 - \sigma_B (\alpha F_1 + F_3) - \alpha_B^2 (\alpha^2 F_{11} + F_{33})] \quad (5)$$

in which, the constants F_1 , F_3 , F_{11} and F_{33} have been found from Equation (3).

c. Triaxial Test

Specimens with solid cylindrical configuration and its axis parallel to the axis of rotational symmetry of graphite are tested by a state of triaxial compressive stresses. The imposed stress state is defined by

$$\begin{aligned} \sigma_3 &= \sigma_T \\ \sigma_1 &= \sigma_2 = -\beta \sigma_T \\ \beta &= \text{a proportional constant} \end{aligned} \quad (6)$$

Substituting this into Equation (1), one can determine the constant F_{12} by

$$F_{12} = \frac{1}{2\beta^2 \sigma_T^2} [1 + \sigma_T (2\beta F_1 + F_3) - 2\beta^2 \sigma_T^2 (F_{11} + \frac{2}{\beta} F_{13} + \frac{1}{2\beta^2} F_{33})] \quad (7)$$

d. Torsion Test

Finally, the constant F_{44} can be determined from torsion test of thin walled cylinders having the same configuration as those for the biaxial test. Let T be the applied end torques which will cause failure in the cylinder. Then

$$\sigma_4 = \frac{T}{2\pi a^2 t} \quad (8)$$

in which, a is the average radius, and t , the thickness of the cylinder.

Then from Equation (1) the constant F_{44} is given by

$$F_{44} = \frac{4\pi^2 a^4 t^2}{T^2} \quad (9)$$

4. Correlation of Test Data

The strength criterion Equation (1), was applied to four different graphite materials: ATJ, JTA, AIJ-S and H-327. The test data used for determining the strength constants are given in Table 2 and the values of the obtained constants are shown in Table 3. The theoretical predictions and test data are shown plotted in Figures 4-7. With the exception of the quadrant depicting biaxial compression, satisfactory correlations were obtained for all grades of graphite tested.

5. Effect of Temperature and Irradiation

The strength criterion previously given in Equation (1) does not include the effects of temperature and neutron irradiation. When evaluating the HTGR core and core support structures however, the above mentioned conditions should be included in the formulation of the strength criterion.

In the presence of temperature and irradiation, the strength constants may be assumed to be functions of temperature and fast neutron fluence. Thus, (see Equation 1)

$$\begin{aligned} F_1 &= F_1(T, \gamma) \\ F_3 &= F_3(T, \gamma) \end{aligned} \quad (10)$$

and $F_{11} = F_3(T, \gamma)$, etc.

where T = temperature and γ = fast neutron fluence. For simplicity, it is further assumed that the F_1 , F_3 , F_{11} ,, etc. terms in Equation (10)

TABLE 2

Test Data Selected For Strength Characterization

Test Type Graphite	Uniaxial Strength (PSI)				Biaxial Strength (PSI)		Triaxial Strength (PSI)		Shear Strength (PSI)
	R	R'	Z	Z'	σ_2	σ_3	σ_3	$\sigma_1 = \sigma_2$	τ
ATJ	3500	8500	2600	8200	1580	2000	-12980	-2190	2710
JTA	7000	20,000	4000	20,000	4670	2000	—	—	5450
ATJ-S	4200	10,000	4200	10,000	3200	2200	-3590	-3830	4700
H-327	1600	4000	2000	5000	1170	1200	—	—	—

TABLE 3

Computed Strength Constants

Constant Graphite	F_1 $10^{-3} \text{ (psf)}^{-1}$	F_3 $10^{-3} \text{ (psf)}^{-1}$	F_{11} $10^{-7} \text{ (psf)}^{-2}$	F_{33} $10^{-7} \text{ (psf)}^{-2}$	F_{13} $10^{-7} \text{ (psf)}^{-2}$	F_{12} $10^{-7} \text{ (psf)}^{-2}$	F_{44} $10^{-7} \text{ (psf)}^{-2}$
ATJ	0.1681	0.2627	0.3361	0.4690	-0.1977	0.9476*	1.3620
JTA	0.1318	0.1122	0.0909	0.0641	-0.0676	—	0.3367
ATJ-S	0.1381	0.1381	0.2381	0.2381	-0.1499	0.0493	0.4527
H-327	0.3750	0.3000	1.5620	1.000	-0.5578	—	—

4-10

Biaxial Fracture Surface for ATJ Graphite

Figure 4

Biaxial Fracture Surface for JTA Graphite

Figure 5

Biaxial Fracture Surface for ATJ-S Graphite

Figure 6

Biaxial Fracture Surface for H-327 Graphite

Figure 7

can be defined by the expression

$$\begin{aligned}F_a &= f_1(T) g_1(\gamma) \\F_3 &= f_2(T) g_2(\gamma) \\F_{11} &= f_{11}(T) g_{11}(\gamma) \dots, \text{etc.}\end{aligned}\tag{11}$$

The functions f_1 , f_2 , f_{11} , ..., can be determined from the four basic experiments outlined previously in section 3, while the functions g_1 , g_2 , g_{11} , ..., etc. can be determined from similar types of tests carried out with irradiated specimens.

a. Temperature Effect

In general, the strength of graphite increases with increasing temperature and reaches a maximum between 2000 and 2500°C. Beyond this temperature range graphite particles begin to vaporize and the strength subsequently decreases. Strength vs. temperature data obtained from Union Carbide Corporation for ATJ-S and ATJ graphites are shown respectively in Figures 8 and 9⁽²²⁾.

Available test results thus far are not sufficient to enable the characterization of the temperature dependence function. Additional test efforts, especially with HTGR graphites should be undertaken.

b. Irradiation Effects

Very little test data are available for the strength of graphite under irradiation conditions. Results reported in Reference (23) indicate that the compressive strength of graphite increases as a function of neutron exposure and can be from 100-170% greater than that of non-

Tensile and Compressive Strengths of ATJ-S Graphite at High Temperatures

Figure 8

Short-Time Tensile Strength vs. Temperature
for ATJ Molded Graphite

Figure 9

irradiated graphite at room temperature. Test data for KC-graphite (in the grain direction) are shown in Figure 10. No data are available for tensile or biaxial irradiated strengths. Additional experimental work in this area should be carried out as soon as possible.

6. Graphite Failure Criteria - Statistical Approach

The fracture of brittle materials is often described by Weibull theory; rupture under stress is assumed to occur at the most serious flaw. The theory is extended to biaxial stress situations and descriptions of brittle material failure under biaxial stress by these methods has been proposed⁽²¹⁾. Although the fracture surfaces obtained by currently available methods offer adequate descriptions for biaxial tensile failure and to a lesser degree for tensile-compressive failure of various grades of graphite tubing, no comparable description of graphite behavior under biaxial compressive stress is available. In order to obtain a more comprehensive theory for graphite failure in the bi-compressive quadrant we will analyze this situation by itself utilizing various statistical methods.

Initially we will be concerned with type ATJ graphite; the data were obtained from tests on cylindrical specimens with inside diameter of 0.885 inch, outside diameter of 0.985 inch. The stress values at failure for specimens with gauge length-to-mean diameter ratios (L/D) of 1.23, 1.87, 2.67 and 3.64 appear in Figure 11. The radial stress is omitted from our analysis since it is substantially less and directly related to the tangential and longitudinal stress. For more specific informa-

The Effect of Room-Temperature Irradiation on the Strength of KC Graphite.

Figure 10

Figure 11

tion on the testing procedure see Reference (25).

a. ATJ Graphite Regression Analysis

An initial inspection of the data in Figure 11 at once suggests the possibility of a linear relationship between the tangential and longitudinal compressive stresses in ATJ graphite failure of the form $\sigma_3 = m\sigma_1 + b$ with m being the slope and b the σ_3 intercept of the line. The physical explanation is that a failure which occurs at coordinates (x, y) on the line $y_3 = mx_1 + b$ would as well have occurred at any other point, (x', y') , which lies on the given line ($x' < 0$, $y' < 0$). The first two columns of Table 4 list the coordinates at which fracture occurred for specimens with $L/D = 2.67$ and related calculations. To initially test the significance of an apparently linear relation we calculate the correlation coefficient r ,

$$r = \frac{\sum(x_i - \bar{x})(y_i - \bar{y})}{(\sum(x_i - \bar{x})^2 \sum(y_i - \bar{y})^2)^{\frac{1}{2}}} = -0.93 \quad (12)$$

A value of -0.93 indicates a strong possibility of a linear or almost linear relationship so we are justified in pursuing the proposed analysis.

For the regression line of σ_3 on σ_1 least square estimates of m and b and the proposed equation are given by,

$$m = \frac{\sum(x_i - \bar{x})(y_i - \bar{y})}{\sum(x_i - \bar{x})^2} = 1.4, \quad b = \bar{y} - m\bar{x} = -12 \quad (13)$$

$$\sigma_3 = -1.4 \sigma_1 - 12 \quad \text{or} \quad Y = 1.4 X - 12 \quad (14)$$

TABLE 4a

Stress at Fracture for ATJ Graphite L/D = 2.67

<u>x = σ_1</u>	<u>y = σ_3</u>		
-2.25	-9.5	$\bar{x} = -5.42$	$\bar{y} = -4.53$
-2.60	-8.9		
-4.60	-6.25	$\Sigma(x_i - \bar{x})^2 = 31$	$\Sigma(y_i - \bar{y})^2 = 69$
-5.75	-5.0		
-5.85	-2.8	$\Sigma(x_i - \bar{x})(y_i - \bar{y}) = -43$	
-6.0	-3.0		
-6.40	-3.25		
-6.50	-2.0		
-6.50	-4.0		
-6.75	-1.6		
-7.0	-3.5		

TABLE 4b

Confidence Regions for ATJ Graphite L/D = 2.67

<u>X</u>	<u>Y</u>	<u>c = 97.5</u>	<u>c = 99</u>
0	-12	(-15.1, -8.9)	(-15.8, -8.2)
-1	-10.6	(-13.4, -7.8)	(-14.1, -7.1)
-2	- 9.2	(-11.8, -6.6)	(-12.4, -6.0)
-3	- 7.8	(-10.2, -5.4)	(-10.8, -4.8)
-4	- 6.4	(- 8.6, -4.2)	(- 9.3, -3.5)
-5	- 5.0	(- 7.2, -2.8)	(- 7.8, -2.2)
-6	- 3.6	(- 5.85, -1.35)	(- 6.4, -0.8)
-7	- 2.2	(- 4.5, 0)	(- 5.1, 0)
-8	- 0.8	(- 3.2, 0)	(- 3.8, 0)

column 3 is $Y \pm t_{97.5} [v^2(Y|X) + s^2(y)]^{\frac{1}{2}}$

column 4 is $Y \pm t_{99} [v^2(Y|X) + s^2(y)]^{\frac{1}{2}}$

The line (14) appears in Figure 11.

In our analysis we are tacitly assuming that:

- a) $y_i = mx_i + b + e_i$, where e_i is a random error
- b) e_i has mean 0, $\sigma^2(e_i) = \sigma^2(y)$, and the e_i 's are uncorrelated
- c) the x_i 's contain no measurement errors; all the errors are in the y_i 's.

Heuristically this means that a tangential stress, x , is chosen and the resultant longitudinal stress, y , necessary to fracture the specimen is measured; variation in y will be caused by differing flaw shape and size plus error of measurement.

i. Confidence Regions for Mean Fracture Line

In order to obtain confidence regions for the accuracy of our regression line (14) we estimate the variances of y and of Y ,

$$\text{Est. Var}(y) = s^2(y) = \frac{\sum_{i=1}^N (y_i - \bar{y})^2}{N-2} = 0.9 \quad (15)$$

$$\text{Est. Var}(Y|x) = v^2(Y|x) = s^2(y) \left[\frac{1}{N} + \frac{(x - \bar{x})^2}{\sum_{i=1}^N (x_i - \bar{x})^2} \right] \text{ at any } x. \quad (16)$$

The appropriate multiplier for a 90% confidence region⁽²⁴⁾ is $[2F(0.90, 2, N-2)]^{\frac{1}{2}}$; so the probability that the mean fracture line lies between the two hyperbolic sections defined by

$$Y \pm [2F(0.90, 2, 9)]^{\frac{1}{2}} v(Y|x) \text{ is } 90\% \quad (17)$$

where Y is given by (14) and $v(Y|x)$ by (16). Equivalently, the mean longitudinal fracture stress, Y , for a sample all with tangential stress

equal to x will be covered by the interval in (17) 90% of the time.

ii. Confidence Regions for Stress Values

In order to obtain an $c\%$ confidence region for the stress at failure we use the t -distribution. The appropriate curves are,

$$Y \pm t_{c, N-2} [v^2(Y|x) + s^2(y)]^{\frac{1}{2}} \text{ or equivalently} \quad (18)$$

$$\text{or} \quad y = -1.4x - 12 \pm 0.95 t_{c, 9} [1.09 + (x - 5.42)^2/31]^{\frac{1}{2}} \quad (19)$$

by (14), (15) and (16). Then for any $0 < c < 100$, the probability that the point of fracture stress for an ATJ type graphite cylinder with $L/D = 2.67$ lies between the two hyperbolic sections defined by (19) is $2c\%$ and the probability that it lies below the upper curve given by (19) is $c\%$, or

$$P(\sigma_3 < 1.4\sigma_1 - 12 + 0.95 t_{c, 9} [1.09 + (x - 5.42)^2/31]^{\frac{1}{2}}) = c/100. \quad (20)$$

The confidence regions for $c = 97.5$ and $c = 99$ are depicted in Figure 12 and the calculations are listed in Table 4b.

iii. Regression of σ_1 on σ_3

Our choice of assuming σ_1 is fixed and allowing σ_3 to carry all the error is arbitrary, we could have assumed the reverse or even that both σ_1 and σ_2 have an inherent error (this may be closest to the physical situation). These possibilities are under further investigation as well as the situation where σ_1 and σ_3 are jointly described by a non-linear relationship. The σ_1 on σ_3 regression line is given by

$$\sigma_1 = -0.62\sigma_3 - 8.2 \quad (21)$$

and appears in Figure 11; it obviously does not differ much from the σ_3 on σ_1

Figure 12

line. Similarly, their respective confidence regions will not differ much so we may as well restrict ourselves for the present to the σ_3 on σ_1 regression line.

Perhaps a more serious consideration in the above procedure is the true form of the e_i 's; regardless of whether the axial stresses are linearly related or not the distribution of y about its true mean is still to be determined. The usual assumption of normality of the e_i 's in regression analysis is not completely valid, especially for large values of σ_1 , since σ_3 cannot be positive but it may serve as an initial approximation.

b. Weibull Estimation

The nature of the fractures, which were caused by highly localized collapse, indicates failure initiated by inherent material defects; this, in turn, suggests that failure might follow a Weibull distribution as in the tensile situation. The advantages of this approach are obvious; once the Weibull modulus and normalizing factor are chosen for any given L/D tubing we can easily calculate the corresponding probability estimates and confidence regions for tubing of other lengths.

From the regression line (14) we extrapolate the values of σ_3 had σ_1 been constant at -7.0 and then estimate the Weibull modulus, m , and normalizing factor, σ . Plotting $\log_{10} \left(\frac{1}{1-F} \right)$ against $\log(\sigma_3 | \sigma_1 = -7.0)$ our least square estimate of m is 2.45 and we estimate $\sigma = 2.8$. We now have, for $t \leq 0$

$$P(\sigma_3 \leq t | \sigma_1 = -7.0) = F(t) = 1 - e^{-(t/2.8)^{2.45}} \quad (22)$$

as our estimate of the distribution of the longitudinal stress necessary for the tube to fail given it is under a tangential stress of -7.0 (kpsi).

For any $-7 \leq \sigma_1 < 0$ our estimate of the distribution is,

$$P(\sigma_3 \leq t | \sigma_1) = F_{\sigma_1}(t) = 1 - \left[e^{-\left(-\frac{g(t)}{\sigma}\right)^m} \right] \quad (23)$$

where $g_{\sigma_1}(t) = t + 1.4\sigma_1 + 9.8$ for $t < -1.4\sigma_1 - 9.8$, $m = 2.45$ and $\sigma = 2.8$. Now for tubes of the same diameter, D , but length $L' = \alpha L$ with $0 < \alpha < \infty$, we get

$$\exp\left[-\alpha\left(-\frac{g(t)}{2.8}\right)^m\right] = \exp\left[-\left(\frac{g(t)}{\sigma_\alpha}\right)^m\right] \text{ or} \quad (24)$$

$$\sigma_\alpha = 2.8 \alpha^{-1/m}$$

So our estimate of the distribution is,

$$P(\sigma_3 \leq t | \sigma_1, L' = \alpha L) = 1 - \exp\left[-\left(\frac{g(t)}{\sigma_\alpha}\right)^m\right] \quad (25)$$

with m as above and σ_α given by (24).

The obvious difficulty with this model is that probability of zero is assigned to the set of coordinates (σ_1, t) with $0 > t > -1.4\sigma_1 - 9.8$, a possibility which obviously cannot be discounted. Our distributions (23) and (25) are of course dependent on the accuracy of the estimates for m and σ ; various confidence regions which would account for these variations are under investigation.

Remarks

The available data indicates that the longitudinal-tangential stress relationship in the biaxial compressive quadrant might be linear and that

the variation about the regression line might be a Weibull variable. Further refinement of these models is necessary to determine whether an adequate fit to the experimental results is obtained by these methods and whether other models should be pursued. The data for types ATJ-S and JT-50 graphite does not seem to allow any near linear relationship; a parabolic type of fracture surface seems warranted. This and other possible models are currently being analyzed.

4.2.7 HTGR Core Seismic Evaluation

The initial phase of this effort was concentrated on reviewing the work carried out thus far. This includes studies of the preliminary safety analysis reports for the Fulton Generating Station, the HTGR Nuclear Training Manual (April 22-26, 1974), the paper entitled "Design Development of the HTGR Core and Its Support Structure - Seismic Consideration^(26,28) and various internal BNL and LASL trip reports dealing with HTGR seismic programs based on personal discussion with Gulf General Atomic employees.

The objective of this seismic evaluation study is to provide an independent engineering evaluation of the adequacy of the seismic design of the HTGR. In order to determine whether dynamic failure is indeed possible, it is important to study in detail the dynamic response of the core when subjected to earthquake excitation. Information obtained from the dynamic response will permit evaluation of the forces and ultimately the stresses from which the possibility of failure could be investigated.

It should be pointed out, however, that the core response is a complex problem because of the non-linearities inherent in the existing designs. Gaps are present between horizontally adjacent graphite elements. For a vertical stack, coupled horizontal and rotary motion may allow separation of adjacent vertical elements during seismic excitation. Furthermore, the static stress-strain relationship (i.e. constitutive law) of graphite is non-linear^(20, 29) and as a consequence the system response is also non-linear.

At the present time Gulf General Atomic has developed several digital computer programs to study the response of the graphite core. One of these is used to evaluate the horizontal response of a horizontal array of graphite core elements acting in a single plane. Clearance effects are modeled as spring elements with gaps between them. The other programs deal with vertical columns and more complex aspects of the dynamic core models.

Since none of the Gulf General Atomic programs are currently available and, in order to carry out an independent review of the core seismic response, it is necessary to develop our own evaluation techniques. With these independently developed methods, it will be possible to verify the behavior of the system to input forcing functions for changes in such characteristics as clearances, non-linear springs and damping, and non-linear material properties. Simple shake table tests will be carried out to check some of these results.

Initial Review

In evaluating the available seismic response information it became obvious that the following general data necessary for the seismic analysis were not available:

(1) Material properties - the modulus of elasticity, poisson's ratio, density and yield strength applicable to the fuel elements, the top, bottom and side reflector elements, the support posts and blocks, the borinated shields and the dowels. These properties were not available at any of the applicable temperatures including ambient, operational and nvt temperature conditions.

(2) Geometric conditions - dimensions of all reflector elements, support blocks and posts, borinated shield, spring packs and dowels; the operational clearances between all elements including the dowel pin and hole clearances.

(3) GA test program - the scaling method used in three dimensions as applied to the various blocks, clearances, springs and posts; the forcing function characteristics; the test fixture structural and dynamic characteristics.

Analytical Modeling and Initial Results

An analytical evaluation of the seismic response information presented in the HTGR Nuclear Training Manual has been initiated. In particular, Section 5 of the Manual is concerned with the Core Seismic Program. A figure in the Manual shows the core motion characteristics in

response to an input of 3.5 Hz. The core is pictured as consisting of three masses with clearances between each. Springs are located at the outside ends of the segmented core masses. The central mass represents the core while the two outer masses represent the reflector blocks. The springs represent the spring packs.

One of the cores looked at was composed of 3944 blocks arranged in 493 vertical stacks. Reflector blocks are arranged around the outside circumference of the core stacks. Radial springs are located between a borinated shield and the outer shell of the structure. The borinated shield is adjacent to the outside surface of the reflector blocks. The model represented in the HTGR Manual and shown in Figure 13 shows a single layer through a typical cross section of the core.

It was decided that the information presented in the HTGR Manual could be evaluated analytically. To do this a mathematical model representing a single layer of elements was considered. This is shown in Figure 14. The core structure is represented by a series of masses $m_1, m_2, \dots, m_i, \dots, m_n$. The elastic properties between adjacent masses are indicated by K_i . The elements designated by K_1 and K_{n+1} represent the elastic constants of the reflector blocks when in contact with the spring packs and borinated shield. The elements designated by K_2 to K_n represent the elastic properties of adjacent fuel elements when in contact. For the initial model, general damping is assumed to occur whenever there is a relative velocity between adjacent masses. Clearances between adjacent

CORE MOTION CHARACTERISTICS

Figure 13

MATHEMATICAL MODEL of SINGLE LAYER of ELEMENTS

$M_1, M_2, \dots, M_N$	MASSSES
$K_1, K_2, \dots, K_{N+1}$	SPRINGS
$C_1, C_2, \dots, C_{N+1}$	DAMPERS
$A_1, A_2, \dots, A_{N+1}$	INITIAL CLEARANCES
X_{01}, X_{02}	FOUNDATION DISPLACEMENTS
$X_1, X_2, \dots, X_N$	MASS DISPLACEMENTS

Figure 14

masses are given by A_i . Motional inputs to the system are provided by X_{01} on the left and X_{02} on the right.

The equations of motion for the system in Figure 14 are given by:

$$\begin{aligned} m_1 \ddot{X}_1 + K_1(X_1 - X_{01}) + C_1(\dot{X}_1 - \dot{X}_{01}) + K_2(X_1 - X_2) + C_2\left(\frac{\dot{X}_1 - \dot{X}_2}{X_1 - X_2}\right) &= 0 \\ m_2 \ddot{X}_2 + K_2(X_2 - X_1) + C_2(\dot{X}_2 - \dot{X}_1) + K_3(X_2 - X_3) + C_3(\dot{X}_2 - \dot{X}_3) &= 0 \end{aligned} \quad (26)$$

In general,

$$m_i \ddot{X}_i + K_i(X_i - X_{i-1}) + C_i(\dot{X}_i - \dot{X}_{i-1}) + K_{i+1}(X_i - X_{i+1}) + C_{i+1}(\dot{X}_i - \dot{X}_{i+1}) = 0 \quad (27)$$

where K_i are non-linear functions of the related displacements $(X_i - X_{i-1})$.

A computer program was written to describe the motion of the general, in-line, n degree of freedom system. Options include the ability to introduce one- or two-dimensional input arrays to describe the non-linear parameters and the acceptance of arbitrary forcing functions.

Initially several simple cases were considered. In particular, two and three degree of freedom systems were investigated for harmonic inputs at either end. The inputs were either in phase or out of phase with each other. These cases were investigated since some response characteristics could be predicted. These known solutions were used to check out the accuracy of the computer program.

Six cases were solved and the results examined to check out the program. Specifically, the program was tested to determine whether

the introduction of non-linearity into some of the parameters could be properly processed. These parameters include the clearances, the non-linearities, the damping and the inputs. The six cases, and the range of the parameters used in each, are shown in Figures 15a through 15f. For all of these, a simple sinusoidal forcing function of unit amplitude ($A=1$) and adjustable frequency (B) was used. In the general program, the inputs may be independently specified. However, for the test cases to be presented the inputs are of equal magnitude and either in phase or 180 degrees out of phase with each other.

Case 1, Figure 15a, was a simple, undamped, linear, two degree of freedom system with symmetrical inputs at either end. Equal masses and springs were used. Because of symmetry, the response should be equivalent to that of a single degree of freedom system. This occurs if both masses respond with equal displacement so that the relative motion ($X_1 - X_2$) is zero. A typical output from the computer for Case 1 is shown in Table 5. This shows the values of the displacements X_1 and X_2 , the frequency B , the inputs X_0 and the velocities of each mass. The range of the frequency (B) was varied from zero to a value of 10 rad/sec. This range of inputs covers the resonant characteristics of the system. The printout is in intervals of 0.5 rad/sec. As expected, the listed values of X_1 and X_2 in Table 5 are identical.

Case 2, shown in Figure 15b, depicts a linear, damped, spring-mass system. The damping was introduced only at the outer ends

CASE 1 UNDAMPED, LINEAR SPRING-MASS SYSTEM

Figure 15a

CASE 2 DAMPED LINEAR SPRING-MASS SYSTEM

Figure 15b

CASE 3 DAMPED SPRING-MASS SYSTEM WITH NON LINEAR BREAK-POINT

Figure 15c

CASE 4 DAMPED SPRING - MASS SYSTEM WITH NONLINEAR BREAK-POINT

Figure 15d

CASE 5 DAMPED SPRING-MASS SYSTEM WITH CLEARANCE

Figure 15e

CASE 6 DAMPED SPRING-MASS SYSTEM WITH CLEARANCE

$$\begin{aligned} M_1 = M_3 &= .167, \quad M_2 = 1.67 \\ K_{11} = K_{21} = K_{31} = K_{41} &= 0 \\ K_{12} = K_{42} &= 39, \quad K_{32} = K_{22} = 390 \\ C_1 = C_4 &= .1258, \quad C_2 = C_3 = 1.258 \\ B &= 4, 5, 6 \end{aligned}$$

Figure 18F

TABLE 5

X_1	X_2	B	X_0	$\dot{X}_1$	$\dot{X}_2$
.2969E-05	.2969E-05	.5000E+00	.5000E-02	.8098E-03	.8098E-03
.2392E-04	.2392E-04	.5000E+00	.1000E-01	.3416E-02	.3416E-02
.8078E-04	.8078E-04	.5000E+00	.1500E-01	.7809E-02	.7809E-02
.1913E-03	.1913E-03	.5000E+00	.2000E-01	.1397E-01	.1397E-01
.3732E-03	.3732E-03	.5000E+00	.2500E-01	.2189E-01	.2189E-01
.5436E-03	.5436E-03	.5000E+00	.3000E-01	.3152E-01	.3152E-01
.1020E-02	.1020E-02	.5000E+00	.3499E-01	.4284E-01	.4284E-01
.1518E-02	.1518E-02	.5000E+00	.3999E-01	.5580E-01	.5580E-01
.2155E-02	.2155E-02	.5000E+00	.4498E-01	.7036E-01	.7036E-01
.2946E-02	.2946E-02	.5000E+00	.4998E-01	.8647E-01	.8647E-01
.3906E-02	.3906E-02	.5000E+00	.5497E-01	.1041E+00	.1041E+00
.5050E-02	.5050E-02	.5000E+00	.5996E-01	.1231E+00	.1231E+00
.6392E-02	.6392E-02	.5000E+00	.6495E-01	.1434E+00	.1434E+00
.7944E-02	.7944E-02	.5000E+00	.6994E-01	.1651E+00	.1651E+00
.9720E-02	.9720E-02	.5000E+00	.7493E-01	.1879E+00	.1879E+00
.1173E-01	.1173E-01	.5000E+00	.7991E-01	.2119E+00	.2119E+00
.1399E-01	.1399E-01	.5000E+00	.8490E-01	.2369E+00	.2369E+00
.1650E-01	.1650E-01	.5000E+00	.8988E-01	.2628E+00	.2628E+00
.1927E-01	.1927E-01	.5000E+00	.9486E-01	.2896E+00	.2896E+00
.2232E-01	.2232E-01	.5000E+00	.9983E-01	.3171E+00	.3171E+00
.2564E-01	.2564E-01	.5000E+00	.1048E+00	.3453E+00	.3453E+00
.2925E-01	.2925E-01	.5000E+00	.1098E+00	.3740E+00	.3740E+00
.3316E-01	.3316E-01	.5000E+00	.1147E+00	.4032E+00	.4032E+00
.3735E-01	.3735E-01	.5000E+00	.1197E+00	.4327E+00	.4327E+00
.4184E-01	.4184E-01	.5000E+00	.1247E+00	.4625E+00	.4625E+00
.4663E-01	.4663E-01	.5000E+00	.1296E+00	.4923E+00	.4923E+00
.5172E-01	.5172E-01	.5000E+00	.1346E+00	.5222E+00	.5222E+00
.5710E-01	.5710E-01	.5000E+00	.1395E+00	.5520E+00	.5520E+00
.6278E-01	.6278E-01	.5000E+00	.1445E+00	.5815E+00	.5815E+00
.6876E-01	.6876E-01	.5000E+00	.1494E+00	.6108E+00	.6108E+00
.7503E-01	.7503E-01	.5000E+00	.1544E+00	.6396E+00	.6396E+00
.8158E-01	.8158E-01	.5000E+00	.1593E+00	.6679E+00	.6679E+00
.8841E-01	.8841E-01	.5000E+00	.1643E+00	.6956E+00	.6956E+00
.9552E-01	.9552E-01	.5000E+00	.1692E+00	.7226E+00	.7226E+00
.1029E+00	.1029E+00	.5000E+00	.1741E+00	.7487E+00	.7487E+00
.1105E+00	.1105E+00	.5000E+00	.1790E+00	.7739E+00	.7739E+00
.1184E+00	.1184E+00	.5000E+00	.1839E+00	.7981E+00	.7981E+00
.1265E+00	.1265E+00	.5000E+00	.1889E+00	.8212E+00	.8212E+00
.1348E+00	.1348E+00	.5000E+00	.1938E+00	.8431E+00	.8431E+00
.1434E+00	.1434E+00	.5000E+00	.1987E+00	.8637E+00	.8637E+00
.1521E+00	.1521E+00	.5000E+00	.2036E+00	.8830E+00	.8830E+00
.1611E+00	.1611E+00	.5000E+00	.2085E+00	.9009E+00	.9009E+00
.1702E+00	.1702E+00	.5000E+00	.2133E+00	.9173E+00	.9173E+00
.1794E+00	.1794E+00	.5000E+00	.2182E+00	.9321E+00	.9321E+00
.1888E+00	.1888E+00	.5000E+00	.2231E+00	.9454E+00	.9454E+00
.1983E+00	.1983E+00	.5000E+00	.2280E+00	.9570E+00	.9570E+00
.2079E+00	.2079E+00	.5000E+00	.2328E+00	.9669E+00	.9669E+00
.2177E+00	.2177E+00	.5000E+00	.2377E+00	.9751E+00	.9751E+00
.2275E+00	.2275E+00	.5000E+00	.2426E+00	.9815E+00	.9815E+00
.2373E+00	.2373E+00	.5000E+00	.2474E+00	.9861E+00	.9861E+00
.2472E+00	.2472E+00	.5000E+00	.2522E+00	.9889E+00	.9889E+00
.2571E+00	.2571E+00	.5000E+00	.2571E+00	.9900E+00	.9900E+00
.2670E+00	.2670E+00	.5000E+00	.2619E+00	.9891E+00	.9891E+00
.2768E+00	.2768E+00	.5000E+00	.2667E+00	.9865E+00	.9865E+00
.2867E+00	.2867E+00	.5000E+00	.2715E+00	.9820E+00	.9820E+00
.2965E+00	.2965E+00	.5000E+00	.2764E+00	.9758E+00	.9758E+00
.3062E+00	.3062E+00	.5000E+00	.2812E+00	.9677E+00	.9677E+00

because the response for such a condition is known to be essentially equivalent to that of a one degree of freedom system. Figure 16 is a plot of the computer results for an input frequency of 6.0 rad/sec and a damping coefficient of 0.1256. Figure 16a shows the response for the first 10 seconds. The response has within it the initial conditions of zero displacement and zero velocity. Figure 16b shows an extended plot of the maximum values only. As can be seen from the figures the initial transient has diminished and the response is attaining its steady state level. This checks with the known response for these conditions.

Case 3, shown in Figure 15c, involves a damped, spring-mass system with a non-linear breakpoint. This case is introduced to establish the capability of the program to handle non-linear spring characteristics. For this case, the spring has a hard-soft breakpoint characteristic. K_{11} is the spring constant for small deflections while K_{12} is the spring constant for larger relative deflections. The breakpoint occurs at a relative displacement of 10. Springs K_1 and K_3 both have this non-linear characteristic. K_2 is a linear spring over the entire range.

Figure 17 shows a plot of the displacements X_1 and X_2 . Comparing these results with Figure 16a, the plot for an identical but linear system, the effects of the non-linear springs are readily identified. Beyond this the breakpoint is encountered which introduces the non-linearity and hence, as expected, the curves differ.

Figure 16a

CASE 2

Figure 16b

Figure 17 (8:6) CASE 3

Case 4, shown in Figure 15d, is similar to Case 3 except that it is a three degree of freedom system. Results for inphase inputs and response less than that which is necessary to reach the breakpoint as shown in Table 6, reveal that X_3 is identical to X_1 , while X_2 is greater than either. This is as expected for a linear three degree of freedom system. After 4 seconds, which is in accordance with the time required for the breakpoint to be reached in the previous case, the values for X_3 differ from those of X_1 .

Results for out of phase inputs are given in Table 7. Here it should be noted that the response agrees with linear theory. X_3 is equal but opposite to X_1 , and X_2 is essentially zero. The reason for this is that the displacements are small and the breakpoint is not reached. Thus the response should indeed remain linear.

Case 5, shown in Figure 15e, represents a damped, spring-mass system with clearances. The two external springs have a spring constant of zero until the relative displacement is equal to the clearance. At this point, the spring assumes a value of 39.478. The center spring is linear over the entire range. Computer results for the response are plotted in Figure 18. The effects of the non-linearity are identifiable by the double curvature in both the response curves for X_1 and X_2 . As depicted in the Figure, the double curvature for one variable occurs in the negative region while the double curvature for the other takes place in the positive region. The reason for this phenomena is due

TABLE 6

CASE 4 $X_{01}=X_{02}$

X_1	X_2	X_3	X_{01}	X_{02}	TIME
.9175E+00	.9052E+00	.9175E+00	.9098E+00	.9098E+00	.3400E+01
.3474E+01	.3685E+01	.3474E+01	.9612E+00	.9612E+00	.3450E+01
.5787E+01	.6234E+01	.5787E+01	.8367E+00	.8367E+00	.3500E+01
.7635E+01	.8225E+01	.7635E+01	.6374E+00	.6374E+00	.3550E+01
.8837E+01	.9550E+01	.8837E+01	.3817E+00	.3817E+00	.3600E+01
.9267E+01	.1130E+02	.9267E+01	.9102E-01	.9102E-01	.3650E+01
.8866E+01	.9633E+01	.8866E+01	-.2073E+00	-.2073E+00	.3700E+01
.7652E+01	.8339E+01	.7652E+01	-.4872E+00	-.4872E+00	.3750E+01
.5714E+01	.6258E+01	.5714E+01	-.7235E+00	-.7235E+00	.3800E+01
.3211E+01	.3559E+01	.3211E+01	-.8952E+00	-.8952E+00	.3850E+01
.3502E+01	.4738E+00	.3502E+01	-.9869E+00	-.9869E+00	.3900E+01
-.2591E+01	-.2725E+01	-.2591E+01	-.9905E+00	-.9905E+00	.3950E+01
-.5373E+01	-.5750E+01	-.5373E+01	-.9056E+00	-.9056E+00	.4000E+01
-.7731E+01	-.8320E+01	-.7731E+01	-.7398E+00	-.7398E+00	.4050E+01
-.9670E+01	-.1110E+02	-.9670E+01	-.5079E+00	-.5079E+00	.4100E+01
-.1033E+02	-.1118E+02	-.1033E+02	-.2306E+00	-.2306E+00	.4150E+01
-.1036E+02	-.1137E+02	-.1211E+02	.6721E-01	.6721E-01	.4200E+01
-.1021E+02	-.1124E+02	-.1105E+02	.3591E+00	.3591E+00	.4250E+01
-.9680E+01	-.1040E+02	-.1053E+02	.6188E+00	.6188E+00	.4300E+01
-.8265E+01	-.9270E+01	-.9589E+01	.8233E+00	.8233E+00	.4350E+01
-.7004E+01	-.7590E+01	-.6351E+01	.9543E+00	.9543E+00	.4400E+01
-.4584E+01	-.4993E+01	-.4665E+01	.1000E+01	.1000E+01	.4450E+01
-.1576E+01	-.1034E+01	-.1852E+01	.9564E+00	.9564E+00	.4500E+01
.1137E+01	.1369E+01	.1619E+01	.8273E+00	.8273E+00	.4550E+01
.4453E+01	.4637E+01	.4024E+01	.6244E+00	.6244E+00	.4600E+01
.6870E+01	.7465E+01	.7123E+01	.3657E+00	.3657E+00	.4650E+01
.8898E+01	.9692E+01	.8844E+01	.7427E-01	.7427E-01	.4700E+01
.1062E+02	.1100E+02	.1013E+02	-.2238E+00	-.2238E+00	.4750E+01
.1230E+02	.1168E+02	.1361E+02	-.5018E+00	-.5018E+00	.4800E+01
.1166E+02	.1196E+02	.1095E+02	-.7350E+00	-.7350E+00	.4850E+01
.1141E+02	.1153E+02	.1059E+02	-.9026E+00	-.9026E+00	.4900E+01
.1076E+02	.1065E+02	.9512E+01	-.9895E+00	-.9895E+00	.4950E+01
.9178E+01	.9200E+01	.8488E+01	-.9880E+00	-.9880E+00	.5000E+01
.6258E+01	.7248E+01	.6523E+01	-.8983E+00	-.8983E+00	.5050E+01
.4857E+01	.4424E+01	.4135E+01	-.7284E+00	-.7284E+00	.5100E+01
.1244E+01	.1207E+01	.9641E+00	-.4933E+00	-.4933E+00	.5150E+01
-.2232E+01	-.2132E+01	-.1912E+01	-.2143E+00	-.2143E+00	.5200E+01
-.4674E+01	-.5325E+01	-.4916E+01	.8397E-01	.8397E-01	.5250E+01
-.7460E+01	-.7950E+01	-.7349E+01	.3747E+00	.3747E+00	.5300E+01
-.9075E+01	-.9858E+01	-.8963E+01	.6320E+00	.6320E+00	.5350E+01
-.9893E+01	-.1083E+02	-.1116E+02	.9328E+00	.9328E+00	.5400E+01
-.1023E+02	-.1137E+02	-.1165E+02	.9592E+00	.9592E+00	.5450E+01
-.1042E+02	-.1138E+02	-.1138E+02	.9999E+00	.9999E+00	.5500E+01
-.9712E+01	-.1074E+02	-.1122E+02	.9513E+00	.9513E+00	.5550E+01
-.8767E+01	-.9697E+01	-.9551E+01	.8178E+00	.8178E+00	.5600E+01
-.7458E+01	-.8049E+01	-.7121E+01	.6112E+00	.6112E+00	.5650E+01
-.5093E+01	-.5576E+01	-.5114E+01	.3500E+00	.3500E+00	.5700E+01
-.2347E+01	-.2594E+01	-.2546E+01	.5749E-01	.5749E-01	.5750E+01
.4782E+00	.5895E+00	.7541E+00	-.2401E+00	-.2401E+00	.5800E+01
.3418E+01	.3720E+01	.3182E+01	-.5163E+00	-.5163E+00	.5850E+01
.5847E+01	.6397E+01	.5979E+01	-.7463E+00	-.7463E+00	.5900E+01
.7688E+01	.8458E+01	.7667E+01	-.9097E+00	-.9097E+00	.5950E+01
.9651E+01	.9532E+01	.9811E+01	-.9018E+00	-.9018E+00	.6000E+01
.9801E+01	.9894E+01	.9056E+01	-.9853E+00	-.9853E+00	.6050E+01
.1020E+02	.9635E+01	.9576E+01	-.8908E+00	-.8908E+00	.6100E+01

TABLE 7

CASE 4 $X_{01} = -X_{02}$

X_1	X_2	X_3	X_{01}	X_{02}	TIME
-.7397E-01	.1716E-12	.7397E-01	-.8022E+00	.8022E+00	.1660E+02
-.5641E-01	.2176E-12	.5641E-01	-.5899E+00	.5899E+00	.1665E+02
-.2098E-01	.2251E-12	.2098E-01	-.3250E+00	.3250E+00	.1670E+02
-.2890E-02	.2225E-12	.2890E-02	-.3096E-01	.3096E-01	.1675E+02
-.2649E-01	.1998E-12	.2649E-01	-.2658E+00	.2658E+00	.1680E+02
.4066E-01	.1591E-12	-.4966E-01	.5388E+00	-.5388E+00	.1685E+02
.7679E-01	.1241E-12	-.7539E-01	.7637E+00	-.7637E+00	.1690E+02
.9484E-01	.3967E-13	-.8684E-01	.9204E+00	-.9204E+00	.1695E+02
.9172E-01	-.2457E-13	-.9172E-01	.9948E+00	-.9948E+00	.1700E+02
.9039E-01	-.9497E-13	-.9039E-01	.9804E+00	-.9804E+00	.1705E+02
.8168E-01	-.1526E-12	-.8168E-01	.8786E+00	-.8786E+00	.1710E+02
.6477E-01	-.1969E-12	-.6437E-01	.6980E+00	-.6980E+00	.1715E+02
.4199E-01	-.2238E-12	-.4199E-01	.4552E+00	-.4552E+00	.1720E+02
.1596E-01	-.2305E-12	-.1596E-01	.1717E+00	-.1717E+00	.1725E+02
-.1169E-01	-.2160E-12	.1169E-01	.1271E+00	-.1271E+00	.1730E+02
-.3820E-01	-.1835E-12	.3820E-01	-.4145E+00	.4145E+00	.1735E+02
-.6129E-01	-.1736E-12	.6129E-01	-.6650E+00	.6650E+00	.1740E+02
-.7890E-01	-.7154E-13	.7890E-01	-.8560E+00	.8560E+00	.1745E+02
-.8947E-01	-.2642E-14	.8947E-01	-.9795E+00	.9795E+00	.1750E+02
-.9205E-01	.6620E-13	.9205E-01	-.9994E+00	.9994E+00	.1755E+02
-.8640E-01	.1297E-12	.8640E-01	-.9371E+00	.9371E+00	.1760E+02
-.7304E-01	.1816E-12	.7304E-01	-.7920E+00	.7920E+00	.1765E+02
-.5315E-01	.2171E-12	.5315E-01	-.5763E+00	.5763E+00	.1770E+02
-.2851E-01	.2336E-12	.2851E-01	-.3090E+00	.3090E+00	.1775E+02
-.1730E-02	.2289E-12	.1730E-02	-.1415E-01	.1415E-01	.1780E+02
.2597E-01	.2037E-12	-.2597E-01	.2820E+00	-.2820E+00	.1785E+02
.5095E-01	.1622E-12	-.5095E-01	.5529E+00	-.5529E+00	.1790E+02
.7138E-01	.1021E-12	-.7138E-01	.7745E+00	-.7745E+00	.1795E+02
.9544E-01	.3467E-13	-.8544E-01	.9268E+00	-.9268E+00	.1800E+02
.9186E-01	-.3529E-13	-.9186E-01	.9964E+00	-.9964E+00	.1805E+02
.9008E-01	-.1041E-12	-.9008E-01	.9770E+00	-.9770E+00	.1810E+02
.8025E-01	-.1625E-12	-.8025E-01	.8703E+00	-.8703E+00	.1815E+02
.6725E-01	-.2065E-12	-.6725E-01	.6858E+00	-.6858E+00	.1820E+02
.4060E-01	-.2775E-12	-.4060E-01	.4401E+00	-.4401E+00	.1825E+02
.1477E-01	-.2374E-12	-.1477E-01	.1551E+00	-.1551E+00	.1830E+02
-.1323E-01	-.2211E-12	.1323E-01	-.1438E+00	.1438E+00	.1835E+02
-.3060E-01	-.1857E-12	.3060E-01	-.4298E+00	.4298E+00	.1840E+02
-.6244E-01	-.1321E-12	.6244E-01	-.6774E+00	.6774E+00	.1845E+02
-.7069E-01	-.6735E-13	.7069E-01	-.8646E+00	.8646E+00	.1850E+02
-.9993E-01	.3738E-14	.9993E-01	-.9744E+00	.9744E+00	.1855E+02
-.9195E-01	.7494E-13	.9195E-01	-.9973E+00	.9973E+00	.1860E+02
-.8585E-01	.1395E-12	.8585E-01	-.9311E+00	.9311E+00	.1865E+02
-.7208E-01	.1915E-12	.7208E-01	-.7817E+00	.7817E+00	.1870E+02
-.5188E-01	.2266E-12	.5188E-01	-.5624E+00	.5624E+00	.1875E+02
-.2703E-01	.2414E-12	.2703E-01	-.2930E+00	.2930E+00	.1880E+02
.2204E-03	.2347E-12	-.2204E-03	.2664E-02	-.2664E-02	.1885E+02
.2766E-01	.2068E-12	-.2766E-01	.2991E+00	-.2991E+00	.1890E+02
.5224E-01	.1613E-12	-.5224E-01	.5668E+00	-.5668E+00	.1895E+02
.7236E-01	.9952E-13	-.7236E-01	.7850E+00	-.7850E+00	.1900E+02
.8601E-01	.2944E-13	-.8671E-01	.9330E+00	-.9330E+00	.1905E+02
.9198E-01	-.4736E-13	-.9198E-01	.9977E+00	-.9977E+00	.1910E+02
.8973E-01	-.1124E-12	-.8973E-01	.9732E+00	-.9732E+00	.1915E+02
.7967E-01	-.1715E-12	-.7967E-01	.8619E+00	-.8619E+00	.1920E+02
.6211E-01	-.2157E-12	-.6211E-01	.6735E+00	-.6735E+00	.1925E+02
.3920E-01	-.2431E-12	-.3920E-01	.4250E+00	-.4250E+00	.1930E+02

FIGURE 18 CASE 5

to the non-linear, non-symmetric spring constant employed for this problem. Neither K_1 nor K_3 can support tensile loads. Consequently, for relative in phase motion, when K_1 is in compression, K_3 is zero and vice versa. This leads to the symmetric-antisymmetric response depicted.

Case 6, shown in Figure 15f, is a three degree of freedom system with all four springs having similar non-linear characteristics. The designated clearances are all equal. Figure 19 shows the response for $B=4.0$ inphase inputs. As is indicated, X_1 and X_3 are equal in magnitude except that the ordinate is displaced differently for each. This is due to the clearance effect associated with every spring which results in a cyclic reversal of the effective spring constant acting on masses m_1 and m_3 . As is to be expected the cyclic change for m_1 and m_3 is 180 degrees out of phase.

Figure 20 shows a plot of maxima for the X_1 displacement at different frequencies, $B=4.0, 5.0$ and 6.0 . Note that the maximum response occurs at the lower frequency. The reason for this is that symmetric clearances will lower the system natural frequencies below those predicted for the linear system.

Figure 21 depicts the response of Case 6 for each mass, resulting from out of phase inputs. In this case, X_3 is a mirror image of X_1 while X_2 is essentially zero. Because of the symmetrical configuration of the system, this characteristic is to be expected. The results for this case are especially interesting since this configuration is closely related

Figure 19

Figure 20

Figure 21

to the single plane core model.

Results from Cases 1 through 6 show that the program can predict the response behavior of non-linear, multi-degree of freedom system. Future work will be concerned with parametric studies of the system behavior with non-uniform clearances, non-linear springs and non-linear damping. Once more complete information is obtained, actual system material and geometric properties will be introduced and forcing functions related to the expected seismic conditions will be employed. Computer code modifications to enable the entire scope of the parametric study, the evaluation of vertical block arrays and combined arrays will be pursued.

5. Materials (J. Chow)

The work in this program area covers four groups of materials:

(1) metals, (2) graphites, (3) PCRV, and (4) other materials which include control rods and thermal barrier insulation. Since inception of the program the effort has been directed toward the assessment and review of the state of art and to identify the critical areas pertinent to HTGR safety. The activities during this report period included attending the General Cooled Reactors at Gatlinburg in May 1974 and several meetings with the GA staff on the mechanical and corrosion properties of metals used in an HTGR. Also discussions were held with several persons of the Dragon Project and the AVR project. Wherever it is established that adequate information or data are lacking for structural integrity analysis, experimental programs will be initiated to obtain this data.

5.1 Metal Components

Our assessment and review indicates that many of the metals components proposed for use in HTGR primary coolant circuits are special alloys which have had only limited service experience. It is concluded that there is insufficient information available on long-term effects on all the highly alloyed materials such as Incoloy 800, Hastelloy B, Hastelloy X and Inconel X750. Furthermore, in most cases elevated temperature design data necessary for inelastic and creep-fatigue interaction analysis are not available. Although Incoloy 800 has been qualified for use in the ASME Pressure Vessel Code Cases 1331-8, 1592, and 1593, more information on its long-term behavior

is needed. It is judged that the mechanical behavior of these materials at elevated temperature must be better understood before the safety of the components made of these metals can be adequately analyzed. The immediate objective of this task is to identify any critical information that may be lacking for safety analysis and if this information is not being generated in the basic program, set up experimental programs to obtain these data. In our review it is clear that high cycle fatigue behavior of the metals used in the primary circuit is one of the areas where information is critically needed. Therefore, high cycle fatigue testing of Incoloy 800 at temperatures up to 1400°F, the most widely used metal in the primary circuit, has been initiated.

5.1.2 High-Cycle Fatigue Properties of Incoloy 800

To determine the cyclic behavior of a metal at elevated temperatures, both high-cycle and low-cycle fatigue tests are performed. A high-cycle fatigue test is load controlled where the alternate tension and compression stresses are held within the limit specified for the test. In a low-cycle fatigue test, the alternate strain is controlled and held constant throughout the test. In a high-cycle fatigue, the specimen is generally loaded below the gross yielding stress whereas in low-cycle tests, the alternate loading causes an appreciable plastic deformation. Each of these two fatigue testing techniques studies a different mode of failure. The design criteria used for the components establishes the type of test results that are needed. Low-cycle fatigue tests are being performed at

ORNL and GA; however, no high-cycle fatigue data are being generated on the HTGR materials.

High-cycle fatigue data are important and required in the design of equipment undergoing low amplitude vibrational stresses. Failures resulting from the high-cycle, low amplitude cyclic stresses generally originate at the surface of a component. Therefore, surface defects due to corrosion or oxidation could provide conditions for failures due to this type of cyclic stresses. Conditions for high-cycle fatigue failure could exist in the metal components of a HTGR. It has been shown that the surfaces of the metal are preferentially^(1,2) attacked at the grain boundaries by the impurities in the helium coolant stream and that vibrational stresses are present due to high velocity helium and rotating equipment.

BNL has three MTS closed loop electro-hydraulic testing units, which were previously used in high-cycle fatigue testing of LMFBR material, available for use in the HTGR program. During this report period these testing units have been set up for testing HTGR materials. These testing units can stress the specimen in a push-pull mode up to 40 cps. Tests can be run at elevated temperature and the testing units are being adapted to run in the HTGR helium environment.

High-cycle fatigue testing of Incoloy 800 in air at 1200, and 1400°F has been initiated. The data obtained to date are plotted in the form of alternating stress vs. cycles to failure (S-N curve) and are shown in Figure 22. Also shown in the figure are data of Incoloy 800 tested at 1100°F. These data were developed earlier at BNL under the support of the LMFBR program. The

Figure 22

material tested at 1100°F was obtained from Batelle, Columbus. The material being tested at 1200 and 1400°F was obtained from HEDL and it is material that was stocked for use in the AEC's irradiation effects program. All the material was solution annealed at 2100°F prior to the test.

The data on Incoloy 800 show that the endurance limits at 1100 and 1200°F are 32,000 and 27,000 psi, respectively. At 1400°F the material does not appear to have an endurance limit.

During the next report period we intend to continue testing in air to establish a base for comparison of tests to be made later in a helium environment. Also included in the testing program will be material (Incoloy 800) that was received from GA. A helium loop into which impurities can be introduced is being built for pre-exposing fatigue and creep-rupture specimens. Design of helium environmental chambers to be incorporated with the fatigue and creep-rupture testing equipment is underway.

5.2 Graphite Components

Assessment and review of data available on the H-451 near isotropic core graphite that is proposed for use in large HTGR's and the ATJ graphite for use in the core support column are continuing.

A log of the H-451 graphite (log 108, phase III, furnace 2) has been purchased from Great Lakes Carbon Company. Plans are being made to saw up the log and distribute material to the various groups that are studying the properties of this graphite. This log has a density of 1.76 g/cc, the total ash is 33 ppm and a tensile strength of 2,200 psi.

Attempts have also been made to obtain ATJ graphite core support post materials that would meet the GA specifications for use in our test program. To date we have not been able to obtain a relevant specification for the support post materials and information on the purchase of this material.

Our preliminary assessment and review indicate that integrity of the core support post appears to need further study and evaluation. The posts may or may not be made of ATJ graphite and are about 7 feet high and about 6 inches in diameter. Although the design stress is relatively low (about 500-600 psi) and the safety factor is high, these posts can not readily be inspected and are not replaceable over the reactor life of 30 to 40 years. The ATJ graphite is fairly high in impurities and the corrosion oxidation rate could be significant if oxidants can get to the lower plenum of the reactor during periods of steam ingress due to failure of steam generator tubes.

A subcontract is being negotiated with Aerospace Corporation for study on the fracture of graphite. Aerospace Corporation has developed a technique⁽³⁰⁾ and an analytical model for the fracture of graphite based on microstructural parameters. The proposed studies will cover the ATJ and H-451 graphites before and after exposure to helium with added impurities.

5.3 Prestressed Concrete Reactor Vessel (PCR/V)

Assessment of the technology and the review of the state of the art is continuing. A subcontract was given to the Radiation Division of DAS at

BNL to carry out a review of the concrete properties.

5.4 Other Materials

Except for a very preliminary evaluation of the problems involved with the thermal barrier cross section and on the insulation materials, no activity has been initiated in this area.

6. Safety Instrumentation and Control Systems

6.1 Primary Coolant Monitoring (G. Bennett, H. Isaacs)

Low levels of impurities may lead to oxidation of graphite, metal surfaces and fission products, during the normal operation of HTGR's, and possibly alter their behavior and predictability. High levels of impurity concentration would follow a steam generator or a depressurization accident. At present there may be a need to closely determine impurity concentrations and their characteristics in order to define the effects due to impurities, and to activate alarm systems (see Section 2).

Primary coolant impurity monitoring, for "moisture" and other contaminants, is essential safety instrumentation for HTGR's. A proven system which is capable of sensitively and accurately identifying a wide range of impurities in the helium coolant is a mass spectrometer. The principal advantages of a mass spectrometer are:

- (1) Speed of response; scans through a wide mass range require $\ll 1$ second.
- (2) Applicable to a wide range of contaminants including "moisture", hydrocarbons, particulates, fission products.
- (3) Identification of chemical composition of impurities.
- (4) Compatible as an extension to the gas chromatography typically used for reference analysis in existing and proposed HTGR's.

Thus, MS has a shorter response time than GC, which is especially useful in dynamic studies. Both systems have limitations in uniquely identifying sample components. Mass spectrometers of very high resolution can distinguish between molecules of nearly identical weight (e.g. CO and N₂); more modest resolution requires analysis of the molecular fragments produced by the ionization process. Similarly, different substances can have identical retention times in a chromatographic column, requiring identification, e.g. by analysis of retention time vs. the number of structural units in the molecules of the substances in question. The combination of both systems offers a maximum in sensitivity and resolution.

Mass spectrometry is not suggested as a primary safety instrumentation system, but rather as a suitable broad range reference analyzer for study of coolant impurities, and for testing of other instruments and procedures.

The application of mass spectrometry to HTGR coolant analysis presently has three components:

- (1) An in-depth design study of MS systems for this application including types of mass analyzers suitable, vacuum systems, sampling systems, gas separators, continuous vs. batch processing, limits of detectability, and the GC-MS combination. The result of this effort will be in a BNL report.
- (2) An experimental coolant impurity analysis system is

to be set up at Fort St. Vrain. The initial effort is a mass spectrometer for analysis of moisture signal gases, e.g. CO and CO₂. The MS system will be preceded by an enriching separator to increase the concentration of impurities in the sample stream. The Fort St. Vrain station has been visited to explore the physical requirements for the system and negotiations with the plant management are nearly complete.

- (3) Design, assembly and operation of a small laboratory scale gas analyzer system, primarily to test some of the procedures which may be used in the Fort St. Vrain effort. At present a VEECO model GA-4 vacuum gas analyzer has been set up with a Varian 500 liter per second ion pump for high vacuum. The analyzer employs a 60° dipole mass analyzer with a simple ionizer and ion collector. Sensitivity is specified as better than 10⁻¹³ torr (nitrogen), impurity detection to 2 parts per million, mass range 2 to 300, resolution equals 150 at mass 75, scanning speeds down to 10 milliseconds per sweep, the analyzer is bakeable to 400°C to minimize system background. Neither the spectrometer nor the vacuum system are ideally suited for the purpose but they are available, and will be

useful in developing improved systems. A gas inlet system is presently undergoing design to allow known volumes of various gas mixtures to be admitted to the spectrometer for calibration studies. Only the most preliminary spectra have been obtained thus far with the present set up. Ultimate pressure obtained in the vacuum systems is 3×10^{-8} torr.

An oxygen activity or oxygen potential meter which uses 15 w/o yttria doped thoria as an electrolyte in an oxygen concentration cell has been designed for monitoring helium. These measurements can be used to determine the relative oxidizing nature of the helium relative to either graphite or metals. The meter may be used in two modes. In the first the actual oxygen potential of the gas will be measured while in the second the potential will be measured after the impurities in the helium have equilibrated with graphite at the measuring temperature. The latter measurement will give a quantitative value for the total oxygen in helium. These measurements and their differences give a measure of the amount of graphite the gas is capable of oxidizing and also can be simply related to the hydrogen to water ratio. In general, the response of these meters is extremely rapid and offers a simple device for monitoring and predicting the behavior of oxygen bearing species in helium.

6.9 Post Accident Monitoring (G. Bennett)

The Summit Generating Station preliminary safety analysis report⁽³¹⁾

has been reviewed to assess the adequacy of specified instrumentation in light of possible post accident monitoring requirements.

One guiding principle which may be used in considering the operating range of post accident monitoring instrumentation is the extreme values of measured parameters which allow some structural integrity to be retained. For example, the present upper limit on PCRV pressure measurement channels is specified as 1000 psig. However, the design safety limit is given as 1131 psig, the minimum structural safety margin is given as a factor of 2⁽³²⁾ so that the PCRV can sustain at least 1552 psig, and ultimate strength analysis of similar PCRV's gives a minimum failure pressure of 2000 psi⁽³³⁾. Thus, a reasonable upper limit for the operating range for post accident pressure monitoring instrumentation for the PCRV is 2000 psi. In principle these post accident monitor channels can serve dual roles as redundant members of the normal primary coolant instrumentation.

Note that the relative probability of attaining such pressures with PCRV is neither assessed nor pertinent here. Pressure relief valves are expected to limit maximum helium pressure to 776 psig. However, the possibility of generating extreme pressures cannot be excluded.

Range limits for core temperature measurements applied to post accident monitoring may be similarly established. Some of the temperature bench marks for core components are listed in Table 8 (adapted from Ref. 32).

Table 8

<u>Temperature °F</u>	
1360-1415	Avg. moderator temp.
1575-1635	Avg. fuel temp.
2000	Incoloy 800 loses structural strength (control rods fail)
2100	Low carbon steel loses structural strength (flow control orifices fail)
2300	321 stainless steel loses structural strength (top plenum elements fail)
2500-2700	Steels melt
>3000	Fuel particle coatings fail
4300	Melting point of fuel and boron-carbide eutectic in control rods
6600	Sublimation of graphite

Above the ~~maximum~~ temperature listed graphite sublimates. The core and support structures evaporate. In that case there is no need (nor practical mechanism) for thermometry. For the Summit Station primary coolant temperature is to be monitored at the inlets to the steam generator and auxiliary heat exchanger, and the main circulator and auxiliary heat exchanger outlets. The specified upper limit for the inlet channels range is 2000°F, and 1000°F for the outlet channels.

The data of Table 8 suggest that a practical post accident monitoring channel could be provided at each site by allowing for range limits of 2100

to 2300°F. This limit implies the possibility of deformation or blockage of the flow control orifices. This extension in range is only moderate for the inlet channels. Justification of this range limit for post accident monitoring of the main circulators and auxiliary heat exchangers can be posited for the condition where secondary cooling is inoperative. Thus, such post accident instrumentation could serve to prevent or minimize damage to essential safety systems, such as the auxiliary heat exchanger circulators.

Consistency now requires that post accident temperature measurement channels be operative over the operating range of the pressure measurement channels and vice versa.

In conclusion, it appears that structural integrity may be a useful parameter against which to measure the range limits for post accident monitoring channels.

Analysis of post accident monitoring will continue with consideration of pertinent accident scenarios.

9. Phenomena Modeling and Systems Analysis (C. Sastre, L. Epel, J. Skalyo)

9.3 Component and Subsystem Characteristics (L. G. Epel)

For purposes of studying the behavior of major components of the HTGR under anticipated transient and accident conditions mathematical models, usually translated into computer programming language, are generally used. Thus, if boundary and initial conditions are known for a particular component (such as the auxiliary circulator, for example) the response of the component can be simulated without the need to analyze the other parts of the reactor system.

On the other hand, the analysis of the overall HTGR plant requires the modeling of its major components, including the ability for these components or subsystems to interact with each other (see Section 9.5). In either case then, models of HTGR components and subsystems must be constructed for the purpose of safety studies. Examples of such models in the form of codes used by General Atomic include DMOR (steam generator), CORCON (core and cavity heat transfer), IONEXCHANGER (tank radiation sources), and RECA (primary loop behavior).

Two major components specific to the HTGR design are the once-through steam generator and the steam turbine-driven helium compressor. These two components are unique in that they both serve to couple the primary loop to the secondary loop; the former by the transfer of thermal energy and the latter by the transfer of mechanical energy. Because of their importance to the overall systems code (see Section 9.5) these two components were selected

for study and modeling during the first half of fiscal year 1975. They will be incorporated into the systems code in the calendar year 1975.

Steam Generator Heat Transfer and Pressure Drop

The HTGR steam generator consists of a main steam and a reheater bundle. The main steam bundle is composed of an economizer section, an evaporator section and a superheater section. Helium flows on the shell side while steam/water is on the tube side. The bundles utilize helical, multistart tube coils to achieve a high ratio of tube surface area to steam generator volume. The tube coil geometry consists of equal-length tubes arranged in columns of equal height and approximately equal helix angle. The longitudinal pitch spacing is constant within each row but varies between rows. The transverse pitch spacing is constant over the full bundle diameter.

It has been established in various works that the heat transfer and pressure drop characteristics in coiled tube geometries are quite different from those of straight tubes. Considerable effort has been expended towards characterizing and quantifying these differences. The literature applicable to such systems has been reviewed and the best of the available correlations have been programmed in a set of Fortran routines. These routines are basically concerned with predicting the pressure drop and the heat transfer coefficient for both the steam/water side and the helium side of the steam generator/reheater. The sources of the correlations used are presented in this section.

Steam/Water Side - A bibliography of almost 100 references in the open literature on the subject of curved duct heat transfer and pressure drop is given in Appendix I. This bibliography was compiled from References 34 through 37*, computerized literature searches of the Oak Ridge Nuclear Information Center and the NASA/RECON system and many of the heat transfer literature reviews published by Eckert, et al, in the Journal of Heat and Mass Transfer. Of particular interest and importance is a review of this same subject by Hopwood⁽³⁸⁾. Many of the correlations used for the tube side were adapted based on his recommendations. Another interesting paper with many references is that of Gorlov and Rzaev (Reference 41). These two references provide a good introduction to the subject of fluid flow and heat transfer in curved ducts.

The first value needed for the calculation of friction factors in coiled tubes is the Reynolds number at which transition from laminar to turbulent flow occurs (critical Reynolds number). This is taken as 2100 for straight tubes and is computed as a function of the tube-to-coil diameter ratio according to the model of Srinivasan (Reference 39). Next, the single phase friction factor in straight tubes is computed. The procedure used is given by Kerenskii (Reference 42) and evaluates the friction factor in the laminar, transitional and turbulent regimes as a function of Reynolds number and relative roughness. The procedure involves the use of a continuous

*References cited in this section are listed at the end of this report and are not to be confused with those in Appendix I.

relationship between friction factors and Reynolds number over all three flow regimes, utilizing the Gaussian probability function as described in Reference 42.

The third step is to modify the straight tube friction factor for that in a coiled tube. This correction depends on Reynolds number, the tube-to-coil diameter ratio, and the relative roughness of the tube. If the flow is laminar, the model of Ito (Reference 40) is used, while for turbulent flow the model of Gorlov and Rzaev (Reference 41) is utilized. The fluid Prandtl number ratio, evaluated at the wall and at the fluid bulk temperature, is used to correct for the effects of a heated wall on the computed friction factor. Finally, the two-phase friction multiplier is calculated based on Hopwood's (Reference 38) recommendation. Use is made of Smith's (Reference 45) void correlations to derive the multiplier for two-phase flow as a function of steam quality and the liquid-to-vapor density ratio. This multiplier was found to coincide closely with the revised Martinelli-Nelson values as presented by Thom (Reference 46).

Turning now to the computation of the heat transfer coefficient on the tube side, the distinction between Reynolds numbers above or below the critical Reynolds number computed before must again be made. For values of Reynolds number below the critical, the correlation of Akiyama and Cheng (Reference 43) is used. This gives the single phase heat transfer coefficient as a function of Reynolds number, Prandtl number, and tube-to-coil diameter ratio. For single phase turbulent flow, the correlation of Seban and

McLaughlin (Reference 44) is used. The heat transfer coefficient depends on the same three parameters as in the laminar flow situation. The heat transfer coefficient is derived from the computed Nusselt number by using fluid properties at the film temperature in this model; i.e., the average of the wall and fluid bulk temperatures.

The heat transfer coefficient in the two phase region is modeled on a procedure outlined by Hopwood (Reference 38). This algorithm has not been confirmed by experimental data because of a lack of these at the appropriate conditions. Thus, experimental data from the HTGR Steam Generator Design Verification and Support Programs would be helpful in evaluating this method. At any rate, the parameters are the same for the two phase heat transfer correlation as for the single phase situation, with the addition of the fluid quality and the vapor-to-liquid density ratio as parameters in the correlation.

Helium Side - The description of fluid flow and heat transfer on the helium side is, of course, much simpler than for the steam/water side. The complicating factor for the shell side is the complex flow geometry. The geometry of the steam generator as seen by coolant gas is an irregular in-line tube pattern. Heat transfer and pressure drops in such geometries can only be determined accurately from experimental observations of tests conducted with the exact geometry. However, approximate calculations can be made based on the existing data for flow over staggered and in-line banks of tubes. Most of this data are reported in References 47 through 51.

The friction factor for the flow of helium over banks of tubes is obtained from the correlations of Gunter and Shaw as given by Knudsen and Katz (Reference 51) for both the laminar and turbulent flow regimes. The pertinent variables are Reynolds number based on a volumetric equivalent diameter, the viscosity ratio (where the viscosities are evaluated at the wall and bulk temperatures), and three correlation constants.

The helium heat transfer coefficient is found from Colburn's correlation as given by Knudsen and Katz (Reference 15). The Nusselt number is obtained in this correlation as a function of Reynolds number based on tube diameter, the film temperature, the helium pressure, and two correlation constants. The helium film properties are calculated at a temperature averaged between the wall and the bulk temperature.

Turbo-Compressor Model

The mathematical model of the steam turbine-driven helium compressor is based on the availability of performance maps for both the turbine and the compressor. The actual algorithms used to model the turbo-compressor transient behavior will depend on the form in which these maps will be made available from GA. Traditionally, compressor performance characteristics are given as plots of dimensionless pressure rise versus corrected flow rate with dimensionless speed as a parameter. Superimposed on such a map are contours of constant efficiency, or oftentimes the efficiency is given in a second plot having the same abscissa and parameter as the pressure rise plot. Additionally, a dimensionless form of compressor power may be represented on a third plot in

the same way as the first two variables are mapped. The information represented in such mappings, together with consideration of energy balances across the compressor, allow for a complete thermodynamic description of the helium conditions into and out of the machine.

Similar data are available for the turbine, and given the mass moment of inertia of the rotor of the turbo-compressor, the dynamics of the coupled machine can be described.

The precise algorithm to be used in the systems code awaits the actual form of the turbine performance characteristics. In many representations of these data, an iterative scheme is needed because an unknown variable (typically the isentropic enthalpy drop) is imbedded in one or both of the coordinates. One such algorithm has in fact been developed and successfully tested for use in the systems code under development at BNL, but it is hoped that a better mapping of turbine characteristics, if available, will allow for a more simplified and speedier technique to be incorporated into this program.

9.4 Physical and Chemical Process Models (C. Sastre)

The effort on modeling of the reactor dynamics concentrated on evaluation of the GA code BLOOST-7. This code serves to analyze reactivity addition accidents initiated by control rod motions, loss of nuclear poisons, etc. Given reactivity changes corresponding to a postulated core configuration change, as computed by other codes, and temperature coefficients of reactivity also computed by other codes, BLOOST-7 can compute reactor power levels and core temperatures as a function of time. As the code does not include the

balance of the plant, then the relationship between inlet and outlet coolant temperature, and variations in coolant flow rate need to be supplied in the form of simple relations or tables.

In its present form BLOOST-7 represents the reactor kinetics by a point reactor model and the effect of fuel and moderator temperatures on reactivity by means of temperature coefficients, not necessarily constant. The relevant core temperatures are computed by means of a restricted version of TAC 2D, a general purpose two dimensional heat transfer code. The heat transfer model is restricted to cylindrical geometry and hence the heat conduction in the fuel and moderator is solved in a cylindrical "unit cell" for an average channel running the length of the reactor.

BLOOST-7 has been used at ENL to represent a 1000 MWe reactor, with reference parameters as given in the Fulton Station PSAR, for the purpose of exploring parameter sensitivities in the neighborhood of normal operating conditions. Included in the studied parameters were thermal conductivities, gaps between fuel stick and graphite block, radiation sensitivity and flow rates.

The convergence of the temperature calculation is dependent on the time steps specified in the input stream, and the proper selection of such steps was found to require considerable amounts of experimentation and judgment.

The version of TAC 2D used in BLOOST-7, being two dimensional, takes into consideration the axial heat conduction along the "unit cell" which

happens to be quite negligible in comparison with the radial heat flow. This suggests that it would be possible to simplify the heat transfer algorithm or, perhaps, trade the unnecessary programming detail for a better description of the heat conduction in the graphite block in the plane perpendicular to the coolant flow as it is done in OXIDE-3, for instance. However, it is premature to judge whether the corresponding coding effort is justifiable.

TAC 2D - Two Dimensional Heat Conduction Code (A. Romano, S. Nicolosi)

The digital computer code, TAC 2D, was developed at General Atomic Company for determining temperature solutions of two-dimensional thermal systems encountered in an HTGR. The code^(56,57) was made operational on the BNL CDC6600 computer by Science Applications, Inc. under contract to BNL. The user's manual sample problem, a textbook problem with a known solution, and several problems pertinent of HTGR safety have been run. The results of these calculations will be discussed.

The basic heat transport equation to be solved is:

$$\nabla k \nabla T + S = \frac{\partial}{\partial t} \rho C T$$

where k = thermal conductivity

T = local temperature

S = volumetric heat generation rate

ρ = density

C = specific heat

t = time.

The heat transport equation is replaced by its equivalent set of linear finite difference equations. This set of equations is then solved by an implicit alternating gradient method⁽⁵⁸⁾.

The problem whose temperature distribution as a function of time has to be obtained is set up as a set of boundaries and mesh points. Heat transfer across internal boundaries may be by radiation, conduction and convection. Heat transfer across external boundaries is by convection only, although insulated boundaries, constant temperature boundaries, convection and radiation can be accommodated by the proper specification of the thermal parameters of the coolants in the external boundaries.

The code is designed to treat transient and steady-state, two-dimensional heat transfer problems. Geometrically, the problem may be defined in rectangular, circular or cylindrical coordinates. The code can accept material, gas and coolant parameters in a functional form. The thermal parameters may be functions of temperature, time, and location. The number of different materials, gases and coolants are limited to 15 each. There are also certain limitations on the number of mesh points, internal gaps and coolant boundaries.

All of the thermal parameters are defined as FORTRAN IV arithmetic statement functions. This requires recompilation of four subroutines, namely MADATA, FLODAT, SPEC and HC. These subroutines recompile new input for the materials properties and coolant parameters.

The TAC 2D program input and output is illustrated in the user's

manual⁽⁵⁶⁾ by means of a sample problem in which the steady-state temperature distribution for a helium circular blower disc is calculated. The sample problem is complex and employs many of the options and subroutines of the program. Successful execution of the sample problem was achieved on the CDC 6600 computer.

The running program was verified by solving for the steady-state temperature distribution of a simple textbook heat transfer problem⁽⁵⁹⁾. The correct solution for this sample problem was also obtained.

The program has been recently made operational on the new BNL CDC 7600 computer. Several minor modifications were required. Many of the calculations made on the 6600 have been repeated on the 7600 for verification of the program.

HTGR Core Symmetrical Triangle

A heat transfer problem pertinent to HTGR safety was set up for solution with the TAC 2D program. The core of an HTGR is made up of many hexagonal fuel elements (blocks) each of which contains fuel holes and coolant holes in a symmetrical pattern, as shown typically in Figure 23. The problem examines the heat transfer from a quadrant of fuel through the gas gap and graphite web to a quadrant of helium coolant. A "dummy" coolant having adiabatic parameters is introduced in order to accommodate circular geometry (Figure 23).

Steady-state and transient solutions of heat transfer problems involving normal and off-normal conditions were made. The emphasis of the

SYMETRICAL TRIANGLE
OF FUEL BLOCK

ASSUMED GEOMETRY

GRID LINES

Figure 23

study was to determine which of the core thermal parameters had the most significant effect on the fuel centerline temperature and thereby on fuel particle coating failure.

The operating conditions shown in Table 9 were used as input to determine the "normal" temperature profile across the symmetrical section chosen. The results shown in Figure 24 are in good agreement with GA data⁽⁶⁰⁾. A maximum He bulk temperature of 1400°F was chosen for these calculations although the value was varied in other calculations for comparison. A parameter sensitivity analysis was then performed by arbitrarily varying one or two of the parameters listed in Table 9. Fuel conductivity, heat rate, and coolant flow rate are very sensitive parameters. Gap emissivity, gap thickness, gap gas conductivity and graphite conductivity were found to be less sensitive parameters. An upset condition involving both an increase in gap thickness and a lower gap gas conductivity (due to the presence of Xe, Kr, etc.) which might occur near end of life of fuel was also examined. The condition was found to be only moderately sensitive to fuel centerline temperature. Results of the parameter sensitivity analysis are summarized in Table 10. In examining the results one should recall that these are steady-state calculations assuming no corrective action has taken place. The results do point out, however, that very local undiscovered hot spots could occur due to high heat flux or a partly blocked flow channel which would lead to fuel particle damage and enhanced fission product release. Although fuel

Table 9

Input Data for TAC 2D Solution of Symmetrical Fuel Block

Under Normal Operating Conditions

Maximum Helium Bulk Temperature	-	1400°F
Volumetric Heat Rate (Fuel Volume)	-	44 kw/liter
Helium Flow in Coolant Channel	-	26 lbs/hr
Helium Heat Transfer Coefficient	-	300 Btu/hr-ft ² °F
Graphite Block Conductivity	-	16 Btu/hr-ft °F
Fuel Stick Conductivity	-	4 Btu/hr-ft °F
Helium Conductivity	-	0.21 Btu/hr-ft °F
Helium Heat Capacity	-	1.24 Btu/lb °F
Emissivity in Gap	-	0.7
Gap Thickness	-	0.003"

Figure 24

Table 10

Summary of Results of Parameter Sensitivity

Analysis of Symmetrical Fuel Element

<u>Problem No.</u>	<u>Description of Data Input</u>	<u>Maximum Fuel C.L. Temp. °F</u>
5	Reference	2004
11X	Zero emissivity in gap	2019
13	Gap increased to 0.03"	2066
6	Upset gap condition	2086
8X	Graphite $k = 8.0$ Btu/hr-ft °F	2120
12	Fuel $k = 2.0$ Btu/hr-ft °F	2192
3X	Heat rate doubled	2607
7X	Heat rate tripled	3326
10X	Coolant flow = 1.0 lbs/hr	5267
8	Upset gap condition and graphite $k = 8$	2202
9	Upset gap condition and fuel $k = 2$	2297

failure by kernel migration occurs progressively over all operating temperature ranges, abrupt failure of irradiated particles (near end of life) is greatly enhanced at temperatures $\geq 2900^{\circ}\text{F}$.

Another type of calculation which can be performed with the TAC 2D program is a transient heat transfer calculation with varying time history input data. A simulated loss of cooling transient was programmed using the geometrical configuration discussed above. Normal operating conditions (Table 9) were used as input. Coolant flow was then assumed to drop to 1/26 of normal flow and go undiscovered for 100 seconds. The results, shown in Table 11 indicate that fuel failure temperatures are reached in ≈ 100 seconds.

A model of a TRISO fuel particle has been developed at BNL which can be treated by TAC 2D. In this program, one-half of a spherical TRISO particle is modeled in circular geometry as a right circular cylinder (with a semicircular base). The vertex of the model corresponds to the center of the particle (Figure 25). The dimensional data are that cited by GA⁽⁶¹⁾ for the TRISO reference particle. In this model the particle is surrounded by a layer of graphite matrix 0.0012 inches thick. The matrix is the environment of the particle in the fuel stick. The temperature of the matrix is specified to be at the temperature of the fuel rod by surrounding it with an isothermal coolant that has a high flow rate (10^6 lbs/hr) and a high heat transfer coefficient (10^6 Btu/hr-ft²-°R). In this manner the temperature of the matrix is forced to be the temperature

Table 11

Temperature Rise Due to Loss of Cooling
Without Corrective Action

<u>Time, Sec.</u>	<u>Coolant Temp. °F</u>	<u>Max. Fuel Temp. °F</u>
20	1410	2108
40	1414	2339
60	1418	2559
80	1422	2775
100	1426	2989

Figure 25

of the surrounding coolant.

An average heat generation rate of 4.4×10^7 Btu/hr-ft³ was used for the fuel kernel. The thermal conductivities used were based upon GA data⁽⁶²⁾:

Buffer	4.37	Btu/hr-ft °F
PyC	4.37	Btu/hr-ft °F
SiC	7	Btu/hr-ft °F
Matrix	4	Btu/hr-ft °F
Fuel	4	Btu/hr-ft °F

The two cases studied were those of a symmetrical fuel particle with no internal gaps present and an extreme amoeba effect which was modeled by replacing one-quarter of the inner PyC layer with a Kr-Xe stagnant gas gap.

In each case studied the temperature gradient within the particle was very small (approximately 1°F from the center of the kernel to the surrounding matrix). The temperature gradient within the isolated particle is 94°F/inch as compared with the typical fuel stick gradient of 630°F/inch.

Heat transfer models of the HTGR thermal barriers are presently being set up for calculation using the TAC 2D program code. The model will be used to estimate the effects of postulated accident transients on the thermal barriers including component parts such as the liner, the cooling coils, concrete, etc.

9.5 Consolidated Plant Models - TFR Computer Program (L. G. Epel)

The analysis of overall plant performance under anticipated transient and accident conditions involves the simulation of major primary and secondary loop components and their interaction with each other, the control system, the plant protection system and operator actions. Such simulations are usually done with computer programs that, to varying degrees of accuracy, and for various ranges of operating conditions, describe component behavior as they interact with the overall plant as a whole.

The program that is used for such analyses at General Atomic is the proprietary code TAP. This code was requested from GA at the beginning of this fiscal year but has not yet been received. General Atomic also utilizes other codes that simulate major portions of the overall plant for specialized conditions (e.g. RATSAM calculates the primary loop response to rapid depressurization) but has no integrated plant transient response code besides TAP for HTGR Steam Turbine Power Plant analyses*. There is a plant transient analysis code in use in the European community for the DRAGON program named TUBER, but as far as can be ascertained it is not applicable to HTGR systems for the purposes described above.

In the United States the proponents of various reactor systems

*Another GA code, REALLY, is used for very short term analyses of HTGR Gas Turbine Power Plant transients. Such plants do not have secondary loops containing steam/water or steam plant components such as steam generators, steam turbines, condensers, etc.

have developed systems codes that model portions of their particular type of plant. Thus, for example, Westinghouse has a proprietary code for simulating transient behavior in the Fast Flux Test Facility called IANUS*. Similarly, blow down in pressurized water reactors is studied using the Westinghouse code FLASH. Among the best known of such computer programs for water reactors is the Idaho Nuclear Corporation series of RELAP codes. One of these, RELAP3 (mod 62), has been greatly extended and improved at Brookhaven National Laboratory during the past two years and is now used at BNL to simulate primary and secondary loops of both boiling and pressurized water reactor systems over a large range of anticipated transient and accident conditions. A recent version of the Brookhaven-modified RELAP3 is described in reference (63). This version has a number of features beyond those of the other versions that make it attractive for plant transient response studies; some of these are:

- (a) Automatic time-step selection to minimize computation time.
- (b) Capability to vary fuel rod gap size during transient.
- (c) Extended restart capabilities.
- (d) Automatic calculation of reactivity weight factors.
- (e) Capability to modify and refine steam table mesh.
- (f) Extended component trip sequence options.

*A more recent version of this proprietary code, modified to be more applicable to the LMFBR Demonstration Plant, is called DEMO.

(g) Capability to simulate various contributions to reactivity feedback separately. For HTGR applications, for example, the moderator temperature coefficient would be separated from the Doppler feedback.

The early RELAP3 codes achieved some degree of detail by expending relatively large computer times in the solution of given problems. This objection was largely removed by utilizing the implicit method of solution in mod 62 but even with this improvement some transients required relatively large computer running times. The Brookhaven-modified version improves on the mod 62 version by utilizing a time-step optimizing algorithm that allows very significant computer time savings beyond that afforded by the implicit scheme.

Since the RELAP3 code as modified at Brookhaven is an excellent and available, non-proprietary systems code, it was chosen as a starting point for developing a systems code applicable to the HTGR. This is an alternative to starting a completely new program, which it was felt, would require at least several man-years and would, to a certain extent, involve redoing what has been done before. The RELAP series of codes represents many tens of man-years of effort and the Brookhaven-modified version of RELAP3 represents many man-years beyond that. Since this latter version was readily available and already included many of the features desirable in a consolidated plant computer code and since the objection of large computer running time had largely been overcome, it was made the basis for

the TFR (Two Fluid RELAP) code.

Description of Programming Done

In principle, the algorithms that are built into RELAP to solve the conservation of mass, momentum and energy equations are applicable to any fluid, not solely to steam/water. Thus the general structure of the code was maintained, but provisions for accommodating two different fluids simultaneously and for differentiating between the two fluids at any point in the program had to be designed. This was the major concept from a programming point of view. Once such a design was implemented the necessary fluid-dependent characteristics had to be inserted in the appropriate subroutines in order to specialize the code to particular fluids. For example, the block data subroutine containing enthalpy and specific volume as a function of pressure and temperature of steam/water had to be supplemented with data for helium. Beyond that, subroutines that implicitly assume a given fluid (steam/water) had to be amended to reflect the impact of the second fluid (helium). Examples of this include the subroutine that computes heat transfer in the core, the one that predicts two-phase flow out of a rupture, and the subroutine that determines the two-phase friction multiplier.

The computer programming necessary to convert the BNL version of RELAP3 to a code which could accommodate two dissimilar fluids simultaneously was largely done with funds from the Fast Reactor Safety Division at BNL. Cooperation between that Division and the HTGR Safety Division was fostered

early in the fiscal year when it was apparent that both safety groups could utilize a two-fluid systems code; one specialized to helium and steam/water, the other to sodium and steam/water. In this way a sizeable programming effort was conserved. Specializing TFR to helium by incorporating helium thermo-physical properties and by rewriting those subroutines whose coding depends on the specific fluid characteristics was, of course, done entirely in the HTGR Safety Division.

With these three major steps (change the code structure to accommodate two fluids simultaneously, incorporate a second set of thermo-physical properties, and rewrite certain subroutines to reflect the addition of a second fluid obeying different physical laws) TFR became a program that could simulate gas-cooled, steam-generating reactor systems in a general way. To test the code, several problems were run on the CDC 7600. The number of volumes in these problems ranged from 1 to 14, the helium pressures ranged from 2 to 80 atmospheres, the helium temperatures ranged between 70°F and 1400°F and the time steps ranged between 1 millisecond and 15 milliseconds (although none of these represent any inherent limits). The types of transients investigated included pump trips and simplified blowdown accidents and in each run the code appeared to function properly. These tests were, to a certain extent, abstract with respect to an actual HTGR system because of the manner in which the compressors and steam generators were simulated. The development of models for specific components of the HTGR (described in section 9.4) and their incorporation into this

program will specialize the code to truly represent an HTGR. The following describes some of the salient changes made to portions of the program to take into account the introduction of helium as the second fluid.

BLOCK DATA DATA1 (thermodynamic properties)

This block data subroutine contains, among other thermodynamic properties, tabulations of enthalpy and specific volume as a function of pressure and temperature. The helium properties were computed using relationships discussed in reference (64) and were tabulated for 17 pressures ranging from 14.504 psia to 2000 psia. At each of these pressures, 31 temperatures, ranging from 32°F to 3000°F were selected for computation of the desired properties. Compressibility effects in helium were also included, even though these can essentially be neglected in the temperature-pressure ranges likely to be investigated in connection with HTGR safety.

Briefly, the specific volume was obtained from

$$v = (1 + Bp) \frac{RT}{p} \quad (28)$$

where the symbols have the usual meaning and B, the second virial coefficient is evaluated as

$$B = 1.13T^{-4/3} - 2370T^{-3} \quad (29)$$

In Equation (29) the temperature is in degrees Kelvin and B is measured in units of l/bar.

The specific enthalpy is given by the expression

$$h = 5.196 T_c + 3.13 (T^{-1/3} - 4715T^{-2}) p - 0.285 \quad (30)$$

where T_c is in degrees centigrade, T is in degrees Kelvin, p is in bars and h is measured in units of kilojoules per kilogram. Equation (30) is based on zero enthalpy at 32°F and 14.504 psia (1 bar).

The structure of the tables in this subprogram lends itself to rapid and easy updating. This will make it convenient to change any of the numerical values in these tables in the future, or of extending the tables, or refining the mesh, if this is ever warranted.

SUBROUTINE QUICKY (pressure derivatives)

The main purpose of this subroutine is to compute the values of two partial derivatives for the fluid ($\partial p / \partial u$) at constant p and ($\partial p / \partial p$) at constant pu , to facilitate the implicit integration scheme performed in subroutine NIFTE. The calculation of these two derivatives becomes almost trivial if it is assumed that the perfect gas laws can be applied to helium. This assumption causes an error of about one percent in helium properties under the most severe conditions encountered in the helium tables. Furthermore, studies have shown that the results of the systems calculations are quite insensitive to the accuracy of these two derivatives in any case.

If the equation of state for a perfect gas

$$pv = RT \quad (31)$$

is used in conjunction with the assumption that the volumetric specific heat of helium is a constant, i.e.

$$u = u_0 + c_v(T - T_0) \quad (32)$$

then it is a simple matter to arrive at

$$[\partial p / \partial u]_p = \frac{p}{c_v T} \quad (33)$$

$$[\partial p / \partial p]_{pu} = RT_0 - \frac{Ru_0}{c_v} \quad (34)$$

In relation (Equation 32) the constants u_0 and T_0 can be chosen arbitrarily and independent of one another. However, in order to make it and the equation of state consistent with the datum of zero enthalpy base at 32°F and 14.504 psia, u_0 is determined to be -244.11 Btu/lb at $T_0 = 32^\circ\text{F}$. The derived expressions for the two partial derivatives under the foregoing approximations are seen to be very simple and therefore are calculated in TFR whenever needed, rather than tabulated and found by interpolation as was done in the original program with steam/water.

SUBROUTINE LEAK (choked flow)

This subroutine originally (i.e. in the steam/water code) was written to compute the flow rate of steam/water through junctions connecting any two volumes or out of an opening in the pressure boundary. It was based on experimental work done by Moody and includes the limiting mass flow as defined by his two-phase choked flow model described in reference (65). Briefly, the choked flow is computed based on stagnation pressure and stagnation enthalpy in the source volume and, if it is less than the computed inertial flow, is chosen to represent flow for that junction.

While this is an acceptable model for the steam/water side, choked flow in helium is a much simpler phenomenon to describe analytically.

Therefore, the compressible gas laws were utilized for helium at all flow junctions and if the critical pressure ratio was exceeded at any junction the helium mass flow was computed from the simple formula

$$w = CA \sqrt{\frac{p}{T}} \sqrt{\frac{kg}{R} \left(\frac{2}{k+1} \right)^{\frac{k+1}{k-1}}} \quad (35)$$

where w = mass flow rate

C = discharge coefficient

A = area of the opening

p = stagnation pressure

T = stagnation temperature

k = ratio of specific heats, c_p/c_v

g = gravitational constant

R = specific gas constant

If it is assumed that the stagnation pressure and temperature are closely approximated by the thermodynamic pressure and temperature, as is commonly done, the choked flow rate is conveniently represented by a constant times the pressure divided by the square root of the absolute temperature.

If the critical pressure ratio is not exceeded, the revised programming considers the mass flow rate through a junction to equal the computed inertial mass flow.

SUBROUTINE HTRC (core heat transfer)

The original coding of this subroutine for steam/water was for the purposes of determining the heat transfer mode in the core, estimating the heat transfer coefficient in the core and approximating the critical heat

flux there. There were seven modes of heat transfer considered and correlations for the heat transfer coefficient for each mode was built into the subroutine. In addition, the critical heat flux was computed in six ranges, depending on pressure and mass flux. All of this is, of course, inappropriate for the HTGR core and a simple form of Nusselt's equation was coded into TFR instead. The particular one chosen is given in reference (66) and is

$$Nu = 0.020 Re^{0.8} Pr^{0.4} \quad (36)$$

In (Equation 36) the properties are those appropriate to the bulk helium. Although the convective transfer coefficient in the core is not believed to be critically misrepresented by Equation (36), changing the present correlation to a more suitable one in the future, should the need arise, is easily facilitated by the structure of the present subroutine.

Conclusion

The foregoing briefly outlines some of the essential changes made to an existing systems code to make it appropriate for studies of the HTGR plant. Further improvements can be made as the need arises (e.g. including CACS components for calculations of loss of normal cooling, modifying core kinetics models as needed, etc.). However, the present program has been tested in a number of arbitrarily fabricated simulations and found to operate properly.

9.7 HTGR Code Library (L. G. Epel, J. Skalyo)

A survey of computer programs used by the General Atomic Company to study HTGR safety-related problems was made and a list of 105 codes relevant to this matter was compiled. These codes were classified into 7 subject categories for purposes of summarizing their characteristics here. The 7 broad areas of technology and the number of codes in each of these categories follow:

1. Cross Section Preparation - 13 codes
2. Core Physics - 9 codes
3. Fluid Flow, Inventory, and Heat Transfer - 18 codes
4. Fission Product Behavior - 8 codes
5. Radiation Phenomena - 15 codes
6. Stress Analysis and Seismic Effects - 21 codes
7. Mechanical Component Design - 21 codes

Brief descriptions of all of the 105 codes are given in Appendix II*. Each of the 7 sections in this Appendix (one for each category above) begins with a summary sheet for the codes in that section. The summary sheet outlines the functional area of every individual code in that section, ranks its operational and documentation status, indicates its proprietary nature, describes roughly the number of cards in the source program, indicates whether machine language is used in the source program, lists any other codes needed for input to the code, states whether a test problem exists for the code, and finally, ranks the relative importance of the code

*Appendix II was prepared by SAI (Science Applications, Inc.) as part of a subcontract for the HTGR Safety Evaluation Division.

to HTGR safety analysis. Following the summary sheet, more detailed descriptions of the codes in each section are given. This Appendix serves as a useful guide to the computer programs used by GA in safety studies such as the PSAR's and LTR's.

A number of codes have been selected from among the 105 to be converted to operational status on the BNL CDC 7600 computer system during calendar year 1974. This includes 3 from section 1, 3 from section 2, 5 from section 3, and 1 from section 7. The lower numbered sections relate to more fundamental phenomena and as such have been emphasized in the early stages of developing the code library. For each of the codes converted to the CDC 7600, the HTGR Safety Code Library maintains, as a minimum:

1. The CDC 7600 operational version of the source program,
2. the data for a sample problem run at BNL,
3. the output of the sample problem run at BNL,
4. the original UNIVAC 1108 operational version of the source program,
5. documentation on the code, such as the user's manual.

The 17 codes converted during the first half of FY 1975 constitutes the beginning of the HTGR Safety Code Library. Additional ones will be added to the Library during the second half of this fiscal year and in subsequent years as deemed appropriate. Some of the codes collected to date have already been distributed to other laboratories when requested by them.

The capability of the Safety Group to do an in-depth study of the safety of gas-cooled reactors requires an ability to reproduce the results of a vendor calculation and additionally an ability to make an independent calculation. General Atomic Company (GA) is a commercial source for the HTGR and has long been originating and developing codes necessary for the design and construction of the HTGR. GA is therefore a most convenient source for starting an HTGR code library with its attendant documentation. It is necessary therefore to convert codes from the GA Univac 1108 computer system to the BNL Control Data 7600 computer (the 7600 is more than 10 times faster than the 1108).

Many of the GA codes have been made available to the public through the Argonne Code Center (ACC) and merely need conversion to 7600 operation. However, a major share of the codes necessary for a safety evaluation are of a proprietary nature; this is a natural consequence of a code customization for a particular concept of reactor design. These codes present a problem for conversion to 7600 operation as in many cases documentation or a user's guide is not available at the time the code is converted. The necessity to convert any particular proprietary code has therefore been dictated by its relation to the developing experimental program in HTGR safety or its immediate importance in an analysis of an accident sequence.

A large group of programs can quite naturally be classified as HTGR cross section and nuclear data manipulation codes. Since each manufacturer

of nuclear reactors has its own design method and nuclear data base, it is necessary to be able to reproduce the GA determined cross sections in order to make comparisons with results obtained from non-vendor related cross section preparation codes. Additionally, comparisons can be made with results obtained from a newer data base which may evolve from future measurements. It is noted that the National Neutron Cross Section Center (NNCSC) located at BNL is presently involved in formulating a reference data base which they will check by comparing calculations with thermal reactor benchmarks. For the present instance, the cross section preparation codes will defer such cross section evaluation and will use the ENDF/B⁽⁵⁷⁾ card image tapes of the individual nuclide cross sections as prepared by NNCSC as a starting point.

It is noted that the major broad group averaged cross section code is MICROX. It is the present standard of reference for HTGR calculations, but it is proprietary. The non-proprietary GGC4 code has been obtained from the ACC and is primarily a thermal reactor design code. GGC5 is a major revision of GGC4 that is used for both fast and thermal reactor problems, but it is also proprietary. It is hoped that GGC5 and MICROX will soon be made available to the Safety Group by GA when they are suitably documented.

While all relevant codes are listed in Appendix II, a few comments about the current status of those particular codes which are being converted or have already been converted are given below. Common to all codes is the fact that slight numerical differences will occur in the 1108 and 7600

calculations when running a test problem. The 7600 has a 60 bit word in comparison to the 36 bit word of the 1108 for floating single precision numbers (transparently it is the 48 bit coefficient in the 6600 word versus the 27 bit coefficient in the 1108 word). In some cases the numerical differences are not slight and a more detailed effort has been expended to check that the code is working correctly. A brief description of the particular codes is given in Appendix II. As most codes pertain to cross section calculations, we show in Figure 26 an illustration of how these codes develop broad group averaged cross sections from the ENDF/B card image data tapes.

1. CRECT⁽⁶⁸⁾ - Corrects data on an ENDF/B card image tape.
This program is operative on the 7600.
2. LISTFC⁽⁶⁸⁾ - Gives an interpreted listing of data from an ENDF/B card image tape. This program is operative on the 7600.
3. RIGEL⁽⁶⁸⁾ - Retrieves, merges, changes mode or the arrangement of data on an ENDF/I tape. The main purpose to date is to change the format of the ENDF/B card image tape into binary form suitable for use in the FLANGE code; it is operational on the 7600.
4. ENDFB2⁽⁶⁹⁾ - Converts the ENDF/B card image tape of the nuclide cross section into a binary format for use in GFE2 and GAND2. This code is operative on the 7600 and is very

Schematic Diagram of the Preparation of
Neutron Cross Section Data Libraries at
Gulf General Atomic

Figure 26

similar to RIGEL except that it is specifically written for efficient use of the 1108.

5. FLANGE⁽⁷⁰⁾ - Prepares thermal cross section transfer arrays; it additionally utilizes the File 7 data of the ENDF/B tape dealing with the thermal scattering law. This code is operative on the 6600 and should be soon converted to the 7600 without difficulty. Two sample problems are given in the report. Problem 1 is an example of obtaining group-averaged cross sections with no scattering matrices being generated. It is for U235 (MAT=1102) and has not been checked as yet. Problem 2 generated a scattering kernel for H bound in H₂O (MAT=1002) and has been run successfully. Some small numerical differences are found, but are attributed to word size differences of the 1108 and 7600.
6. GAND2⁽⁶⁹⁾ - Prepares group cross sections from ENDFB2 binary data tapes. The code is operative on the 7600. Again the calculations differ in a few places from the 1108 version of the sample problem listed in GA-7542. The physics and numerics involved in the calculation have been investigated and it is concluded that the code is working as intended. The sample problem is concerned with PU239 (MAT=1159). Large disparities appear in the final table of the GAND2 sample problem where cross sections at discrete velocity points are

computed. Here a 5580 point resonance cross section calculation (GAR data) is being prepared from 0.414 to 3355 eV. In the unresolved resonance region, a statistical distribution of resonance levels is generated with a random number generator. The sequence of random numbers generated in the calculation is dependent on the seed and multiplier of the algorithm and on the computer word size. Disclaimers to this effect are given in the code documentation and it is noted therein that the effect could have a real consequence in fast reactor calculations only.

While large differences in the various cross sections in the unresolved region are obtained at the individual velocity points, group cross section averages over hundreds of points should be quite similar between the 7600 and 1108 calculations done for the thermal reactor.

Figure 27 is an illustration of the effect; the output from a GAR calculation is plotted for the highest calculated energies (≈ 3.355 keV). The channel widths are $\Delta V = 142$ m/s. The 1108 sample problem output is drawn with two calculations on the 7600 which differ only in that the seed number of the random generator algorithm is different. The report⁽⁶⁹⁾ also notes that "the disadvantage is that when, as may be the case in certain Doppler coefficient calculations for fast reactors,

only a small range of energies and hence a small number of the statistically generated resonances contribute most of the calculated effect, the result will have a non-negligible statistical uncertainty due to the random nature of the process used to generate the set of "pseudo" single level parameters."

7. GFE2⁽⁶⁹⁾ - Prepares fast group cross sections from ENDFB2 binary data tapes. The code is operative on the 7600. Here again small differences with the 1108 calculation of the sample problem are found. These differences are noted in the calculation of some of the inelastic downscattering arrays. Double precision is required for these calculations and again, as in GAND2, the differences are numerical in origin and due to computer word size differences. These differences are seen in downscattering groups which are five or so orders of magnitude smaller than the leading terms in the array. Larger terms are in excellent agreement. The 7600 generated cross sections if used in a reactor calculation would therefore give results identical to the results obtained using 1108 generated cross sections.
8. MAKE⁽⁷¹⁾ - Prepares a fine group cross section tape from GFE2 for input to GGC4. The code is operative on the 7600 and a simple problem has been checked.

9. PRINT⁽⁷¹⁾ - Is a utility routine used to read the fast tapes produced by MAKE. The code is operative on the 7600 and has been utilized in reading a fast tape prepared for GGC4 that was obtained from the ACC.
10. WTFG⁽⁷¹⁾ - Prepares a fine group thermal cross section tape from either a GAND or FLANGE calculation for input to GGC4. The code is operative on the 7600 and a simple problem has been checked.
11. SPRINT⁽⁷¹⁾ - Is a utility routine used to read the thermal tape produced by WTFG. The code is operative on the 7600 and a simple problem has been checked.
12. GGC4⁽⁷²⁾ - Prepares broad group averaged cross sections. The code as obtained from ACC is operative on the 7600 and a sample problem using 2 of 3 data tapes, which were also supplied, has been run and gives agreement with the problem output given in GA-7157⁽⁷³⁾.
13. INTERP - Calculates broad group cross sections from MICROX OUTPUT data tapes. The code has been obtained from LASL and is operative on the 7600. A sample problem utilizing the MICROX OUTPUT data tapes prepared by GA has been checked with a sample problem run at LASL and exact reproduction is obtained.

14. OXIDE3⁽⁷⁴⁾ - Is a code analyzing the transient response of the core following accidental inleakage of an oxidant into the HTGR primary coolant stream. This code is a proprietary code and the code received by us is not necessarily the same as that used to run the sample problems given in either GA-LTR-7 or in Amendment 1 to LTR-7 (format differences are evidenced between the latter two documents indicating some program modification. The code has been modified to run on the 7600 and preliminary investigation indicates it is running properly although answers to the problem illustrated in GA-LTR-7 are not reproduced exactly. The imposition of consistency in the lengths of labeled common used in the various subroutines (this is probably not necessary on the 1108) and the modification of some apparent errors were not sufficient to obtain agreement. Disclaimers in the letter of D. S. Duncan to R. A. Clark dated January 6, 1975, pertaining to the detailed accuracy of the values given in GA-LTR-7 has additionally led us to defer a comparison of 7600 and 1108 calculations until later. In the interim, a literature survey of the values of the various input parameters will be initiated as noted in another area of this report.

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SECTION 1

**CROSS SECTION PREPARATION
CCODE ABSTRACTS**

1. CROSS SECTION PREPARATION CODES

Code	Library Number	Functional Area	Proprietary (P)	Operational Status	Documentation Status	Test Problem	Number of Cards	Machine Language	Input Codes	Relative Importance
CRECT		cross section data tape manipulations	-	A	A	yes	300	no	CHECKR, ENDFB2	A
ENDFB2		cross section data tape conversions	-	A	A	yes		no	GAND2, GFE2, CHECKR, CRECT	A
FLANGE		thermal cross section transfer arrays	-	A	A	yes	3300	no	GASKET, RIGEL, HEXSCAT	A
GAND2		prepares cross sections from basic data	-	A	A	yes	2000	no	ENDFB2	A
GANDY3		unresolved resonance cross sections	-	A	A	yes		no		C
GAROL		resolved resonance cross sections	-	A	A	yes	2800	yes		B
GFE2		prepares fast group cross sections from basic data	-	A	A	yes	2000	no	ENDFB2	A
GGC-4		prepares broad group averaged cross sections	-	A	A	yes	1200	no	MAKE, PRINT, SPRINT, MIXER, WTFG, MGT, COMBIN, MST	A
GGC-5		prepares broad group averaged cross sections	P	B	B	yes		no	MAKE, MST, PRINT, MIXER, WTFG, SPRINT, MGT, COMBIN	B
MAKE		modifies fast cross section data tapes	-	A	A	yes	1000	no		A

Operational Status: production-released (A), production-not released (B), underdevelopment (C).

Documentation Status: report released (A), internal reports (B), underdevelopment (C).

Relative Importance for Safety Analysis: primary (A), alternate (B), minor support (C).

1. CROSS SECTION PREPARATION CODES (Cont'd.)

Code	Library Number	Functional Area	Proprietary (P)	Operational Status	Documentation Status	Test Problem	Number of Cards	Machine Language	Input Codes	Relative Importance
MICROX		prepares broad group averaged cross sections	P	B	B	yes	no	MTAPE, GAND2	A	
MTAPE		cross section data tape manipulations	P	B	B	yes	no	WTFG, MAKE	A	
WTFG		thermal cross section data tape manipulation	-	A	A	yes	1650 no	MGT, COMBIN, SPRINT	A	

Operational Status: production-released (A), production-not released (B), underdevelopment (C).

Documentation Status: report released (A), internal reports (B), underdevelopment (C).

Relative Importance for Safety Analysis: primary (A), alternate (B), minor support (C).

CRECT

Copies all or part of an ENDF/B card image tape permitting insertions, deletions and/or changes to be made to the data. Uses one or two input data tapes and uses zero, one or two output data tapes. The running time depends upon the input data tape length. Full (23,000 record) tape takes 9 minutes to read.

ENDFB2

Conversion of ENDF/B cross section data from a card image to a binary tape for use by GAND and/or GFE. Processes files one through seven of the current ENDF/B format. Typical computer running times for one material are less than a minute. Total time determined by position of material on the card image tape. A full card image tape (containing approximately 23,000 records) takes about 9 minutes to read.

FLANGE

Computes energy transfer arrays for use in thermal neutron spectrum codes such as GGC, THERMOS, and MICROX from scattering law data [S(alpha,beta) data] generated by codes such as GASKET. Interpolation and integration of the S(alpha,beta) data. Upscattering is computed by detailed balance in the present production version (FLANGM).

GAND2

Prepares the cross sections needed for detailed computations of neutron energy spectra in reactors from a file of basic nuclear data in the ENDF/B version II or III formats (also retains the ENDF/B version I capability of GAND). Fine group cross sections and scattering transfer array coefficient arrays for use in the GAF code are prepared using modified versions of the routines used to prepare GGC-4 data tapes as well as the GANDY code for unresolved resonance region cross section generation. Pointwise resolved resonance region cross sections for use in the GAR and MICROX codes as well as the GAROL option of the GGC-5 code are prepared using the PSEUDO code to generate statistical resonances and the FSDP3 code to generate the cross sections from both the statistical resonances and the resolved resonance parameters. Cross section values in the thermal neutron energy range may be prepared for non-moderator nuclides (those nuclides for which no detailed scattering law data is given in file 7) using either a strictly pointwise calculation or an ultra-fine mesh averaging procedure using a neutron energy spectrum obtained by interpolation from an input spectrum.

GANDY3

The GANDY code evaluates temperature-dependent effective neutron capture, fission, and scattering cross sections in the unresolved resonance region from average resonance parameters. Effective cross sections are evaluated at specified energies and temperatures using the narrow resonance approximation and assuming isolated resonances. Resonance cross sections are treated on a statistical basis, using chi-squared distributions for the partial widths. The line shape functions, psi and chi, are evaluated in a subroutine written by Naliboff which is based upon a method developed by Adler and Naliboff. The statistical averaging is carried out using a quadrature scheme introduced in the RAPTURE code.

GAROL

Computes effective group cross sections for the resolved resonances of a mixture of isotopes in a two-region cell using collision probability methods. Basic cross sections include temperature dependence. The program allows a choice of geometries and can accept an arbitrary table of escape probabilities. A Dancoff correction may be used to account for shadowing effects in a tight lattice, and cross sections may be $1/v$, constant, computed from Breit-Wigner resonance parameters, or given in tabular form. The mesh may be chosen at equal energy or lethargy intervals, proportional to the neutron velocity, or as an arbitrary table of values.

GFE2

Prepares $1/E$ or fission spectrum weighted, zero temperature, infinite dilution group averaged cross sections and scattering transfer arrays for use in the fast sections of the GGC-4 and the GGC-5 codes from a file of basic nuclear data in the ENDF/B version II or III formats (the ENDF/B Version I Format capability of the original GFE code is also retained).

GFE2 is an extended special purpose version of the GAF cross section preparation section of the GAND2 code. The extensions include the computation of P3 instead of P1 scattering transfer arrays and a high speed, zero temperature, infinite dilution resolved resonance calculation via the INDER code.

GGC-4

Produces multigroup cross section sets using basic nuclear data. The code is divided into fast (GAM), thermal (GATHER), and combining sections. Basic nuclear data consisting of fine group averaged cross sections, nuclide densities for mixtures, scattering transfer arrays and resonance parameters, etc., are used as input to GGC-4. Spectrum calculations are performed and used to obtain weighted broad group cross sections for use in transport and diffusion theory codes.

GGC-5

Solves the multigroup spectrum equations with spatial dependence represented by a single input buckling (positive, negative, or zero). Broad group cross sections (shielded or unshielded) are prepared for diffusion and transport codes by averaging with the calculated spectra over input-designated energy limits. The code is divided into three main parts. A fast (GAM) section which covers the energy range from 14.9 MeV to 0.414 eV, a thermal (GATHER) section which covers the energy range from 0 to 2.38 eV, and a combining (COMBO) section which combines fast and thermal cross sections into single sets. Basic nuclear data for data tapes. More refined resolved and unresolved resonance treatments are available in GGC-5 relative to GGC-4.

MAKE

The basic program for creating and modifying cross section data tapes for use in the fast section of GGC-4 and GGC-5. Creation of tapes may be done from cards or from an existing tape. Tapes may be updated by adding new data (either from cards or tape) and/or deleting existing data.

MICROX

MICROX is an integral transport theory flux spectrum code which solves the neutron slowing down and thermalization equations on a detailed energy grid for a two-region lattice cell. The fluxes in the two regions (fuel and moderator) are coupled by collision probabilities; the computation of these collision probabilities is based on the flat flux approximation. A second level of heterogeneity can be treated, i. e., the fuel region may have grain structure. A modification to the original version of MICROX allows the code to model two types of grains simultaneously in both the fast and thermal energy ranges. The neutron leakage effects are simultaneously calculated by applying a simplified B1 method. Basic nuclear data used by MICROX consists of GAM data in the fast energy range, and GATHER data in the thermal energy range. In the epithermal resolved energy range, the MICROX spectrum calculation is performed on an ultra-fine energy grid using GAR data (temperature-dependent doppler-broadened resonance cross sections below 3 - 8 keV) as a complement to the GAM data for the most important nuclides.

The calculated detailed flux and current spectra are used for condensing the basic nuclear cross section data into cell-averaged broad-group cross sections and transfer arrays (up to P1).

MTAPE

MTAPE generates, copies, updates, and edits the GG data tapes used by the MICROX and MICROBURN codes. The GG tape consists of two files (sections) which contain infinite dilution cross sections for the fast and thermal energy ranges, respectively.

WTFG

Basic program for creating and modifying cross section data tapes for use in the thermal section of GGC.

SECTION 2

CORE PHYSICS CODES

2. CORE PHYSICS CODES

Code	Library Number	Functional Area	Proprietary (P)	Operational Status	Documentation Status	Test Problem	Number of Cards	Machine Language	Input Codes	Relative Importance
BLOOST-7		point kinetics, temperature distribution	-	A	A	yes	no		A	
BUGTRI		2-dimensional diffusion theory and depletion	-	A	A	yes 2250	yes		B	
FEVER		1-dimensional depletion and diffusion theory	-	A	A	yes	no		A	
GAMBLE5		2-dimensional diffusion theory	-	A	A	no	yes		A	
GAUGE		2-dimensional diffusion theory and depletion	-	A	A	yes	no		B	
GAZE-2		1-dimensional diffusion theory	-	A	A	yes 6100	no		B	
PDQ7		1-, 2-, and 3-dimensional diffusion theory		A	A	yes 5200	no		B	
TEMCO		reactor temperature coefficients	-	A	A	yes 3100	no		A	
TWOTRAN		2-dimensional transport theory	-	A	A	yes 8200	no		B	

Operational Status: production-released (A), production-not released (B), underdevelopment (C).

Documentation Status: report released (A), internal reports (B), underdevelopment (C).

Relative Importance for Safety Analysis: primary (A), alternate (B), minor support (C).

BLOOST 7

Combines a space-independent reactor kinetics code with a version of the TAC-2D transient heat transfer code. Feedback models are provided for both HTGR and GCFR problems based on temperature changes in a single average fuel element. The capability to study the temperature behavior in non-average elements is also provided. Two-dimensional heat-transfer routines are limited to r-z geometry, 36 regions, 15 materials, 15 radial mesh points, 40 axial mesh points, 6 radial region boundaries, 10 axial region boundaries, and inner and outer axial coolants. Kinetics equations allow 6 delayed groups. The material, gas, and coolant properties and heat generation rates are supplied as FORTRAN arithmetic statement functions. This subroutine is therefore unique to a particular fuel element design and may be considered as part of the input data. For cases in which the initial temperature distribution is unknown it is possible to run the BLOOST code as a pure heat-transfer code omitting reactivity input to the kinetics portion and obtain a temperature distribution appropriate to initial power conditions punched on cards suitable for later use in kinetics problems.

BUGTRI

The two-dimensional multigroup neutron diffusion theory equations for triangular geometry are solved to obtain the multiplication factor and the spatial flux and power distributions. Complete reactor life histories with partial refueling at a number of reload time points can be calculated. The depletion scheme of all burnable nuclides is specified by the user at execution time. A regionwise depletion scheme is used. Concentration dependent self-shielding factors may be applied to any nuclide.

FEVER

Performs a multigroup one-dimensional depletion calculation which allows fuel in various stages of irradiation to be homogenized into the same region for purposes of the diffusion calculation but follows the depletion of each of the subregions separately. The calculation may be interrupted periodically for refueling one or more subregions. Recycling is optional and there is no limit to the number of refuelings which may be performed. A control poison search is available and concentration dependent self-shielding factors may be applied to any nuclide. The nuclide scheme employed is specified by the user.

Flux distribution and multiplication factors are computed by means of a multigroup diffusion routine. The depletion calculations are performed assuming that the flux amplitude, spectrum and spatial distribution remain constant for short time-steps. The flux in the thermal groups is renormalized to the power prior to each short time-step and the spectrum is recalculated after a specified series of short time-steps. For burnup, nuclides belonging to each subregion react to the average fluxes present in the region to which the subregion is assigned. The diffusion calculation is performed using group constants which are volume-weighted properties of one or more subregions homogenized into a diffusion region. The multigroup one-dimensional diffusion routine DCALC, incorporated in FEVER/M1, is based on a direct marchout technique with periodic reconditioning. The conventional fission source iteration has been replaced by Wielandt iterations, solving all group fluxes simultaneously.

GAMBLE 5

The homogeneous two-dimensional multigroup diffusion theory equations with arbitrary group-to-group scattering and arbitrary fission transfer are solved for heterogeneous assemblies in x-y and r-z geometry. Homogeneous logarithmic boundary conditions are used at the outer surface of the assembly and at the surface of nondiffusion regions. The results include the group- and point-dependent neutron fluxes, the power distribution, the neutron multiplication factor (k effective), and a detailed neutron balance.

GAUGE

The two-dimensional few group neutron diffusion theory equations for a uniform triangular mesh are solved to obtain the multiplication factor and the spatial flux and power distribution of reactors with hexagonal core configuration. Complete reactor life histories with partial refueling at a number of reload time points can be calculated. At each discrete time point a control rod search may be performed to maintain criticality at all times. The depletion scheme of all burnable nuclides is specified by the user at execution time. Three modes of operation are possible: 1) straight burnup calculation, 2) control rod criticality search, allowing the adjustment of a number of control rod banks according to a prescribed rod sequencing scheme, 3) a series of static calculations with insertion of rods into fixed prescribed positions.

GAZE-2

One-space dimensional, multigroup neutron diffusion theory code including criticality searches on position of any radius or on volume fractions of any combination of nuclides in any combination of materials.

PDQ7

PDQ7 solves few-group neutron diffusion-depletion problems in one-, two-, and three-dimensions. Adjoint solutions are also available and two overlapping thermal groups may be used in one- and two-dimensional problems. Either pointwise or regionwise depletion may be performed using the HARMONY depletion system. The geometry may be rectangular, cylindrical, or spherical in one dimension, rectangular, cylindrical, or hexagonal in two dimensions, and rectangular or hexagonal in three dimensions. All geometries provide for variable mesh spacing in all dimensions. Zero flux, zero current, and rotational symmetry boundary conditions are available, and boundary value problems may be solved by specifying the flux values on one or more boundaries. The Bettis revised PDQ7 may be used to also solve additive fast-source and simplified PL problems as well as the three-dimensional synthesis eigenvalue problem. Control searches, thermal feedback, and xenon feedback are optional.

TEMCO

Computes reactor temperature coefficients. Given cross sections for each temperature to be considered, the code performs a series of one-dimensional diffusion calculations and then uses the results to generate the temperature coefficient. The one-dimensional diffusion subroutine DCALC is based on a direct marchout technique with periodic reconditioning, solving for all thermal group fluxes simultaneously.

TWOTRAN

TWOTRAN solves two-dimensional particle transport problems. Separate versions are available in x-y and r-z geometries as well as a general geometry version in x-y, r-z, and r-theta geometries. Both direct and adjoint, homogeneous (keff or parametric eigenvalue searches) or inhomogeneous time-independent problems are solved subject to vacuum, reflective, or input specification of boundary flux conditions. Both anisotropic inhomogeneous problems and general anisotropic scattering problems are treated.

SECTION 3

FLUID FLOW, INVENTORY, AND TEMPERATURE CODES

3. FLUID FLOW, INVENTORY, AND TEMPERATURE CODES⁽¹⁾

Code	Library Number		Functional Area	Proprietary (P)	Operational Status	Documentation Status	Test Problem	Number of Cards	Machine Language	Input Codes	Relative Importance
CINDA		network temperature distributions	-	A	A	yes	6000	no	CINGEN	A	
CINGEN		digitizer-geometry parameters	P	B	B	no	<1000	no		A	
CONTEMPT-G		containment analysis	P	A	A	yes	3500	yes		A	
CORCON		temperature transients due to LOFC	-	A	A	yes	1000	no		A	
DMOR		thermal transient	P	C	C	-	2000	no		A	
FLASH-4		fluid transient analysis	-	A	A	yes	11600	no		A	
HEXT		1-dimensional thermal analysis	P	B	B	yes	<1000	no		B	
INGEN		digitizer-geometry parameters							not used at present	C	
ME89		steady-state temperature distributions	-	A	A	yes					
ME112-LION		3-dimensional temperature distribution	-	A	A				At BNL		
OXIDE 3		steam or air/graphite reaction	P	B	A	yes	5000	yes		A	
POKE		core flow and thermal analysis	P	B	B	-	2000	no	GAUGE, FEVER SCANAL, GATT	A	
RATSAM		DBDA core pressure differential	P	C	C	no	2500	yes	TAP		
RECA		thermal transient analysis	P	C	C	-	2000	no		A	
TAC-2D		2-dimensional temperature distributions	-	A	A	yes	4000	no		A	

Operational Status: production-released (A), production-not released (B), underdevelopment (C).

Documentation Status: report released (A), internal reports (B), underdevelopment (C).

Relative Importance for Safety Analysis: primary (A), alternate (B), minor support (C).

3. FLUID FLOW, INVENTORY, AND TEMPERATURE CODES (Cont'd.)

Code	Library Number		Functional Area	Proprietary (P)	Operational Status	Documentation Status	Test Problem	Number of Cards	Machine Language	Input Codes	Relative Importance
TAC-3D		3-dimensional temperature distributions	-	A	A	yes	8000	no		B	
TAP		thermal transient analysis	P	C	B	-	4000	no		A	
THTD		network temperature distribution							not used at present	C	

Operational Status: production-released (A), production-not released (B), underdevelopment (C).

Documentation Status: report released (A), internal reports (B), underdevelopment (C).

Relative Importance for Safety Analysis: primary (A), alternate (B), minor support (C).

CINDA

A program for calculating transient and steady-state temperatures for a network of nodal points having heat capacities and connected by thermal resistances. This gives the program considerable flexibility in handling complex geometries and boundary conditions. Conduction, convection and radiation thermal resistances and internal heat generation can be accommodated. The flexibility of the program is further increased by providing several optional finite difference solution schemes.

CINGEN

Accepts X-Y output from LARR-M digitizer. This data is used to calculate geometry parameters for input to CINDA.

CONTEMPT-G

An extensively modified version of the Phillips CONTEMPT code used for HTGR containment analysis. The code describes the energy balance for the containment atmosphere. It calculates the pressure and temperature of the atmosphere during transients caused by primary or secondary coolant leaks. The calculation takes into account the heat losses to the inner structures and outside walls.

CORCON

The thermodynamic aspects of the maximum hypothetical fission product release accident are evaluated with the CORCON program. This program was used in evaluating the permanent loss of forced circulation (LOFC) in the Fort St. Vrain HTGR. CORCON is used to calculate temperatures throughout the core and its PCRV cavity during the accident. These temperatures are then used as input to the SORS programs in order to calculate the fission product release from the core.

The CORCON program models the active core with spatially distributed after-heat sources; the top, bottom, and side graphite reflectors; the graphite core support blocks, and the water-cooled, steel-lined, concrete PCRV. Heat transfer within this model occurs principally by conduction through the core and reflectors to the peripheral surfaces of the core. The heat is then transferred by thermal radiation and conduction across the space occupied by the gas to the cover plates of the PCRV liner thermal insulation.

By simulating fission product loss and redistribution, the CORCON program redistributes the afterheat during the transient. The calculation of fission product transport in CORCON is performed in a manner which also leads to conservatively high core temperatures.

DMOR

A subroutine in the TAP program which dynamically models the steam generator systems. Models are used to obtain the temperature distribution as a function of time for heat transfer from the helium coolant to the secondary coolant system. All parts of the steam generator system are included in the DMOR models.

FLASH-4

The FLASH-4 program solves the conservation equations of energy, mass, and momentum for a nodal representation of a water reactor plant or hydraulic network during transients. The method of solution is specifically directed at those transients in which system pressure varies in both space and time. A physical characteristic of such a compressible flow solution is that pressure disturbances propagate at local sonic velocity. Since the conservation equations implicitly contain this short time constant process, the numerical integration techniques must also reflect this phenomenon. Accordingly, explicit or forward difference integration methods must use time steps less than this acoustic propagation time to be stable. The FLASH and FLASH-2 programs are restricted by this limitation, though for rapid depressurization of a reactor plant, many of the other processes also have small time constants and this acoustic limit is not overly restrictive. However, for the simulation of slowly varying transients, e. g., a small leak, the small time steps are still demanded by the conservation equations, making the overall computer time prohibitive.

A method of alleviating this time step limitation is to use a fully implicit integration technique for the conservation equations. In FLASH-4, a set of coupled differential equations representing the conservation equations applied to a user supplied network of nodes and flow paths is presented. The system of equations is linearized, and integrated by a fully implicit difference method.

HEXT

This is a special purpose one-dimensional heat conduction code for the HTGR fuel block geometry. Predicts steady state temperature distribution in HTGR graphite-fuel distribution. Analytic solution to problem.

INGEN

Obsolete. Replaced by CINGEN.

ME-89 - Steady-State Thermal Analysis

Steady-State Thermal Analysis is a finite difference computer code for calculating the steady-state temperature distribution in systems of any configuration. The system to be analyzed may be one, two, or three dimensional. To apply the finite-difference method, the system is divided into elements. If the elements are sufficiently small, the temperature throughout each is approximately uniform. Since, at steady-state, the temperature of any point of the system is time independent, there is no net heat flowing into any element. Thus, to obtain the steady-state temperature distribution of a system, a heat-transfer equation for each element connected with all its neighboring nodes must be balanced. Such a heat balance equation is solved by successive iterations until the temperature change per iteration at every node meets a specified condition. It is then assumed that the temperature distribution in the system has reached the steady state.

ME112 - LION

LION - Temperature Distributions for Arbitrary Shapes and Complicated Boundary Conditions - is a digital computer program which is used to solve three-dimensional transient and steady-state temperature distribution problems. The program may also consider sub-cooled nucleate boiling and coolant heat transfer effects. The surface conditions may be forced convection, free convection or radiation and heat may be externally or internally generated. Input to the program consists of structural geometry, physical properties, boundary conditions, internal heat generation rates, coolant flow properties and flow rates.

The program solves the transient heat conduction equations for a three-dimensional field using a first-forward difference method. To ensure the temperature calculation stability, LION can determine the suitable time increment, if the input time increment is specified too large.

OXIDE

The computer program OXIDE was developed to analyze the transient effects of accidental inleakages of steam and/or air to the primary coolant system. The code analyzes the three-dimensional effects of steam-graphite and steam-fuel reactions in the core and simulates the primary system and the secondary containment with respect to heat and mass transfer. The code divides the core into a number of regions (20 maximum) and each region is subdivided into segments (10 maximum), which allows for a maximum of 200 segments to be analyzed. In addition, each segment can be divided into six concentric rings or slabs of graphite around a characteristic coolant channel. This allows simulation of gaseous diffusion and in-core reaction in the graphite away from the coolant channel interface.

Input to the code consists of total primary circuit flow rates and core inlet temperature as functions of time, regional and axial power distribution factors, initial temperatures, and the flow orifice coefficients for each region. The code calculates the spatial transient flow and temperatures as conditions necessary for oxidation calculations. The nuclear heat generation, graphite temperature, coolant temperature, total pressure, steam concentration, reaction product concentrations, steam-graphite and steam-fuel reaction rates, heats of reaction, and graphite burnoff are calculated for each segment during each time step. PPS actions can be simulated in the code.

POKE

A multichannel, one-dimensional flow and thermal analysis code. It provides a means for broadly surveying a large number of segments of the core simultaneously. (TAC2D and RAT3D are used for most detailed calculations of individual regions.) POKE is a steady-state program with which details of primary coolant mass flow, coolant temperature, and fuel distribution can be determined. FLAC (section 7) is used to solve problems beyond the scope of POKE.

RATSAM

The rapid depressurization of the PCRV associated with the DBDA can lead to abnormal pressure differentials and flows across internals of the PCRV. These forces result from pressure gradients established in the primary coolant circuit as the helium coolant exhausts to the containment. The magnitudes of these forces are calculated using an analytical model which divides the primary coolant circuit into distinct volumes or nodes. The thermodynamic conservation laws of mass and energy are applied to each of these nodes. In addition, the one-dimensional momentum equation is applied to the flow paths between the nodes to account for fluid inertia effects. Wall friction is included in the momentum equation and is assumed to be proportional to the square of the mass flow rate and inversely proportional to the fluid density.

The system of differential equations which results is integrated in time, using numerical approximations, to yield the time-dependent thermodynamics properties of interest for each node in response to a rupture in one or more of the nodes. The model also simulates the dynamic behavior of the helium circulators and the heat transfer between the coolant, core, and steam generators.

RECA

The Transient Analysis Program (TAP) gives the time-response performance of the Nuclear Steam Supply (NSS) system and Balance of Plant (BOP) as a whole by accounting for the interactions of the major NSS system components, BOP equipment, and the overall control system. For detailed reactor core cooling evaluations, a two-step approach is used where information on the overall system performance, generated by TAP, is inputted into a precise analytical model of the reactor core. This second analytical model is incorporated into the Reactor Emergency Cooling Analysis (RECA) program, which outputs the time evolution of the important state variables (temperature, pressure, flow, etc.) at various points in the primary system. The program includes a model of the performance characteristics of the CACS and, therefore, is used for the evaluation of reactor core transient behavior during CACS operation.

The analytical model employed by the RECA code is based upon the straightforward application of well established physical laws governing transient heat conduction and steady-state fluid flow. The coolant flow distribution through the core is modeled by an array of coolant channels, one channel for each refueling region, including the partial regions, which have fewer than seven columns, and one for the side reflector.

The governing equations (coolant EOS; conservation of mass flow, momentum, energy; heat conduction) are solved in finite difference form. In obtaining the numerical solution to the governing equations, the fluid dynamics equations are spatially uncoupled from the heat conduction equation. The calculations for each time point proceed by first evaluating all properties such as the heat generation rates, the capacitance of each node, and the thermal resistances between nodes. Then, the flow distribution in the core is calculated by solving the set of fluid dynamics equations simultaneously for the array of coolant channels comprising the core. The boundary condition which must be satisfied is that the plenum-to-plenum pressure drop in each channel be equal.

Following the evaluation of channel flow rates, the node temperatures are calculated, where the nodes are classified as solid nodes, coolant nodes, or boundary nodes. The solid nodes represent volumes of solid mass whose temperatures must be determined, the coolant nodes represent flowing coolants which have a negligible heat capacity but whose temperatures must also be determined, and the boundary nodes represent nodes whose temperatures are known at the beginning of each time step. The node temperature calculations proceed by first evaluating the temperature of the solid utilizing an implicit finite difference formulation. During this calculation step, the coolant nodes are held at constant temperature. The next step is to calculate the temperatures for the coolant nodes, which is done by the method of point relaxation. Following the evaluation of all node temperatures, the calculation sequence is repeated for the next time point.

TAC2D

A program for calculating transient and steady-state two-dimensional temperature distributions in components and structures using the finite difference method. Conduction, convection, radiation and heat generation are accommodated, with variable properties and internal and external flow functions. The use of an orthogonal grid system to define the modal elements and Fortran arithmetic statement functions to specify the material properties, heat transfer coefficients, and fluid flow conditions greatly simplifies the input for TAC2D.

TAC3D

A program for calculating transient and steady-state three-dimensional temperature distributions in components and structures using the finite difference method. Conduction, convection, radiation, and heat generation are accommodated. The elements in the model are defined by a series of orthogonal grid planes, which enables internal evaluation of nodal connections and geometric quantities.

TAP

The analysis of plant performance under anticipated transient and accident conditions mathematically represents the nuclear generating station as a system. The analysis includes mathematical models of all relevant plant components and simulates the interactions of equipment and control systems. The analytical tool used to study the overall plant transient behavior of HTGR's is the Transient Analysis Program (TAP), which incorporates models of all major components of the nuclear steam system, the major control system loops, basic plant protection system actions, and that portion of the BOP and auxiliary equipment having a significant effect on NSS system transients, as well as options for simulating operator actions.

The TAP code computes and outputs transient time responses of reactor power and other pertinent system variables such as primary and secondary coolant flow, temperature and pressure at numerous points throughout the nuclear steam system for use in design and safeguards engineering, equipment selection, development of plant operating strategy, and validation of process control and plant protection schemes.

The TAP code has been used for the evaluation of HTGR performance during normal or anticipated transients and certain accident sequences requiring analysis of the overall plant response. In general, the program employs algebraic and finite difference equations to simulate component response characteristics and system interactions. Implicit integration methods are used whenever necessary to avoid excessive computation time associated with small time steps required by explicit techniques.

THTP

Obsolete. Replaced by CINDA.

SECTION 4

FISSION PRODUCT CODES

4. FISSION PRODUCT CODES

Code	Library Number	Functional Area	Proprietary (P)	Operational Status	Documentation Status	Test Problem	Number of Cards	Machine Language	Input Codes	Relative Importance
FIPER-S	fission product release, 1-dimensional	-	A	A	yes	600	no		A	
FIPER Q	advanced fission product release, 1-dimensional	-	A	A	yes	-	no		B	
FIPER II	fission product release, 2-dimensional	-	A	A	yes	-	-		C	
PAD	plateout in primary loop	P	B	C	-	-	no		B	
RAD	fission product activities in primary coolant	P	B	C	-	-	no		A	
SORS*	volatile fission product release from fuel and fuel element graphite	-	A	A	yes	500	no	CORCON	A	
SORS-G	non-volatile fission product release from fuel and fuel element graphite	-	A	A	yes	250	no	RECA	A	
TRAFIC	fission product release	-	C	C	-	2000	-	SURVEY (P), TREVOR (P)	-	(1)
TRISO	evaluates stresses in fuel particle coatings	P	C	B	-	1000	-			

(1) FIPER series replacement; available late summer 1974.

* Some later proprietary versions may exist.

Operational Status: production-released (A), production-not released (B), underdevelopment (C).

Documentation Status: report released (A), internal reports (B), underdevelopment (C).

Relative Importance for Safety Analysis: primary (A), alternate (B), minor support (C).

FIPER-X, FIPER-S, FIPER-Q, FIPER II

The FIPER codes have been written to describe the actual escape of fission products from the coated fuel particles and their transport to the coolant stream. Tests to determine the release of metallic fission products from fuel material, the diffusion of metallic fission products in graphite, and the vaporization of metallic fission products have been made. The results of these tests are used in the FIPER codes to predict the fraction of metallic fission products released into the coolant gas. There are three versions of the one-dimensional FIPER code. The FIPER-X code was written to numerically solve general linear and nonlinear time-dependent diffusion equations with a number of diffusing substances. The FIPER-Q code is basically the FIPER-X code with a specialized input and output format that is used exclusively for fission product transport analyses. The FIPER-S code was developed to determine total reactor core fission product metal release utilizing simplified models. This code is based on conservative simplifying approximations of the transport process. Preliminary comparisons of FIPER-S indicate that results are within approximately a factor of 2 greater than FIPER-Q results for most reactor operating conditions. No further development of FIPER-S is presently being considered. The FIPER II code is a two-dimensional code that makes possible the numerical evaluation of time-dependent concentration profiles and release rates for highly anisotropic geometries, and couples regions with diffusion coefficients that differ by orders of magnitude. The FIPER II code will be used as a research tool for analyzing experimental diffusion data in order to improve the understanding of the transport process.

FIPER-S is the FIPER series code which is used the most because of its speed. FIPER-Q is also used, but is much slower than FIPER-S. FIPER II is very seldom used. The TRAFIC code which will supersede FIPER-S will include many of the new improvements made in FIPER-S along with additional features and will still be a relatively fast code.

PAD

PAD (plateout activity distribution) solves the coupled, nonlinear differential equations that describe the mass transport between an average coolant concentration and a surface concentration separated by a thin boundary layer. The concentrations of the fission products may be coupled through the decay process or by a concentration dependent sorption process. The code is used to calculate the plateout distribution of fission products in the primary coolant circuit.

RAD

Computes steady-state gas-borne and plateout activities of all the significant fission products in the primary coolant system. RAD computes the total fission product inventory and requires FIPER calculations to provide the sources from graphite into the coolant.

SORS, SORS-G

The time-dependent core temperature data generated by CORCON is used as input to the SORS programs which calculate the time-dependent fission product release from the fuel particles and fuel element graphite.

TRAFIC

TRAFIC is still under development and is intended to be an improved version of FIPER-S using completely different transport equations. Thermal diffusion, radioactive decay, and mixed chemical species interactions on the coolant hole boundary are being considered. Conservation of mass and correct treatment of the time-dependent boundary conditions are additional improvements over the FIPER-S treatment.

TRISO

Coated fuel particle design code which predicts stress levels in SiC layers due to buildup of internal fission product gases. Irradiation effects and interaction of the particle layers with neighboring layers are accounted for. A closed form solution of the thermal stresses attained in creeping stress problems is found.

SECTION 5

RADIATION CODES

5. RADIATION CODES

Code	Library Number	Functional Area	Proprietary (P)	Operational Status	Documentation Status	Test Problem	Number of Cards	Machine Language	Input Codes	Relative Importance
CONTAIN- MENT CLOUD										
CONTAIN- MENT SHIELD		radiation doses							A	
FMC		radiation doses in complex geometries	A	A			no	LUX, GAD, STEM, GAMGAD, NEAT		
GAMLEG		gamma ray cross sections	-	A	A	yes	no		A	
GAMTRAN		radiation doses from station operation								
H DOSE		exposure of personnel to radioactivity		C						
IONEXCHAN- GER		radiation source terms								
MORSE		radiation doses in complex geometries	-	A	A	yes	no		A	
MUSCAT		shield design	P	B	B	yes	3600 no		C	
PATH		shield design	P	B	B	yes	4600 no		A	
RADIOISO- TOPE		radiation source terms								
SHIELD CLOUD		radiation doses							A	
1 DFX version of DTFX		1-dimensional trans- port theory	P	B	A		no		A	
QAD-F		radiation doses			A					
CTD		radiation transport	P	B	B	no			A	

Operational Status: production-released (A), production-not released (B), underdevelopment (C).

Documentation Status: report released (A), internal reports (B), underdevelopment (C).

Relative Importance for Safety Analysis: primary (A), alternate (B), minor support (C).

CONTAINMENT CLOUD

CONTAINMENT SHIELD

The direct gamma radiation contribution from the containment building to an external receptor is calculated by program CONTAINMENT SHIELD. This program models the containment building in the following manner:

1. The reactor vessel and its internals is represented as an "object shield" for this calculation.
2. The circular concrete crane wall inside the containment is modeled in close resemblance to the actual wall when applicable.
3. The cylindrical containment wall is divided into 10 vertical belts of different thickness to model effects of ground, external walls, or internal structure.
4. The top section is hemispherical and of constant thickness.
5. The steel liner is included in both the cylindrical and hemispherical regions, by an equivalent amount of concrete.

The source strength spectral information is input to the program (10 energy groups). The program assumes uniform dispersion of the source in the containment space.

A point-kernel solution is used, having a series of equivalent point sources distributed throughout the containment. Attenuation in the containment atmosphere and in the air outside is considered. Buildup is also considered, using the Berger form of the buildup factor, best fit to 20 mean free paths, as provided in DASA-1892-3, "Weapons Radiation Shielding Handbook".

For the specific case of control room doses due to direct radiation, the control room wall thickness has been added to the containment wall thickness, and all radial partitions and equipment in the containment have been neglected.

CTD

Another one-dimensional transport theory version of the DTFX or 1DFX code.

FMC

The flexible Monte Carlo programs FMC-N and FMC-G were originally written in assembly language for the IBM 704. These programs have been converted to FORTRAN for the CDC1604B by Pratt and Whitney. The programs apply Monte Carlo methods to simulate neutron and gamma ray life histories, respectively, in a source-shield configuration. As the names imply, the codes are designed for flexibility in the geometrical, material, nuclear, and source descriptions of source-shield configurations and variance reduction techniques.

GAMLEG

GAMLEG generates legendre components of the Klein-Nishina photon scattering cross section and calculates absorption, fluorescence, and group-to-group transfer cross sections. The cross sections generated by GAMLEG are used in transport codes such as DTF to calculate one-dimensional photon transport through arbitrary materials.

GAMTRAN

GAMTRAN is a point kernel shielding program developed by Stone and Webster and has been used extensively in the analysis of shield thickness for past shield design studies. The GAMTRAN program had its origin in the naval reactors program and earlier versions of it have been used successfully for shield design on many shipboard applications. It is similar to a number of point kernel programs but embodies a number of features which aid the analyst in the modeling of such components as steam generators and heat exchangers.

IONEXCHANGER

The IONEXCHANGER program generates source terms for the GAMTRAN and QAD-F shield analysis programs. Building of isotopic activity in the resin bed of a demineralizer assuming a constant feed flow rate and constant feed activity are computed by the IONEXCHANGER program. An option is also provided in the program allowing the user to describe a tank system which represents a typical gas holdup tank scheme. In the tank model, the outflow, or bleed flow, need not equal the inflow. Program IONEXCHANGER also has a library with approximately 200 different isotopes. This program also calculates the energy released in seven energy groups. The IONEXCHANGER program was written at Stone and Webster and has been in use since 1967.

MORSE

MORSE is a multipurpose neutron and gamma ray transport Monte Carlo code. Some of its features include the ability to treat the transport of either neutrons or gamma rays of a coupled neutron and secondary gamma ray problem, the incorporation of multigroup cross sections, and option of solving either the forward or adjoint problem, modular input-output, cross section, analysis and geometry modules, debugging routines, time dependence for both shielding and criticality problems, albedo option at any material boundary, one-, two-, and three-dimensional geometry packages, and several types of optional importance sampling. It is especially useful in the computation of radiation doses in complex reactor configurations.

MUSCAT

The multiple-scattering code, MUSCAT, is employed to determine the radiation streaming from the plenum above the reactor through the cold coolant ducts into the steam generator cavity where an assessment of neutron scattering and activation is of importance. Special codes such as FMC and MUSCAT are utilized for the radiation analyses in regions of complicated geometrical configuration such as below the reactor core.

PATH

The PATH code is a flexible point-kernel buildup/attenuation program, and is used for gamma calculations. If heating results are desired, energy-absorption cross sections and buildup factors are substituted. Shielding calculations of the direct gamma dose rates from the spent fuel in the fuel handling machine can be done with the PATH code. Dose rates due to scattered gammas may be computed by a combined use of the MUSCAT and PATH codes. Shielding design of the fuel transfer casks and auxiliary service casks are also performed with PATH.

RADIOISOTOPE

The RADIOISOTOPE program generates source terms for the GAMTRAN and QAD-F shield analysis programs. If the initial source activities in a closed volume are known at some time, t , program RADIOISOTOPE will calculate the residual activity at the end of one or several decay periods. The program can also determine the effects of a continuous recirculation filter system. A library with information on approximately 200 different isotopes is contained within the program. The program calculates the energy released in seven energy groups of gamma radiation accompanying decay for the appropriate isotopes. These data are used as input into the shield analysis program. Program RADIOISOTOPE was written at Stone and Webster and has been in use since 1966. Additional isotope information is added to the program library as necessary.

SHIELD CLOUD

The control room dose from the leakage plume is calculated using the SHIELD CLOUD computer code. This computer code calculates the dose rate transmitted through shielding by gamma radiation emanating from a cloud of finite or infinite dimensions. The cloud is assumed to be hemispherical and composed of radioisotopes which have leaked from a reactor containment after a hypothetical DBDA. The gamma rays of the isotopes contained in the cloud are grouped according to energy, and the dose rate due to each energy group is calculated, the total dose rate being the sum of these values. The method of calculation is to consider an element of volume of the cloud as an isotropic point source. A dose rate equation for this elemental source and dose point is developed using point kernel techniques and buildup factors and is then integrated over the solid angle of the cloud using Simpson's Rule to give the dose rate from the entire cloud at the point of interest. This point is separated from the cloud by a flat shield whose radius is at least as large as the cloud radius. The dose point is on the cloud centerline and on the shield surface.

Input data is the number of energies present, and their values in MeV; the source term in MeV/cc-s; the cloud radius; and the shield material and its thickness.

The computer prints out all input data, as well as absorption and attenuation coefficients, buildup factors, slant penetration factors, the time after the accident, the dose rate due to each gamma energy, and the total dose rate.

IDFX

The linear time-independent Boltzmann equation for particle transport is solved for the energy, space and angular dependence of the particle distribution in 1-D slabs, cylinders, and spheres. Independent source or eigenvalue (multiplication, time absorption, element concentration, zone thickness, or system dimension) problems are solved subject to vacuum, reflective, periodic, or white boundary conditions. A complete energy transfer scattering matrix is allowed for each legendre component of scattering. Solutions to the adjoint transport equation are also obtained.

QAD-F

QAD-F is a point kernel radiation shielding program which is available from the Radiation Shielding Information Center at the Oak Ridge National Laboratory (ORNL). Results from QAD-F have been compared favorably with GAMTRAN results. The program employs generalized geometry routines which allows the construction of a three-dimensional mathematical model of any complex source-geometry configuration. Shield analysis calculations are typically done on a basis of seven gamma radiation energy groups. Sources of radiation are based on computer calculations of the radioactivities balance throughout the plant. The various possible modes of plant operation with various attendant concentrations in different flow paths are considered in determining the maximum source strengths for each shielded area. As a result, the array of sources actually used for the shield design is intentionally unrealistic in that the maxima cannot simultaneously occur in the various parts of the plant. The sources used for shielding calculations represent the worst-case assumptions for each source, not a real balance of radioactivity for normal operating conditions. Programs QAD-F and GAMTRAN are used to compute the dose rates at the shield surface directly opposite the major sources to be shielded. These represent maximum values as the dose rates fall off both horizontally and vertically along a shield wall due to the tremendous slant path shielding which can be involved.

SECTION 6

**STRESS ANALYSIS AND SEISMIC
CODE ABSTRACTS**

6. STRESS ANALYSIS AND SEISMIC CODES⁽¹⁾

Code	Library Number	Functional Area	Proprietary (P)	Operational Status	Documentation Status	Test Problem	Number of Cards	Machine Language	Input Codes	Relative Importance
ASAAS		stress in axisymmetric solids	-	A	A					
FESIC		graphite stress, dimensional change								
HELICON										
LIMITA II		transient dynamic analysis								
MARC		local pipe indentation stiffness	P	A	A	yes	large	no		A
MEC-21-7094		piping flexibility analysis	-	A	A	yes	-	no		A
NC CODE		piping system stress intensities								
NONLINEAR QUASI-STATIC ANALYSES		load analyses, concrete								
PIPE			-	A	A	yes	-	-		B
PIPESTRESS		computes stiffness matrix of piping system								
PISCES		stresses for large deformations	-	A	A					
PITRUST		stresses in pressure relief pipes								
SAASHI		stress in axisymmetric and plane solids	-	A	A					

(1) SAFESOLIDS and SAFE-2D (section 7) are also used in stress analysis of major HTGR structures.

Operational Status: production-released (A), production-not released (B), underdevelopment (C).

Documentation Status: report released (A), internal reports (B), underdevelopment (C).

Relative Importance for Safety Analysis: primary (A), alternate (B), minor support (C).

6. STRESS ANALYSIS AND SEISMIC CODES (Cont'd.)

Code	Library Number	Functional Area	Proprietary (P)	Operational Status	Documentation Status	Test Problem	Number of Cards	Machine Language	Input Codes	Relative Importance
SAFE		graphite stress,	P	B	B	-	1000	no	FESIC	
GRAFITE										
SAFESHELL		stress in axisymmetric shells	P	A	A	yes	1000	no		
SHELL 1		stress in thin symmetric shells								
SHOCK 3		seismic responses at each node of piping system								
STARDYNE		linear elastic structures	-	A	A				Stone & Webster Code	C
STRESS COMBINER		maximum piping stresses	.	.	.	.	.	.	not used at present	C
STRU DL II		structural engineering calculations								
TREVOR/STRESS		core temperature and stress distributions	P	C	C	-	4000	no		

Operational Status: production-released (A), production-not released (B), underdevelopment (C).

Documentation Status: report released (A), internal reports (B), underdevelopment (C).

Relative Importance for Safety Analysis: primary (A), alternate (B), minor support (C).

ASAAS

This is a finite element computer code. It can be used to determine stresses and displacements in arbitrary axisymmetric solids, including problems involving asymmetric mechanical and thermal loads and asymmetric temperature-dependent mechanical properties. All dependent variables, including the mechanical properties are input by Fourier series expansions of the circumferential coordinate. The mechanical loads can be surface pressures, surface shears, and nodal point forces.

The explicit stiffness relations for the axisymmetric solid ring elements of triangular cross section are based on the classical theorem of potential energy and the assumption that within any element the displacement variation in the R-Z plane is linear. All dependent variables, including the material properties are expanded into "Fourier Series". The harmonics are coupled and all the equilibrium equations are solved simultaneously. The algorithm used to solve the equations is a block modified square root Cholesky method with iterative refinement.

FESIC

The FESIC code computes the graphite axial and radial thermal and irradiation strains in a web at discrete points along the entire length of the core as a function of core lifetime. Besides strains, the code calculates the graphite axial stress distribution in a web and along the core length as a function of time. The stress calculations in FESIC are based on simplified conservative equations which include the effects of irradiation creep. At locations of maximum axial stress indicated by the FESIC code, a more detailed stress computation is performed with SAFE-GRAFITE.

LIMITA II - Mathematical Model

LIMITA II is a plane frame, nonlinear, transient dynamic analysis computer code. A plane frame is simulated as an assembly of discrete lumped masses connected by beam elements. Under any loading the equilibrium at each mass point is ensured by satisfying the individual equation of motion.

For the total structure, the governing equations of motion are solved by a linear acceleration integration method, starting from some known initial state of the system at time 0.0. The nonlinear effects, such as plasticity and large deflections are included.

The Von Mises yield surface is used along with the PRANDTL-REUSS flow relations. The stress strain curve is assumed isothermal B - linear with isotropic hardening and kinematic hardening models.

For large deflection analysis, the geometry is modified (if necessary) at the end of each load increment so that the total loading is applied to the deformed structure of the next load increment. This procedure thus follows the large deflection load-deflection curve.

MARC

The MARC nonlinear finite element analysis program, which is used to obtain the local pipe indentation stiffness, came into the public domain in December 1971. It is written in FORTRAN IV in a general form with variable dimensions passed down to the subroutines. A library of elements is available directly in the program.

The elastic-plastic and large displacement analysis is affected in a series of piecewise linear increments. Creep and thermal effects which cause initial strains are analyzed as a series of steps in which an increment of initial strain occurs at the start of each step. Optional facilities enable the lowest eigenvalue to be obtained after each applied increment of load. This eigenvalue furnishes the factor which must be used to scale the next increment of load to cause collapse.

Controls have been added which allow the specification of loading or creep for a total number of increments or time steps, respectively. These controls are referred to as automatic load controls. The automatic load control for creep selects the time step for each increment so that the resulting stress and strain changes remain within a specified limit. A higher order step-by-step integration in time, known as the residual load correction, may be specified for creep problems. This residual load correction feature stabilizes creep solutions.

The behavior is the classical theory of isotropic, elastic-plastic, time-independent materials, with a Mises yield criterion, isotropic strain hardening, temperature-dependent elastic properties, and equivalent yield stress. Perfect plasticity is assumed when no strain hardening is specified.

NCCODE

Internal moments/forces computed by SHOCK 3 as the seismic responses of the piping system are then combined with responses from deadweight, pressure, thermal, and all other mechanical loads to complete stress analysis of the seismic Category I and some nonseismic piping. For ASME Code Class 1 piping, stress intensities and cumulative usage factors of the piping system are computed by the computer program NCCODE.

NONLINEAR QUASI-STATIC ANALYSES OF TWO-DIMENSIONAL CONCRETE STRUCTURES

A finite element computer program capable of performing analyses of axisymmetric and plane structures in which applied loads, material properties, and section properties can vary independently with time. This program is used for viscoelastic and ultimate load analyses.

PIPE

PIPE is used for thermal weight and pipe flexibility analyses. This program adopts a tensor analysis treatment of Castigliano's theorem as the computational method and basically performs the following functions:

1. The program will compute the forces, moments, stresses, and translational and rotational deflections at all data points specified in the input and at the end of each bend. Stresses are computed according to the formulae provided in the piping code, ANSI B31.1.0-1967.
2. The loading may be due to thermal, pressure or weight effects, the results of which may be obtained individually or in combination.
3. There is no limitation to the geometry or complexity of the piping system. The analysis can include the effects of extraneous anchor or restraint movements, the flexibility at expansion joints or anchors, and of components such as valves, fittings, etc.

4. The computations are accurate to at least three significant figures.
5. The physical properties and material constants may be changed anywhere on the system to accommodate changes in piping materials, specifications, etc.
6. The flexibility and stress intensification factors recommended by the code are applied at bends and forged tees.
7. Hanger and support loadings can be accurately obtained. From this data, suitable hangers and supports may be selected.
8. The program checks the input data and piping geometry and will terminate the problem if they are not plausible. Lengths of tangents, components, etc., and angles of bends are indicated on the output.
9. Isometric sketches may be obtained from a plotting machine, via a tape generated by the program.

PIPESTRESS

Seismic analyses of seismic Category I and some nonseismic piping, which includes all ASME Code Classes 1, 2, and 3 piping systems, are performed by the modal analysis response spectra method. Each piping system is idealized mathematically as an elastically coupled dynamic structural model in three-dimensional space. Inertial characteristics of the piping system are simulated by discrete masses of piping components, including all eccentric masses, such as valves, valve operators, etc., lumped at selected nodes. The stiffness matrix of the piping system is calculated by the computer program PIPESTRESS. Modal seismic responses at each node of the piping system, due to amplified response spectra excitation applied at its support points, are calculated by the computer program SHOCK 3. The modal analysis technique used in the SHOCK 3 computer program computes the peak response quantities for each mode. Normal mode, linear elastic, and small displacement theory are incorporated in SHOCK 3 and PIPESTRESS computer programs.

PISCES - Violent Stress Program

The PISCES system of codes is a set of static and dynamic finite difference codes based on computational methods developed by the Atomic Energy Commission in connection with the calculation of nonlinear, large amplitude responses of structures, fluid bodies, and solid media for the design of nuclear devices. The codes solve the fundamental partial differential equations of continuum mechanics expressed in the explicit difference forms developed by Wilkins for Lagrange codes. Complex geometries, time dependent boundary conditions, space dependent initial conditions, and nonlinear material models are entered into the codes with no degradation in code efficiency or accuracy.

For all the codes, the space continuum is subdivided into zones or cells. For the dynamic codes, the time continuum is divided into time steps while the static codes used the time steps as an iteration variable. In the Lagrange formulations, each zone contains a constant mass element of material that moves and distorts in space and time (i.e., the zone volume may change). The motion for all the zones approximates the continuum motion.

PITRUST

Maximum stress intensity in the run pipe or header under full discharge loads (thrust, bending, and torsion) and internal design pressure is computed by the computer program PITRUST. The PITRUST computer program is based upon Bijlaard's method of calculating local stresses and experimental results.

SAAS III

This is a finite element computer code. It can be used to determine displacements, stresses, and strains in axisymmetric and plane solids with different orthotropic, temperature-dependent material properties in tension and compression including the effects of internal pore fluid pressures and thermal stresses. The mechanical loads can be surface pressures, surface shears, and nodal point forces, as well as acceleration or angular velocity.

The explicit stiffness relations for the two-dimensional solid elements of triangular cross section are based on the classical theorem of potential energy and assumption that within any element the displacement variation in the plane is linear. The stiffness matrix is assembled in a band form. A simplified Gaussian elimination method, which takes advantage of the concentration of the elements of the stiffness matrix along the main diagonal, is the algorithm used to solve the equilibrium equations.

SAFE-GRAFITE

The finite element computer code SAFE-GRAFITE is used in the stress analysis of graphite elements. This code can solve both elastic and viscoelastic problems. The viscoelastic option can be used to find the effects on stresses and strains of the irradiation induced creep in graphite. Calculations are often made first with FESIC to locate points of maximum stress; more detailed calculations are then made with SAFE-GRAFITE.

SAFE SHELL

SAFE SHELL is a general purpose code used to design and analyze axisymmetric thin shell structures of arbitrary generatrices under axisymmetric mechanical and/or thermal loading conditions. The intersection of two or more shells can be treated. The analysis is limited to elastic, static conditions. The code has been used for a variety of HTGR components such as reactor lines sections, etc.

SHELL 1

This is a finite difference stress analysis computer code. It can be used to determine the forces, moments, shears, displacements, rotations, and stresses in a thin shell of revolution subject to arbitrary loads expanded in Fourier series of up to 150 terms. Single-layer shells with up to 30 simply-connected branches may be analyzed. Poisson's Ratio may change at discontinuity points, and Young's Modulus and the thermal coefficient of expansion may be different at each point. The allowed types of loading include elastic restraints, pressures in three orthogonal directions, temperature changes which have a gradient through the shell thickness, and simplified input for weight of the shell or earthquake forces.

The equilibrium equations for a thin shell are based on the linear theory of Sanders. Sanders' equations are expanded and modified slightly to handle a broader range of problems. All pertinent load, stress, and deformation variables are expanded into Fourier Series. The individual Fourier components of stress and deflection are found separately by solution of the finite-difference forms of the appropriate differential equations. The algorithm used to solve these equations is a minor modification of the Gaussian elimination method.

STARDYNE

The MRI STARDYNE Analysis System consists of a series of compatible digital computer programs designed to analyze linear elastic structural models. The system encompasses the full range of static and dynamic analyses. The static capability includes the computation of structural deformations and member loads and stresses caused by an arbitrary set of thermal, modal applied loads, and prescribed displacements. Utilizing the normal mode technique, dynamic response analyses can be performed for a wide range of loading conditions, including transient, steady-state harmonic, random and shock spectra excitation types. Dynamic response results can be presented as structural deformations and internal member loads.

STRESS COMBINER

STRESS COMBINER computes piping stress intensities in a similar fashion to NCCODE, but for ASME Code Class 2 and Class 3 piping.

STRUDL II

STRUDL, an acronym for Structural Design Language, is a series of computer programs for solving problems in structural engineering. It is applicable to both framed and continuous mechanics problems, either static or dynamic in nature. In frame analysis, the member stiffness matrix is computed from beam theory, while in finite element analysis, the element stiffness matrix is computed from energy considerations.

TREVOR/STRESS

Uses power distributions from a two-dimensional code such as GAUGE to obtain complete core temperature distributions. Using stress analysis subroutines, a survey of the core is made to determine maximum stresses in graphite blocks. More accurate local stress analysis calculations are then made to obtain failure information.

SECTION 7

**MECHANICAL COMPONENT DESIGN
CODE ABSTRACTS**

7. MECHANICAL COMPONENT DESIGN⁽¹⁾

Code	Library Number	Functional Area	Proprietary (P)	Operational Status	Documentation Status	Test Problem	Number of Cards	Machine Language	Input Codes	Relative Importance
AXIPACE		axisymmetric elastic	P	B	A	yes	1700	no	A	A same for unequal
DSAP		dynamic elastic response	P	B	A	yes	1000	no	A	
DYNAM		stability of boiling flow	-	A	A	yes	200	no	A	
EQUAL		response spectrum	P	B	A	yes	<1000	no	A	
FLAC*		incompressible flow	-	A	A	yes	500	no	A	
GENSIZE		steam generator design	P	A	B	no	5000	no	C	
SAC		stress and deformation	P	B	A	yes	400	no	A	
SAFE CRACK**		viscoelastic, crack, plastic	-	A	A	yes	3200	no	A	
SAFE CREEP		viscoelastic analysis	-	A	A	yes	5700	no	C	
SAFE PLANE		2-dimensional elastic analysis	-	A	A	yes	1100	no	C	
SAFE SOLIDS		3-dimensional elastic	P	A	A	yes	3000	no	B	
SAFE-2D		2-dimensional elastic	-	A	A	yes	1900	no	C	
SAFE-3D		3-dimensional elastic analysis	-	A	A	yes	5200	no	C	
SHAKE		standard dynamics	P	B	A	yes	1000	no	A	not used at present
SHELL-3D		3-dimensional elastic shells	-	A	A	yes	3400	no	B	
STAC		temperature and thermal stress	P	B	A	yes	4400	no	A	

(1) The general heat transfer codes TAC2D, TAC3D, and CINDA are also used for mechanical component design. These codes are described in section 3. The stress analysis codes ASAAS, SHELL 1, SAAS III, STARDYNE, MARC, LIMITA II, PISCES, STRUDL II, ME89, and ME112 are also used. These codes are described in section 6.

*Some later proprietary versions may exist.

**3-dimensional version under development.

7. MECHANICAL COMPONENT DESIGN (Cont'd.)

Code	Library Number	Functional Area	Proprietary (P)	Operational Status	Documentation Status	Test Problem	Number of Cards	Machine Language	Input Codes	Relative Importance
STRESS ANALYSIS OF SHELLS OF REVOLUTION	shell stress analysis									
SUPERHEAT	steam generator performance	P	A	B	yes	8000	no		C	
TEPC-2D	thermal, elastic, plastic, creep	P	B	A	yes	2000	no		A	
UNEQUAL	response spectra	P	B	A	yes	4000	no		A	
Vessel Penetration Analysis	local penetration stresses									

Operational Status: production-released (A), production-not released (B), underdevelopment (C).

Documentation Status: report released (A), internal reports (B), underdevelopment (C).

Relative Importance for Safety Analysis: primary (A), alternate (B), minor support (C).

AXIPACE

A higher order polynomial finite element program for the elastic analysis of heterogeneous two-dimensional plane and axisymmetric bodies of arbitrary geometry. Any mixture of triangular membrane and triangular ring elements may be specified. Both types use a truncated cubic polynomial displacement field shape function with vertex node degrees of freedom consisting of the in-plane displacement magnitudes and first derivatives. Loads consisting of plane or axisymmetric piecewise linear temperature fields defined by node temperatures, node forces and moments, and linearly varying surface pressure can be accommodated. The INFMAT matrix solution routine, which allows for extremely fast solutions for certain geometry types, is used for the stiffness matrix decomposition.

DSAP

A program to determine the time-history dynamic response of linearly elastic structural systems subjected to arbitrary time-varying forces with time-varying constraints placed at arbitrary points on the system. The structural system must first be described by lumped parameter techniques and have modal properties that can be determined by solution of the generalized eigenvalue problem. The eigenvalues and eigenvectors (natural frequencies and mode shapes), which can be obtained from SHAKE or other similar eigenvalue subroutines, are required input for DSAP together with tables of time varying forces, constraints, and/or base acceleration, and their respective geometric locations. The normal mode method and Adams' numerical integration method with controlled relative error are used to integrate a system of simultaneous differential equations.

DYNAM

A program for calculating the stability characteristics of once-through boiling flow (including nonboiling subcooled, nucleate boiling subcooled, bulk or two-phase boiling and superheat regimes) in a single channel. A set of algebraic equations obtained through application of control theory procedures to the governing equations (continuity, momentum, and energy) are efficiently solved by a noniterative procedure. The result is expressed in terms of absolute stability or instability; the margin of stability is not obtained.

EQUAL/UNEQUAL

Programs that calculate response spectra from strong-motion acceleration records, as well as floor response spectra for appendage systems. (One program is for acceleration records digitized at equal time intervals, the other for unequal time intervals.) The procedure for the numerical determination of the response spectra is based on the exact analytical solution to the governing second-order differential equation of motion for a single degree of freedom system. The method gives a three- to four-fold saving of computer time compared to a third order Runge-Kutta method of comparable accuracy.

FLAC

A program for calculating steady incompressible flow distribution in arbitrary flow networks with heat addition. The set of simultaneous equations defining flow, pressure, and temperature distribution in the network are solved by a variation of the point relaxation method.

GENSIZ

A computer program used for sizing once-through steam generators. Depending on the conditions specified, the program can determine the number of tubes required, the tube length, the tube diameter, the shell and tube side pressure drops, and the fluid inventory. Tube wall thicknesses are also calculated according to Sections III and VIII of the ASME Boiler and Pressure Vessel Code. Included in the program are the thermodynamic tables for steam/water and the heat transfer and pressure drop correlations for water, nucleate boiling, post-nucleate boiling, and steam.

SAC

A computer program for stress and deformation calculations of axisymmetric thin-shell structures, including pressure vessels, which can be modeled as simple shell components without axial redundancy. The components which can be used are uniform cylinders, tapered cylinders, plates, rings, domes, and dummy elements. The solution method is based on the classical method of redundants.

SAFE-CRACK

An ultimate load finite element program for the viscoelastic, crack, and plastic analysis of two-dimensional (plane or axisymmetric) reinforced and/or prestressed concrete structures with arbitrary plane geometry subjected to mechanical, thermal, and displacement loading conditions. Continuum elements represent the concrete as a linear viscoelastic material whose response is a function of the previous stress history and the age and temperature of the concrete. Biaxial membrane and uniaxial tension-compression elements used to represent liners and rebar are assumed to be elastic-perfectly plastic. Brittle concrete cracking is determined by examining the stress and strain states of the concrete elements.

SAFE-CREEP

SAFE-CREEP performs a viscoelastic analysis of plane or axisymmetric composite concrete structures with age- and temperature-dependent creep data. The program uses the finite element method which reduces the problem to the solution of a system of coupled integral equations. The method of solution is iterative on the space variable and utilizes the trapezoidal rule integration for the time variable.

SAFE-PLANE

SAFE-PLANE is applied to two-dimensional structures of arbitrary geometry under in-plane loads. Either plane stress or plane strain conditions may be imposed. Mechanical and/or thermal loads are permitted. The finite element method is used to construct a mathematical model by assembling discrete elements. The total potential energy of the structure is determined and subsequently minimized by iteration on components of the displacement field until static equilibrium of the structure is attained. Strains and stresses are computed from the resulting displacements.

SAFE-SOLIDS

A finite element program for the analysis of elastic, three-dimensional heterogeneous composite structures specifying either linear or higher order displacement shape functions or a combination of the two. Tetrahedral, triangular plate and uniaxial beam elements are available with compatible cubic displacement shape functions or linear shape functions. In the latter case, the triangular plane element degenerates to a membrane state and the beam to a bar. Any mixture of cubic and linear temperature fields specified by node temperatures, point forces and moments, and surface pressures. The INFMAT program is used for the matrix solution, allowing an extremely large capacity. Primarily used for final analyses and verification.

SAFE-2D

A finite element program for the elastic analysis of heterogeneous plane, axisymmetric, and combined plane and axisymmetric structures. There are four elements available: triangular rings, triangular plane membrane elements, conical membrane elements, and uniaxial bar elements. The element displacement fields are assumed to be linear. Thermal loading, in-plane and axisymmetric mechanical point forces, and surface pressure can be accommodated. The equations are solved by a direct tridiagonalization method. Primarily used for preliminary analyses.

SAFE-3D

SAFE-3D is a finite-element program for the three-dimensional elastic analysis of heterogeneous composite structures. The program uses the following types of finite elements: 1) tetrahedral elements to represent the continuum, 2) triangular plane stress membrane elements to represent inner or outer case, and 3) uniaxial tension-compression elements to represent internal reinforcement. The structure can be of arbitrary geometry and have any distribution of material properties, temperatures, surface loadings, and boundary conditions.

SHAKE

A program for the determination of the dynamic response of lumped-mass systems with interconnecting massless flexible members. Three-dimensional systems with rotatory inertias and both internal and external redundant members may be treated. Excitation may be sinusoidal base motion, sinusoidal forces, response spectrum, or a step change in velocity. Natural frequencies and mode shapes are determined from a mass-flexibility matrix, and loads and deflections are computed assuming modal damping.

SHELL-3D

A general purpose finite element program in two versions for the elastic analysis of arbitrary three-dimensional thin shells subjected to mechanical, thermal and displacement loading conditions. A shell is idealized as an assemblage of triangular plate elements whose membrane and bending deformations are approximated by polynomial functions. A five degree of freedom version (linear membrane and cubic bending functions) called SHELL 5, and a nine degree of freedom version (cubic membrane and cubic bending functions) called SHELL 9, provide flexibility in idealizing complex shell structures. SHELL 5 results in 3 displacements and 2 rotations and is recommended for smoothly curved (nonintersecting) shells. SHELL 9 results in 3 displacements, 3 rotations, and 3 membrane strains and is recommended for arbitrarily intersecting shells such as branch tees.

STAC

A computer program for the calculation of transient and steady-state temperature and thermal stress distribution in axisymmetric thin-shell structures. This program is a combination of TAC-2D and SAC, and its use can result in extremely rapid solutions to thermal stress problems in components such as thermal sleeves, nozzles, etc. On-line printer plots may be generated to plot stress histories through transients to obtain the most severe stress range.

Stress Analysis of Shells of Revolution

This is a finite element computer code. It can be used to determine the forces, moments, shears, displacements, rotations and stresses in a thin shell of revolution subject to axisymmetric loads. Different orthotropic material properties may be input for each element in a model. The allowed types of loading include internal pressure, temperature changes which may have a gradient through the shell thickness, and simplified input for weight of the shell.

The explicit stiffness relations for the axisymmetric shell elements are based on the classical theorem of potential energy and the usual approximations of thin shell theory. The direct stiffness method (a simple modification of the displacement method) is employed to assemble the equilibrium equations. The algorithm used to solve these equations is derived by applying the Gauss-Jordan method of elimination to a tridiagonal system of equations.

SUPERHEAT

A program for the performance analysis of heat exchangers consisting of many parallel sections. The program is capable of handling shell-side mixing effects. The analysis is limited to heat exchangers with the same number of tubes in each pass of each section.

TEPC-2D

A finite element program for the thermal-elastic-plastic and nonlinear creep analysis of plane and axisymmetric structures. The program considers thermal, mechanical, and displacement loadings which may vary with time in an arbitrary manner and calculates the resulting temperature and/or stresses. The creep behavior of the material is represented by a nonlinear stress-strain law which is formulated from the theory of the mechanical equation of state. Material hardening may be either isotropic or kinematic, or follow a more general nonlinear kinematic hardening rule. The solution technique is formulated on the initial strain approach to problems in creep and plasticity. The Cholesky method is used to decompose the equilibrium equations. Loading and unloading may be considered, so the program may be used to trace stress and strain histories through various operating cycles.

Vessel Penetration Analysis

This computer code performs various analyses on tanks and pressure vessels. All of the analyses are concerned with local stresses at penetrations. Typical problems which can be handled include the following:

1. Applied load stresses at vessel-nozzle junction for:
 - Rigid attachment to cylinder
 - Rigid attachment to sphere
 - Hollow attachment to sphere
2. Pressure discontinuity analysis for thin shell interaction.
3. Allowable load functions on nozzles for each case.
4. Area compensation analysis in accordance with ASME Code.
5. Maximum forces on supports of vessel based on allowable loads on nozzles.

Local stresses due to nozzle loads are found by the method of P. P. Bijlaard. The method of "Johns and Orange" is used for pressure discontinuity stresses.