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ORNL-4968

PLANNING GUIDE
for
HTGR SAFETY
and
SAFETY-RELATED RESEARCH and DEVELOPMENT

MAY 1974

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Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22151
Price: Printed Copy \$10.00; Microfiche \$1.45

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ORNL-4968
UC-77 - Gas Cooled Reactor Technology

Contract No. W-7405-eng-26

REACTOR DIVISION

PLANNING GUIDE FOR HTGR SAFETY AND
SAFETY-RELATED RESEARCH AND DEVELOPMENT

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May 1974

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FOREWORD

This document was prepared by Oak Ridge National Laboratory with contributions from General Atomic Company and the AEC. It identifies work areas and specific tasks related to determining and understanding the safety of high-temperature gas-cooled reactors (HTGRs) in a comprehensive manner; priorities assigned to the various task groups are also listed for guidance in assigning support. It is recognized that, in some instances, design conservatism can substitute adequately for the technology development data discussed herein without adversely affecting overall reactor safety. It should be noted that HTGR owners and manufacturers have full responsibility for safe design, licensing efforts and operation of their plants and for defining and obtaining the technical data needed to execute these responsibilities. Although this guide may be helpful to them in executing these responsibilities, it does not represent the specific development plan or requirements of any manufacturer or operator.

It is anticipated that the Division of Reactor Research and Development will use this planning guide as a source document in planning HTGR safety work consistent with programmatic objectives and resources. The specific work supported will be based on assessment of work priorities and availability of data from industry. Publication and use of parts or all of this guide does not imply a judgement by the AEC that work described in this document is either necessary or sufficient to establish the safety of HTGRs, nor does publication of this planning guide imply any commitment by the government to fund any or all of the research and development work discussed.

ABSTRACT

Additional safety information and safety-related engineering and technology are desirable to ensure better understanding and more accurate definition of the safety characteristics and safety margins under postulated accident conditions for high-temperature gas-cooled reactors (HTGRs) being built and planned in the United States. This report identifies both safety and safety-related work that could be done to further refine the technology and is intended to serve as a planning guide.

The work is divided into seven task areas. The first of these, task area 1, addresses all aspects of systems and safety analysis for HTGRs while the remaining six deal with the following areas of technology:

Task area 2 - Fission-product technology,

Task area 3 - Primary coolant technology,

Task area 4 - Seismic and vibration technology,

Task area 5 - Confinement components (concrete reactor vessel and containment structures),

Task area 6 - Materials technology, and

Task area 7 - Safety instrumentation.

In all of the areas, work is defined on the basis of current understandings of the technology, with the recognition that changes may occur as additional information is developed.

All of the work in task area 1 is described in the safety planning guide. It provides for studies of system and component reliability, analyses of accident event sequences, modeling of component and plant transient responses, and performance of transient analyses. This work would incorporate and consolidate technology from the other areas.

In the safety planning guide (Part II), a principal objective of the work identified in the remaining six technology areas is a detailed and continuing assessment of the state of the technology and the identification of any additional needs for safety research and development. In addition, specific development tasks are identified in many of the task areas. In task area 2, the added activities deal with potential fuel and control-material migration at high

temperatures, the lift-off and transport characteristics of deposited fission products in the primary loop, and the behavior of fission products in the containment. No activities are currently identified in task area 3 beyond the assessment of needs. Four developmental task groups are identified in task area 4, dealing with seismic disturbances and their effects in terms of core response, component response, soil-structure dynamic interactions, and PCRV-support responses. Work is identified in task area 5 on failure analysis of prestressed concrete reactor vessels (PCRVs). In task area 6, activities are identified to improve the understanding of the chemical and structural stability of reactor control materials, particularly boronated graphite, in HTGR environments. No activities are identified in task area 7 beyond the assessment of possible additional needs.

The safety-related planning guide (Part III) identifies a variety of activities, including the work described in the safety planning guide, in each of the six technology task areas. However, it is presumed that most, if not all, of the safety-related work will be carried on outside the HTGR safety program. The work identified in task area 2, beyond that discussed in the safety planning guide, is covered in five task groups dealing with: fission-product release from fuels, fission-product behavior in graphite, the surface chemistry of fission-product behavior in the primary circuit, tritium behavior, and fuel-particle failure mechanisms. Additional work in task area 3 deals with: steam reactions with core materials, along with the effects of coolant composition on such reactions, and the compatibility of primary-system components with the coolant and its potential impurities. An additional task group in task area 4 is concerned with vibrations of primary-system components such as those that may be induced by gas flows. Task area 5 includes study of eight additional task groups (PCRV head failures, concrete properties, prestressing components and materials, PCRV liners and penetrations, thermal barriers, special PCRV instrumentation, PCRV models, and containment convection) as well as development of code rules for prestressed concrete vessels. The additional development work in task area 6 involves further studies of control materials, graphite components, and metallic materials to determine safety-related properties and behavior in HTGRs. Eight additional task groups in task area 7 are concerned with development aspects — failed-fuel location systems, HTGR thermometry, in-core flux monitoring systems, surveillance and diagnostic procedures using noise analysis and dynamic testing, reactor-coolant moisture-monitoring systems, data acquisition and processing systems, post-accident monitoring systems, and plant controllability in relation to safety.

Included throughout the planning guides are activities that would be associated with a Peach Bottom HTGR end-of-life program and the decommissioning of that system. These activities are summarized in an appendix.

1. SUMMARY

Recent commercial sales of high-temperature gas-cooled reactors (HTGRs) have served to emphasize the need for an active and expanded HTGR Safety Research Program. By 1980 domestic HTGR utility sales are predicted to total approximately 50 units, and by the year 2000 the projected installed HTGR generating capacity will be in the hundred-thousand MWe range. Whether such large-scale use of HTGRs will take place will in part be determined by the results of safety evaluations. Thus, it is important that the AEC and industry have timely information with which to evaluate HTGR safety requirements. Specific areas where additional information is desirable have been identified by members of both the AEC staff and the Advisory Committee on Reactor Safeguards (ACRS). This report presents a planning guide for safety and safety-related research and development to provide the needed information.

1.1 ORGANIZATION OF THE PLANNING GUIDE

This document is divided into three major parts. Part I provides general background information for the reader who is not familiar with HTGRs and their safety aspects. It begins with a detailed discussion of the need, objectives, and scope of an overall safety program for HTGRs. This is followed by a general description of a steam-cycle HTGR system and a survey of the safety considerations associated with this system. Also included is a summary of the postulated accidents that have been examined and their expected consequences.

Part II provides a planning guide for research and development work directed explicitly to the safety of HTGRs. This part describes the work currently identified as safety work, and the priority assignments, costs, and schedules estimated for that work. Seven interrelated task areas are identified as follows:

- (1) systems and safety analysis,
- (2) fission-product technology,
- (3) primary coolant technology,
- (4) seismic and vibration technology,
- (5) confinement components (concrete reactor-vessel and containment structures),

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- (6) materials technology,
- (7) safety instrumentation.

Within each task area, task groups are designated which define the studies to be performed. An important aspect of the safety work in each task area is a comprehensive assessment of the current state of the technology and a definition of additional needs; in some areas this is the only work that is currently identified. However, additional safety tasks may be added in any of the task areas as additional information becomes available.

Part III presents a comprehensive description of the safety-related work, as well as the safety work, that can currently be identified with HTGR technology, with the exception that task area 1 is not repeated in Part III because it is treated in its entirety in Part II. For the remaining six task areas, Part III gives both safety-related engineering and technology work and the previously identified safety work to clarify the interrelations among the various tasks.

1.2 BACKGROUND (PART I)

The objective of the overall safety and safety-related planning guide is to outline the activities associated with (1) assessing the status and further needs for HTGR safety technology, (2) estimating the consequences and likelihood of accidents, and (3) producing through research and development the technology needed for the practical evaluation of HTGR safety. The relevant research and development involves basic technology such as the behavior of fission products, reactor materials, heat transfer systems, and containment systems. Further, the identified work requires information on the inherent characteristics of HTGR systems as well as on the characteristics associated with specific designs, and it includes the study of component and system responses to accident situations. Normal and near-normal operating conditions of components and fission-product-control systems are also considered to the extent that their study establishes a basis for predicting the consequences and likelihood of accidents. Finally, also included, insofar as they are safety related, are experimental and theoretical investigations of basic properties and mechanisms; development of methods for the description of fission-product, material, and component behavior; and large-scale tests that may be required to confirm the expected behavior of a specific component or system.

The HTGR is a thermal-neutron-spectrum reactor, uses helium as the coolant and graphite as the moderator, and operates at high temperatures. The reactor core and the coolant loops are contained within a prestressed concrete reactor vessel (PCRVR); and the PCRVR is itself within a containment building. The solid graphite moderator, which completely encloses the fuel, and the gaseous coolant, along with specific design features, combine to give HTGRs safety-related characteristics that differ in many respects from those of light-water reactors.

Inherent and design features of HTGRs which help to mitigate the consequences of postulated accidents include the high-temperature thermal and mechanical stability of the fuel, moderator, and poison materials; the high heat capacity of the fuel-moderator combination; the low heat capacity and nuclear reactivity worth of the coolant; the negative temperature coefficient of reactivity of the core; and the provision of a redundant shutdown system and an independent post-shutdown heat removal system. Other factors to be considered include the absence of metallic cladding of the fuel elements and the tendency for hot carbon to react with moisture that may enter the coolant stream.

All of the characteristics of these systems need to be considered in detailed analyses of postulated accidents to define safety behavior. Accidents that have been considered range from relatively probable events — such as loss of off-site power or turbine trip — to much less likely events with potentially greater consequences. Among the latter are loss of main loop cooling capability, water ingress to the primary system, and primary system depressurization. Analyses of such postulated events have led to the development of numerous mathematical models and computer codes (primarily by the General Atomic Company, the principal U.S. vendor of HTGRs) to describe system behavior. Technology existing or being developed indicates that large HTGR plants can be built and operated safely; however, additional understanding may be required to ensure both adequate safety and acceptable power generation costs.

1.3 SAFETY PLANNING GUIDE (PART II)

Work is identified in the safety planning guide in all of the seven task areas based on current understanding of the status of the technology and recognized needs. These activities are summarized by task group in Table 1.1, which also gives the estimated cost of the work by fiscal year for a 10-year period. All the task areas include a task group for assessment of needs (implicit in all groups of task area 1); work may be added as the safety needs become more clearly defined with the development of additional

Table 1.1. Summary of identified safety work and estimated costs

Task group	Costs (\$1000) by fiscal year										10-year total
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
Task area 1 - Systems and safety analysis											
1.1 Component qualification	60	120	120	120	120	120	120	120	60	60	1,020
1.2 Analysis of event sequences	60	120	120	120	120	60	60	30	30	30	750
1.3 Modeling of plant transient behavior	300	450	480	480	480	450	390	330	270	240	3,870
1.4 Safety analysis and evaluation	90	300	360	300	300	300	300	180	180	120	2,430
Equipment											
Area total	510	990	1080	1020	1020	930	870	660	540	450	8,070
Task area 2 - Fission product technology											
2.0 Need assessment	180	180	180	180	180	180	180	180	180	180	1,800
2.1 Fuel and control material migration		60	60	60							180
2.2 Fission product behavior in the coolant circuit - engineering-scale dynamics	120	340	210	180	120	60					1,030
2.3 Fission products in the containment	30										30
Equipment	50	275	100								425
Area total	380	855	550	420	300	240	180	180	180	180	3,465
Task area 3 - Primary coolant technology											
3.0 Assessment of primary coolant technology	55	60	60	60	30	30	30	30	30	30	415
Equipment											
Area total	55	60	60	60	30	30	30	30	30	30	415
Task area 4 - Seismic and vibration technology											
4.0 Need assessment	50	50	50	50	50	50	25	25	25	25	400
4.1 Core seismic response	735	1430	1800	2050	1450	1050	500	50			9,065
4.2 Component seismic response	50	250	500	800	600	300					2,500
4.3 Soil-structure dynamic interaction	70	150	250	250	100						820
4.4 PCRV support		180	400	500	400	400	300	200	100		2,480
Equipment	810	1705	2125	1000	1000	700	400				7,740
Area total	1715	3765	5125	4650	3600	2500	1225	275	125	25	23,005

Table 1.1. (continued)

Task group	Costs (\$1000) by fiscal year										10-year total
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
Task area 5 – Confinement components											
5.0 Need assessment	50	50	50	50	50	25	25	25	25	25	375
5.1 Failure analysis	25	25	25	25	50	50	50	50	50	25	375
Equipment											
Area total	75	75	75	75	100	75	75	75	75	50	750
Task area 6 – Primary system materials technology											
6.0 Need assessment	50	50	50	60	60	60	70	70	70	70	610
6.1 Control materials	240	260	260	60							820
Equipment											
Area total	290	310	310	120	60	60	70	70	70	70	1,430
Task area 7 – Safety instrumentation											
7.0 Need assessment	450	300									750
Equipment											
Area total	450	300									750
Total	3475	6355	7200	6345	5110	3835	2450	1290	1020	805	37,885

information. Costs beyond the first few years should be regarded as tentative because actual costs and schedules will depend on year-to-year results as well as on the rate at which HTGRs are introduced into the power economy.

In addition to the identification of possible activities, the safety planning guide presents qualitative designations of the relative priorities of the tasks that are listed. This priority rating is intended primarily to indicate areas in which additional support for safety research and development could significantly enhance the evolution of HTGR safety technology. The priority designations are A, B, C, or D, based on current judgments and the following definitions:

- A Tasks in this category should be supported on the suggested time schedule and at essentially the suggested level if undesirable delays in meeting program goals are to be avoided. Assignment to this category implies that the importance of the task and the amount of work required are such that additional support could be accommodated to expedite the attainment of goals.
- B Tasks in this category should be supported on the suggested time schedule, but lower levels of support may be acceptable for a limited period without undesirable delays in overall program goals. Benefits of increased support are likely to be marginal in most cases.
- C Tasks in this category are desirable, but either their effect on other tasks is relatively small or time is not a critical consideration for development of the information. These tasks could be delayed or supported at lower levels temporarily. Increased support is not warranted at present.
- D Tasks in this category are identified for consideration, but support does not appear warranted at this time.

Estimates of work priorities are given in Tables 1.2 through 1.8; in those cases where insufficient information is currently available for reasonable assignment of priorities no assignments were made.

1.3.1 Systems and safety analyses (Task area 1)

This task area evaluates potential accidents by considering the consequences of specific events and accident sequences. The primary concern is the ability to contain radioactivity under all credible conditions. Thus, task area 1 focuses on the central

Table 1.2. Summary of identified safety work in task area I

Task description	Task priority	Costs (\$1000) and manpower (man-years in parentheses) by fiscal year										10-year total
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
Task area 1 Systems and safety analysis												
Task group 1.1 Component qualification		60 (1.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	60 (1.0)	60 (1.0)	1020 (17.0)
1.1.1 Source tasks												
1.1.1.1 Component reliability experience	B											
1.1.1.2 Component design and duty qualification	B											
1.1.1.3 Component application in systems	B											
1.1.2 Synthesis task												
1.1.2.1 Component response evaluation	B											
Task group 1.1 total		60 (1.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	60 (1.0)	60 (1.0)	1020 (17.0)
Task group 1.2 Analysis of event sequences		60 (1.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	60 (1.0)	60 (1.0)	30 (0.5)	30 (0.5)	30 (0.5)	750 (12.5)
1.2.1 Source tasks												
1.2.1.1 Systems functional logic	A											
1.2.1.2 Accident-initiating situations	B											
1.2.1.3 Confidence analysis methods	C											
1.2.1.4 Common mode failure studies	A											
1.2.2 Synthesis task												
1.2.2.1 Accident logical sequence models	A											
Task group 1.2 total		60 (1.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	60 (1.0)	60 (1.0)	30 (0.5)	30 (0.5)	30 (0.5)	750 (12.5)
Task group 1.3 Modeling of plant transient behavior												
1.3.1 Source tasks												
1.3.1.1 Neutronic characteristics	A	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	90 (1.5)	60 (1.0)	1110 (18.5)
1.3.1.2 Component and subsystem characteristics	B	60 (1.0)	90 (1.5)	120 (2.0)	120 (2.0)	120 (2.0)	90 (1.5)	60 (1.0)	60 (1.0)	30 (0.5)	30 (0.5)	780 (13.0)

Table 1.2 (continued)

Task description	Task priority	Costs (\$1000) and manpower (man-years in parentheses) by fiscal year										10-year total
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
1.3.1.3 Physical and chemical process models	A	60 (1.0)	60 (1.0)	60 (1.0)	60 (1.0)	60 (1.0)	60 (1.0)	30 (0.5)	30 (0.5)	30 (0.5)	30 (0.5)	480 (8.0)
1.3.1.4 Consolidated plant models review	A	60 (1.0)	180 (3.0)	180 (3.0)	180 (3.0)	180 (3.0)	180 (3.0)	180 (3.0)	120 (2.0)	120 (2.0)	120 (2.0)	1500 (25.0)
1.3.2 Synthesis tasks												
1.3.2.1 Neutronic behavior models	B	Effort included in source tasks										
1.3.2.2 Component and subsystem behavior models	B											
1.3.2.3 Physical and chemical process models	A											
1.3.2.4 Consolidated plant models	B											
1.3.3 Validation tasks												
1.3.3.1 Neutronic transient models	B	Effort included in source tasks										
1.3.3.2 Component and subsystem transient behavior	B											
1.3.3.3 Physical and chemical process transient behavior	A											
1.3.3.4 Plant transient behavior	B											
Task group 1.3 total		300 (5.0)	450 (7.5)	480 (8.0)	480 (8.0)	480 (8.0)	450 (7.5)	390 (6.5)	330 (5.5)	270 (4.5)	240 (4.0)	3870 (64.5)
Task group 1.4 Safety analyses and evaluation												
1.4.1 Source task												
1.4.1.1 Review of available analyses and results	A	60 (1.0)	180 (3.0)	180 (3.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	60 (1.0)	60 (1.0)	60 (1.0)	1080 (18.0)
1.4.1 Synthesis tasks												
1.4.2.1 Analyses and sensitivity studies	B	30 (0.5)	120 (2.0)	180 (3.0)	180 (3.0)	180 (3.0)	180 (3.0)	180 (3.0)	120 (2.0)	120 (2.0)	60 (1.0)	1250 (22.5)
1.4.2.2 Definition of additional needs	B	Included in task 1.4.1.1										
Task group 1.4 total		90 (1.5)	300 (5.0)	360 (6.0)	300 (5.0)	300 (5.0)	300 (5.0)	300 (5.0)	180 (3.0)	180 (3.0)	120 (2.0)	2430 (40.5)
Task area 1 total		510 (8.5)	990 (16.5)	1080 (18.0)	1020 (17.0)	1020 (17.0)	930 (15.5)	870 (14.5)	660 (11.0)	540 (9.0)	450 (7.5)	8070 (134.5)

Table 1.3. Summary of identified safety work in task area 2^d

Task description	Task priority	Estimated costs (\$1000) and manpower (man-years in parentheses) by fiscal year										
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	10-year total
Task area 2 Fission product technology												
Task group 2.0 Need assessment for fission product technology.												
2.0.1 Goals for fission product technology. Establishment of program needs for fission-product retention based on accident considerations. Updating as required. Coordination of reporting of results by element.	A	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	1800 (30)
Task group 2.0 total costs		180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	1800 (30)
Task group 2.1 Fuel and control material migration												
2.1.1 Source tasks. Quantitative description of mechanisms by which control, fissile, and fertile materials might migrate under postulated accident conditions.												
2.1.1.1 Actinide adsorption and diffusion in graphites	C			60 (1)	60 (1)							120 (2)
2.1.1.2 Control material adsorption and diffusion in graphite	A		60 (1)	b								b
2.1.2 Synthesis tasks. Codes for analysis of movement of Th, B, and other control materials during accidents involving very high temperatures or steam and high temperatures.	b		b									b
Task group 2.1 total operating costs			60 (1)	60 (1)	60 (1)							180
Task group 2.2 Fission product behavior in the coolant circuit engineering-scale dynamics												
2.2.1 Source task												
2.2.1.1 Fission product distribution and re-entrainment. Description of aerodynamics of aerosols in HTGR with emphasis on deposition and re-entrainment in realistically complex geometries. Studies of aerosol adhesion to appropriate substrates as functions of material temperature and time.	B	120 (2)	120 (2)	120 (2)	150 (2.5)	120 (2)	60 (1)					690 (11.5)

Table 1.3 (continued)

Task description	Task priority	Estimated costs (\$1000) and manpower (man-years in parentheses) by fiscal year										10-year total
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
2.2.2 Synthesis task												
2.2.2.1 Fission product distribution code for dusty coolant circuit. Modification of PAD code to take account of role of dust in fission product distribution in primary circuit and formulation of dust re-entrainment model for accident analysis.	A		60 (1)	30 (0.5)	30 (0.5)							120 (2)
2.2.3 Validation tasks												
2.2.3.1 Re-entrainment experiments. Experimental determination of the mobility of solids adhering to surfaces of the primary loop Peach Bottom HTGR. Correlation with need and expectation.	A		160 (2.5)	60 (1)								220 (3.5)
2.2.3.2 Fission product-dust distributions in HTGRs. Measurements of distributions of deposited fission products on steam generators, ducting, and circulators in or from Fort St. Vrain or larger HTGRs. Development of appropriate remote measurement techniques.	a											
Task group 2.2 total operating costs		120 (2)	340 (5.5)	210 (3.5)	180 (3)	120 (2)	60 (1)					1030 (17)
Task group 2.3 Fission products in the containment												
2.3.1 Source task												
2.3.1.1 Review of iodine behavior in containment. Review of work done under other programs and determination of need for further refinement of description of iodine behavior in HTGR containment, including concrete.	C	30 (0.5)										30 (0.5)

Table 1.3 (continued)

Task description	Task priority	Estimated costs (\$1000) and manpower (man-years in parentheses) by fiscal year										10-year total
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
2.3.2 Synthesis task												
2.3.2.1 Code for description of iodine behavior in HTGR containment. Recommend or develop code to describe iodine behavior in containment.	D											
Task group 2.3 total operating costs		30 (0.5)										30 (0.5)
Task area 2 total operating costs		330 (5.5)	580 (9.5)	450 (7.5)	420 (7.0)	300 (5)	240 (4)	180 (3)	180 (3)	180 (3)	180 (3)	3040 (50.5)
Equipment costs		50	275	100								425
Task area 2 total costs		380	855	550	420	300	240	180	180	180	180	3465

^aAdditional tasks may be assigned to this task area as information becomes available.

^bExpenditure and/or priority depends on the result of the need assessment study.

Table 1.4. Summary of identified safety work in task area 3^a

Task description	Task priority	Costs (\$1000) and manpower (man-years in parentheses) by fiscal year										10-year total
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
Task group 3.0 – Assessment of the adequacy of primary coolant technology as regards safety factor	A	55 (1)	60 (1)	60 (1)	60 (1)	30 (0.5)	30 (0.5)	30 (0.5)	30 (0.5)	30 (0.5)	30 (0.5)	415 (7)

^aAdditional tasks may be assigned to this task area as information becomes available.

Table 1.5. Summary of identified safety work in task area 4^a

Task description	Task priority	Costs (\$1000) and manpower (man-years in parentheses) by fiscal year										10-year total
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
Task group 4.0 Need assessment												
4.0.1.1 Need assessment and technology review	A	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	25 (0.5)	25 (0.5)	25 (0.5)	25 (0.5)	400 (8.0)
Task group 4.0 total		50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	25 (0.5)	25 (0.5)	25 (0.5)	25 (0.5)	400 (8.0)
Task group 4.1 Core seismic response												
4.1.1 Source tasks												
4.1.1.1 Seismic motions and effects	C		80 (1.0)	150 (2.0)	150 (2.0)	150 (2.0)	150 (2.0)	100 (1.3)	50 (0.7)			830 (11)
4.1.1.2 Core scale-model seismic tests	A	600 (8)	900 (12)	1200 (16)	1500 (20)	1000 (13)	700 (9)	300 (4)				6,200 (82)
4.1.1.3 Fuel-element impact tests	A	60 (1)	150 (2)	150 (2)	100 (1.5)							460 (6.5)
4.1.2 Synthesis tasks												
4.1.2.1 Analytical model of core seismic response	C	75 (1)	300 (4)	300 (4)	300 (4)	300 (4)	200 (2.5)	100 (1.5)				1,575 (21)
Task group 4.1 total		735 (10)	1430 (19)	1800 (24)	2050 (27.5)	1450 (19)	1050 (13.5)	500 (6.9)	50 (0.7)			9,065 (120.5)
Task group 4.2 Component seismic response		50 (0.7)	250 (3.3)	500 (6.7)	800 (10.7)	600 (8)	300 (4)					2,500 (33.4)
4.2.1 Source tasks												
4.2.1.1 Seismic-induced component motions and their effects	B											
4.2.1.2 Component seismic response tests	B											
4.2.2 Synthesis task												
4.2.2.1 Analytical models of component seismic response	B											
Task group 4.2 total		50 (0.7)	250 (3.3)	500 (6.7)	800 (10.7)	600 (8)	300 (4)					2,500 (33.4)
Task group 4.3 Soil-structure dynamic interaction		70 (1)	150 (2)	250 (3.3)	250 (3.3)	100 (1.4)						820 (11)

Table 1.5 (continued)

Task description	Task priority	Costs (\$1000) and manpower (man-years in parentheses) by fiscal year										10-year total
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
4.3.1 Source tasks												
4.3.1.1 Soil-structure interaction	C											
4.3.1.2 Soil-structure seismic model tests	C											
4.3.2 Synthesis task												
4.3.2.1 Analytical models of soil-structure seismic response	C											
Task group 4.3 total		70 (1)	150 (2)	250 (3.3)	250 (3.3)	100 (1.4)						820 (11)
Task group 4.4 PCR/V support			180 (2.4)	400 (5.3)	500 (6.7)	400 (5.3)	400 (5.3)	300 (4)	200 (2.7)	100 (1.3)		2,480 (33)
4.4.1 Source tasks												
4.4.1.1 PCR/V support-system concepts	C											
4.4.1.2 PCR/V support model tests	C											
4.4.2 Synthesis task												
4.4.2.1 Analytical models of PCR/V seismic response	B											
Task group 4.4 total			180 (2.4)	400 (5.3)	500 (6.7)	400 (5.3)	400 (5.3)	300 (4)	200 (2.7)	100 (1.3)		2,480 (33)
Equipment costs		810	1705	2125	1000	1000	700	400				7,740
Task area 4 total		1715 (12.7)	3765 (27.7)	5125 (40.3)	4650 (49.2)	3600 (34.7)	2500 (23.8)	1225 (11.3)	275 (3.9)	125 (1.8)	25 (0.5)	23,005 (205.9)

*Additional tasks may be assigned to this task area as information becomes available.

Table 1.6. Summary of identified safety work in task area 5^a

Task description	Task priority	Costs (\$1000) and manpower (man-years in parentheses) by fiscal year										10-year total
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
Task area 5 - Confinement components												
Task group 5.0 Need assessment for confinement components												
5.0.1.1 Initial technology assessment	A	50 (1.0)										50 (1.0)
5.0.1.2 Continuing need assessment study	A		50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	25 (0.5)	25 (0.5)	25 (0.5)	25 (0.5)	25 (0.5)	325 (6.5)
Task group 5.0 total		50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	25 (0.5)	25 (0.5)	25 (0.5)	25 (0.5)	25 (0.5)	375 (7.5)
Task group 5.1 Failure analysis												
5.1.2.1 Analytical PCRV failure studies	B	25 (0.5)	25 (0.5)	25 (0.5)	25 (0.5)	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	25 (0.5)	375 (7.5)
Task group 5.1 total		25 (0.5)	25 (0.5)	25 (0.5)	25 (0.5)	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	25 (0.5)	375 (7.5)
Task area 5 total		75 (1.5)	75 (1.5)	75 (1.5)	75 (1.5)	100 (2.0)	75 (1.5)	75 (1.5)	75 (1.5)	75 (1.5)	50 (1.0)	750 (15.0)

^aAdditional tasks may be assigned to this task area as information becomes available.

Table 1.7. Summary of identified safety work in task area 6^a

Task description	Task priority	Costs (\$1000) and man-power (man-years in parentheses) by fiscal year										10-year total
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
Task group 6.0 Need assessment for primary system materials technology		50 (1)	50 (1)	50 (1)	60 (1)	60 (1)	60 (1)	70 (1)	70 (1)	70 (1)	70 (1)	610 (10)
6.0.1.1 Need assessment for control materials. Safety-related assessments of control rod, reserve shutdown system, and burnable poison materials.	A											
6.0.1.2 Need assessment for graphite materials. As above for core support, reflector, and moderator graphites.	A											
6.0.1.3 Need assessment for metallic materials. As above for metallic materials of steam generator, thermal barriers, and circulator.	A											
Task group 6.0 total		50 (1)	50 (1)	50 (1)	60 (1)	60 (1)	60 (1)	70 (1)	70 (1)	70 (1)	70 (1)	610 (10)
Task group 6.1 Control materials												
6.1.1 Source tasks												
6.1.1.1 Reactions of coolant impurities with HTGR control materials. Assessment of potential for boron movement due to water vapor, etc. Effect of temperature, impurity concentrations, B ₄ C content and distribution, and irradiation.	A	160 (4.0)	180 (4.5)	180 (4.5)	20 (0.5)							540 (13.5)
6.1.1.2 Structural stability of HTGR control materials. Irradiation stability of full-size compacts and effects of thermal transients.	B	80 (2.0)	80 (2.0)	80 (2.0)	40 (1.0)							280 (7.0)
Task group 6.1 total		240 (6.0)	260 (6.5)	260 (6.5)	60 (1.5)							820 (20.5)
Task area 6 total		290 (7.0)	310 (7.5)	310 (7.5)	120 (2.5)	60 (1)	60 (1)	70 (1)	70 (1)	70 (1)	70 (1)	1430 (30.5)

^aAdditional tasks may be assigned to this task area as information becomes available.

Table 1.8. Summary of identified safety work in task area 7^a

Task description	Task priority	Costs (\$1000) and manpower (man-years in parentheses) by fiscal year										
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	10-year total
Task group 7.0 – Safety instrumentation assessment												
7.0.1.1 HTGR thermometry. A study to determine thermometry needs for safety and the consequences and probabilities of thermocouple failure adversely affecting the safety status of an HTGR. Includes identifying needs for on-line surveillance.	A	120 (2)	120 (2)									240 (4)
7.0.1.2 In-core flux monitoring. Assess HTGR safety needs for high-temperature in-core neutron detectors; this also involves assessing the state-of-the-art of existing detectors.	A	30 (0.5)										30 (0.5)
7.0.1.3 Surveillance and diagnostics by noise analysis and dynamic testing. Identify incipient failures that require early detection and process variables that might be monitored or dynamically perturbed for analysis related to adequate safety assessment.	C	60 (1)										60 (1)
7.0.1.4 Reactor coolant moisture monitoring. A detailed study of safety requirements.	A	60 (1)	60 (1)									120 (2)
7.0.1.5 Study to identify additional safety instrumentation needs. A comprehensive assessment of status and requirements of all instrumentation related to the safety of the plant.	A	180 (3)	120 (2)									300 (5)
Equipment costs												
Task area 7 total		450 (7.5)	300 (5)									750 (12.5)

^aAdditional tasks may be assigned to this task area as information becomes available.

problems of determining what accidents might take place and their severity. Analyses should consider major accident-initiating conditions and describe events in terms of plant transient effects, equipment response, accident termination effects, and their influence on the ultimate release of fission products. Tentative accident-sequence analyses should also be employed to help identify specific information needs. The values of the basic parameters required would be obtained from the technology studies to be carried out in the other six task areas, from design conditions, and from analyses of thermal, hydrodynamic, chemical, and mechanical conditions.

Specific types of accidents that should be considered initially are: interruption of core cooling, primary loop depressurization, and water ingress; other events may be added later. In all cases, the sequence of events has to be evaluated so as to determine fission-product and other radioactive material during the overall accident. In addition, a variety of conditions needs to be studied for any given specific type of accident. For example, if a depressurization accident takes place, there will be release of gas-borne radioactivity to the reactor containment volume; in addition, the "blow-down" action could lift radioactive particles from surfaces into the coolant stream; further, the transport of radioactivity into the containment has to be evaluated with regard to how this transport affects the escape of radioactivity from the containment. The evaluation is also complicated by the influence of reactor conditions on fuel failure and fission-product transport into the coolant circuit. Such interrelation of factors is present in many accident types; all of the interrelated factors need to be included in the evaluation of accident sequences and their consequences.

The analyses and evaluations to be carried out in task area 1 are identified as tasks, and these are listed and described briefly in Table 1.2; this table also gives the estimated 10-year effort associated with the different task groups and the task priorities.

As summarized in Table 1.2, tasks under task group 1.1 (component response) concern: (1) the collection and evaluation of information about component design, specifications, component response, and likely behavior of system components during accidents and (2) the performance and reliability of equipment and equipment systems similar to those used in large HTGRs. Tasks under task group 1.2 (analysis of event sequences) consider: (1) development of logic models that cover the range of credible accidents and their frequency and (2) potential accidents along with the modeling of such accidents, taking into account the probabilities that certain combinations of events, including common-mode failures, will occur. Because of the large number of possible interactions and feedback

loops, there are many cause-effect couplings which have to be considered in the mathematical modeling. Tasks under task group 1.3 (plant transient behavior modeling) provide: (1) calculational models for estimating the transient response of the primary and auxiliary heat removal systems under various coolant flow conditions, (2) models to describe stresses in components, (3) models to evaluate conditions associated with coolant depressurization and/or steam inleakage, and (4) models for estimating overall plant transient behavior under postulated accident conditions.

Task group 1.4 (safety analyses and evaluation) utilizes the information and models generated in all the task areas. The associated tasks perform parameter and accident analysis studies that assess HTGR accidents and their consequences; the spectrum of credible accidents should be considered.

1.3.2 Fission-product technology (Task area 2)

This task area provides for the assessment of additional needs with respect to fission-product technology, the development of information regarding the behavior of fuel and control materials at very high temperatures, and studies of the behavior of fission products in the primary loop and in the containment under postulated accident conditions. Brief descriptions of the tasks to be performed are given in Table 1.3, along with priorities and the estimated effort over a 10-year period.

Task group 2.0 is concerned with a detailed assessment of the status of fission-product technology as it applies to HTGR safety and the identification of needs for additional technical information. In addition goals should be identified for fission-product release rates and inventories.

The possible migration of fuel and control materials within the core under postulated conditions of severe overheating is examined in task group 2.1. This work will require a detailed assessment of available information before additional research can be defined.

Task groups 2.2 and 2.3 both deal with particular aspects of fission-product behavior. Group 2.2 examines fission-product lift-off from surfaces and transport within the primary loop. This work is directed primarily toward the behavior of aerosols and dust particles, which are expected to carry fission products, and the computer codes used to describe this behavior. Task group 2.3 is concerned with the transport, deposition, and release of fission products in and from the reactor containment. Of particular interest in this task group is the behavior of iodine.

1.3.3 Primary coolant technology (Task area 3)

This task area is concerned with the chemical and mass-transport reactions that may occur between impurities in the primary coolant and the materials in the primary circuit, including graphite, fuel, and structural metals. Expected impurities are water vapor and the products of steam-carbon reactions. The only safety work currently identified in this task area, as shown in Table 1.4, is a detailed assessment of the adequacy of existing technology and definition of possible needs for further data to accurately describe these reactions.

1.3.4 Seismic and vibration technology (Task area 4)

Task area 4 studies the responses of and effects on HTGR components resulting from seismic disturbances. The specific activities and the associated 10-year effort are summarized in Table 1.5.

Task group 4.0 provides for continuing review of the technology and the identification of additional needs. The remaining groups (4.1 through 4.4) address specific seismic effects, primarily through testing of models or components and developing mathematical models to describe responses. These four task groups are directed, respectively, toward core responses, component responses, soil-structure dynamic interactions, and PCRV-support responses.

1.3.5 Concrete reactor vessels and containment structures (Task area 5)

This task area examines the safety aspects of prestressed concrete reactor vessels and concrete containment structures. The activities are summarized in Table 1.6 along with priorities and estimated 10-year effort; these provide for an assessment of the technology and needs as well as of the analytical studies of PCRV failure modes. Additional work may be placed in this task area as a result of the need assessment studies.

1.3.6 Primary system materials technology (Task area 6)

This task area is concerned with the safety characteristics of the materials used in the primary systems of HTGRs. The activities are divided into three categories, corresponding to control components, graphite components, and metallic materials. The tasks are summarized in Table 1.7 along with estimates of the effort required over a 10-year period.

The need assessment task group (6.0) provides for continuing review of the technology in all three categories for the entire period. This is expected to require a relatively low-level effort. The only other activities currently identified in this area are concerned with the technology of control materials (task group 6.1). This work is aimed at developing improved understanding of the chemical stability of control materials, notably boronated graphite in the presence of possible HTGR coolant contaminants, and of their structural stability under irradiation and varying temperature conditions.

1.3.7 Safety instrumentation (Task area 7)

The subject of this task area is the safety instrumentation of HTGRs. Currently the only activities identified in this area are associated with assessing the need for additional instrument protection features to help assure or enhance the safety of such plants (task group 7.0). These activities are summarized in Table 1.8 which also includes estimates of the effort involved.

Several categories of instrumentation may be defined in which comprehensive need assessment requires highly specialized capabilities. Consequently, four tasks are identified to provide the desired review with respect to HTGR thermometry, in-core flux monitoring, surveillance and diagnostics by noise analysis and dynamic testing, and reactor-coolant moisture monitoring. A fifth task provides for overall review to identify any additional safety instrumentation needs in other categories.

1.4 SAFETY-RELATED PLANNING GUIDE (PART III)

The safety-related planning guide presents all of the safety and safety-related work in the technology task areas which was identified in the course of developing a safety planning guide. All of the activities to be discussed are related to the broad subject of safety, but, in many cases, the importance of the technology to other aspects of the design, construction, and normal operation of HTGR plants makes it likely that the information will be developed within other parts of the overall HTGR development program. It is presumed that information produced elsewhere will become available for use in safety studies.

In order to provide a balanced presentation of the interactions among all the various tasks, those activities that were discussed in the safety planning guide are also included here, with the exception of task area 1 (all the activities in task area 1 are

associated with the safety effort, so that task area is omitted from Part III except for its listing in Table 1.9). For clarification, the task groups or tasks that also appear in Part II are identified with asterisks (*) in the tables that summarize the safety-related activities. The estimated effort associated with the safety-related work is expressed only in relative terms; an overall summary of these activities by task area and group is presented in Table 1.9.

1.4.1 Fission-product technology (Task area 2)

The safety and safety-related activities identified in task area 2 deal with fission-product behavior as well as fuel-particle failure under both normal and abnormal operating conditions. Brief descriptions of the identified work are presented in Table 1.10.

As summarized under Part II, task groups 2.0 through 2.3 provide for assessment of the technology and identification of needs for additional information (task group 2.0); studies of the migration characteristics of fuel and control materials at high temperatures (task group 2.1); and studies of the liftoff and transport characteristics of fission products in the primary loop (task group 2.2) and in the containment (task group 2.3).

Other fission-product behavior processes in various parts of an HTGR system are the subject of task groups 2.4 through 2.6. Fission-product release from fuels (task group 2.4) addresses specifically the processes that lead to escape of materials from either intact or failed fuel particles. (Fuel-particle failure, as such, is the subject of task group 2.8.) Experimental data should be obtained from laboratory tests, in-pile experiments and operating HTGRs to provide a basis for developing mathematical models that describe the processes. Task group 2.5 examines the behavior of fission products in graphites; both fuel-matrix and moderator graphites are of interest. A variety of experiments should be performed to evaluate processes such as transport through graphite, by diffusion and transverse helium flow, and evaporation from graphite surfaces. Dependence of the processes on temperature and type and concentration of nuclides should also be determined. Experiments should range from laboratory tests to surveillance of operating reactors. In task group 2.6, which deals with the surface chemistry of fission-product behavior in the primary loop, the subjects of interest include primarily the adsorption and desorption of fission products on metallic, graphite, oxidized metal, and dust surfaces. As in all of these task groups, models to describe the behavior should be developed and checked against experimental and surveillance results.

Table 1.9. Summary of identified safety and safety-related work with estimated relative effort

Task group	Estimated relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task area 1 – Systems and safety analysis										
1.1 Component qualification	1	2	2	2	2	2	2	2	1	1
1.2 Analysis of event sequences	1	2	2	2	2	1	1	0.5	0.5	0.5
1.3 Modeling of plant transient behavior	5	7.5	8	8	8	7.5	6.5	5.5	4.5	4
1.4 Safety analysis and evaluation	1.5	5	6	5	5	5	5	3	3	2
Area total	8.5	16.5	18	17	17	15.5	14.5	11	9	7.5
Task area 2 – Fission product technology										
2.0 Need assessment	3	3	3	3	3	3	3	3	3	3
2.1 Fuel and control material		1	1	1						
2.2 Fission product behavior in the coolant circuit – engineering-scale dynamics	2	5.5	3.5	3	2	1				
2.3 Fission products in the containment	0.5									
2.4 Fission product release from fuels	10.5	13.5	12	10.5	9.5	8.5	8.5	8.5	7.5	
2.5 Fission product behavior in graphites	12.5	13	9	18	14	14	18	16	10	4
2.6 Fission products in the coolant circuit – surface chemistry	7	11.5	5.5	5	2	2	2.5	2.5	2.5	2.5
2.7 Tritium behavior in HTGRs	5	2.5	2.5	1	1		1	1	0.5	1
2.8 Fuel particle failure mechanisms	8.5	10.5	11	9.5	8	3.5	3.5	3.5	4.5	3.5
Area total	49	60.5	47.5	51	39.5	30	34.5	32.5	26	19.5
Task area 3 – Primary coolant technology										
3.0 Need assessment	1	1	1	1	0.5	0.5	0.5	0.5	0.5	0.5
3.1 Steam-core reactions and coolant composition	6.5	12.5	16	11.5	10.5	4.5	3.5	1.5	0.5	0.5
3.2 Primary coolant–component compatibility	0.7	4.1	7.1	4.5	3.2	2.2	0.1	0.1		
Area total	8.2	17.6	24.1	17.0	14.2	7.7	4.1	2.1	1	1
Task area 4 – Seismic and vibration technology										
4.0 Need assessment	1	1	1	1	1	1	0.5	0.5	0.5	0.5
4.1 Core seismic response	10	19	24	27.5	19	13.5	6.8	0.7		
4.2 Component seismic response	0.7	3.3	6.7	10.7	8	4				
4.3 Soil-structure dynamic interaction	1	2	3.3	3.3	1.4					
4.4 PCRV support		2.4	5.3	6.7	5.3	5.3	4	2.7	1.3	
4.5 Reactor internals vibration	1.3	1.3	2.7	2.7	2.7	1.3	1.3			
Area total	14	29	43	51.9	37.4	25.1	12.6	3.9	1.8	0.5

Table 1.9. (continued)

Task group	Estimated relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task area 5 – Confinement components										
5.0 Need assessment	1	1	1	1	1	0.5	0.5	0.5	0.5	0.5
5.1 Failure analysis	2.6	2.6	2.6	2	2	1.5	1	1	1	0.5
5.2 PCRV head failure studies		1.2	3.5	3.5	2.3		0.4	0.4	0.6	
5.3 Concrete properties		2.5	6	10	10	9	6	3.5	1.5	1
5.4 Prestressing components and materials		3	4	2	3	2	0.5	1.5	1.5	
5.5 Liners and penetrations		5	7	5.6	3.6	2.6	0.8	0.8		
5.6 Thermal barrier	1.5	7.5	8	2.5	0.5	0.5				
5.7 Instrumentation		2	5	5	5	4	2	1.5	1	
5.8 Model studies and PCRV surveillance		0.5	4.5	9	12	8.5	6.5	2	1	
5.9 Containment convection studies	1	4	4	3	2					
5.10 Code rule development				0.5	1	1	1.5	1.5	1.5	1.5
Area total	6.1	29.3	45.6	44.1	42.4	29.6	19.2	12.7	8.6	3.5
Task area 6 – Primary system materials technology										
6.0 Need assessment	1	1	1	1	1	1	1	1	1	1
6.1 Control materials	9.2	9.5	10.5	4	1.5	1.5	1.5	0.5	0.5	0.5
6.2 Graphite components	9	10	11	10	10	4	2	2	2	
6.3 Primary loop metallic components	13	23	29	28	23	18	9	5		
Area total	32.2	43.5	51.5	43	33.5	24.5	13.5	8.5	3.5	1.5
Task area 7 – Safety instrumentation										
7.0 Safety instrumentation assessment	7.5	5								
7.1 Development of failed fuel location	3	2	2	2	1	1	1			
7.2 HTGR thermometry development	2	2	2	4	4	2	2			
7.3 Development of in-core flux monitoring	2	3	2	1	1					
7.4 Development of surveillance and diagnostics by noise analysis and dynamic testing	2	3	3	3	2	1	1	1		
7.5 Development of reactor coolant moisture monitoring	2	3	1							
7.6 Development of data acquisition and processing systems	1	2	2	1						
7.7 Development of post-accident monitoring systems	1	2	2							
7.8 Development of plant controllability	1	2	2	2						
Area total	21.5	24	16	13	8	4	4	1		
Total	139.5	220.4	245.7	237	194	136.4	102.4	71.7	49.9	33.5

Table 1.10. Summary of identified safety and safety-related work in task area 2
Task groups or tasks marked with an asterisk (*) are included in the safety planning guide
(Part II of this document)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task area 2 - Fission product technology										
Task group 2.0 Need assessment for fission product technology										
2.1.1 Goals for fission product technology. Establishment of program goals for fission-product retention based on accident considerations. Critical review of technology. Updating as required. Coordination of reporting of results by element.	3	3	3	3	3	3	3	3	3	3
Task group 2.0 totals	3	3	3	3	3	3	3	3	3	3
Task group 2.1. Fuel and control material transport*										
2.1.1 Source tasks. Quantitative description of mechanisms by which control, fissile, and fertile materials might separate under accident conditions.										
2.1.1.1 Actinide adsorption and diffusion in graphites			1	1						
2.1.1.2 Control material adsorption and diffusion in graphite		1								
2.1.2 Synthesis tasks. Codes for analysis of movement of Th, B, and other control materials during accidents involving (a) very high temperatures and (b) steam and high temperatures.										
Task group 2.1 totals		1	1	1						
Task group 2.2 Fission product behavior in the coolant circuit - Engineering-scale dynamics*										
2.2.1 Source task.										
2.2.1.1 Fission product distribution and re-entrainment. Description of aerodynamics of aerosols in HTGR with emphasis on deposition and re-entrainment in realistically complex geometries. Studies of aerosol adhesion to appropriate substrates as functions of material temperature and time.	2	2	2	2.5	2	1				

Table 1.10 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
2.2.2 Synthesis task.										
2.2.2.1 Fission product distribution code for coolant circuit. Modification of PAD code to take account of role of dust in fission product distribution in primary circuit and formulation of dust re-entrainment model for accident analysis. See also 2.5.2.1.		1	0.5	0.5						
2.2.3 Validation tasks.										
2.2.3.1 Re-entrainment experiments. Experimental determination of the mobility of solids adhering to surfaces of the primary loop Peach Bottom HTGR. Correlation with need and expectation.		2.5	1							
2.2.3.2 Fission product and dust distributions in HTGRs. Measurements of distributions of deposited fission products on steam generators, ducting, and circulators in or from Fort St. Vrain or larger HTGRs. Development of appropriate remote measurement techniques (task 2.6.3.2 will supply this information for Peach Bottom).										
Task group 2.2 totals	2	5.5	3.5	3	2	1				
Task group 2.3 Fission products in the containment*										
2.3.1 Source task.										
2.3.1.1 Review of iodine behavior in containment. Review of work done under other programs and determine need for further refinement of description of iodine behavior in HTGR containment, including concrete. Consider other fission product behavior.	0.5									
2.3.2 Synthesis task.										
2.3.2.1 Code for description of iodine behavior in HTGR containment. Recommend or develop code to describe iodine behavior in containment.										
Task group 2.3 totals	0.5									

Table 1.10 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 2.4. Fission product release from fuels										
2.4.1 Source tasks. Description of mechanisms of fission product release, with intent of providing source terms for release codes. Determination of parameters for reference fuel.						2 ^a	2 ^a	2 ^a	2 ^a	2 ^a
2.4.1.1 Release of krypton and xenon	1	1								
2.4.1.2 Release of iodine, tellurium, and selenium		0.5	0.5							
2.4.1.3 Release of metallic fission products	7	7	6	5	4					
2.4.1.4 Release of actinides			1	1	1					
2.4.2 Synthesis tasks. Source terms for fission-product transport codes. Compare FIPER, FIPDIG, and other similar source terms. Formulate new models to match experimental data as necessary. Incorporate feedback from task 2.4.3.										
2.4.2.1 Kr, Xe release codes										
2.4.2.2 I, Te, Se release codes										
2.4.2.3 Metallic fission-product and actinide release codes	0.5	0.5								
2.4.3 Validation tasks. Fission-product distribution in HTGR fuel bodies. Fuel failure, fission-product release from kernels, coated particles, and matrices of Fort St. Vrain and large HTGR fuel. Comparison of results with code predictions. Minimal work on Peach Bottom fuel. Fission-product distributions in relevant experimental fuel bodies as available from other programs (work on fuel from large HTGR not costed).										
2.4.3.1 Kr, Xe release vs burnup in failed fuel		1	1	1	1	1	1	1	1	1
2.4.3.2 I, Te, Se release vs burnup in failed fuel		1	1	1	1	1	1	1		
2.4.3.3 Metallic fission-product release from high burnup fuels	2	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5
2.4.3.4 Actinide retention	(Effort included in task 2.4.3.3)									
Task group 2.4 totals	10.5	13.5	12	10.5	9.5	6.5	6.5	6.5	5.5	5.5

Table 1.10 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 2.5 Fission-product behavior in graphites										
2.5.1 Source tasks. Fission-product behavior in core graphites. Elucidation and description of mechanisms of transport of Cs, Ag, Sr, Ba, and Ce through structural graphites in systems with and without forced He flow. Behavior of fission products at very high temperatures. Vapor pressures of Cs, Ag, Sr, Ba, and Ce over pyrocarbon, matrix carbon, and structural graphites as functions of temperature, surface metal concentration and composition, and gas composition.						2 ^a	2 ^a	2 ^a	2 ^a	2 ^a
2.5.1.1 Iodine adsorption and diffusion in graphites	1	1								
2.5.1.1 Metallic fission-product adsorption and diffusion	5	5	4	4	2					
2.5.2 Synthesis tasks. Codes describing fission-product releases from HTGR elements. Continued development of codes of FIPER type in light of feedback from experimental work in task 2.5.3.										
2.5.2.1 Iodine transport codes	1.5	1	1							
2.5.2.2 Metallic fission-product transport codes										
2.5.3 Validation tasks. Postirradiation examination of fuel element and reflector graphite. Measurement of radial and axial distributions of I, Cs, Ag, Sr, Ce, Zr, and Ru isotopes in fuel elements and reflector graphite from Peach Bottom (driver), Fort St. Vrain and large HTGR. Comparisons of predicted behavior with observation (large HTGR not costed, Fort St. Vrain surveillance expected to cease in 1981. Contingency provided for continuation).										
2.5.3.1 Rare-gas transport in HTGR elements	0.5	0.5								
2.5.3.2 Iodine and tellurium transport in HTGR elements	0.5	0.5								
2.5.3.3 Metallic fission-product transport in Peach Bottom graphite	4	3	1	1						
2.5.3.4 Metallic fission-product transport in Fort St. Vrain graphite		2	3	13	12	12	16	14	8	2
2.5.3.5 Metallic fission-product transport in large HTGR graphite										
Task group 2.5 totals	12.5	13	9	18	14	14	18	16	10	4

Table 1.10 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 2.6. Fission products in the coolant circuit – surface chemistry										
2.6.1 Source tasks. Fission-product behavior on coolant-circuit surfaces. Surface equilibria involving steam generator metals, their oxides, and fission products in appropriate chemical forms. Effects of temperature, steam, and air. Partition of fission products between different "surface phases." Diffusion of fission products in metals.					1 ^a	1 ^a	1 ^a	1 ^a	1 ^a	1 ^a
2.6.1.1 Iodine and tellurium adsorption on metallic surfaces	2	1								
2.6.1.2 Metallic fission-product adsorption on metallic surfaces		1	1	2						
2.6.1.3 Fission-product diffusion in metals		1	1	1						
2.6.1.4 Aerosol studies		0.5	0.5							
2.6.2 Synthesis task. Descriptions of fission-product adsorption and evaporation in coolant circuit. Incorporate adsorption/desorption kinetics into PAD code where necessary. Adjust in light of observation.										
2.6.2.1 Fission-product distribution codes	0.5	0.5								
2.6.3 Validation tasks.										
2.6.3.1 Fission-product deposition in Peach Bottom coolant circuit during operation. Measurement of release from core into coolant, average deposition rate between gas samples, distribution of deposited activities. Characterization of dust; determination of fission products in dust; distribution of fission products between dust on surfaces, surface films, and metal substrates. Measurement of krypton and xenon contributions to daughter activities in circuit. Comparison of observation with model predictions and model adjustment as needed.	1.5									

Table 1.10 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
2.6.3.2 Fission-product distributions in Peach Bottom coolant circuit following operation. Determination of amounts and distributions of all γ emitting nuclides in primary circuit plus ^{129}I , ^{89}Sr , and ^{90}Sr as feasible. Characterization of dust and deposits in steam generator, circulator, ducting, and hot valve. Determination of distributions of activity between deposited dust, surface films, and base metal.	0.5	5	1.5	1						
2.6.3.3 Fission-product deposition in Fort St. Vrain coolant circuit. Measurement of releases from core into coolant, average deposition rate onto steam generator, and distribution of deposited activities as feasible. Characterization of collected dust. Measurement of krypton and xenon contributions to daughter activities in circuit. Comparison of observation with PAD code prediction.	2	1.5	1.5	1	1	1	1.5	0.5	0.5	0.5
2.6.3.4 Fission-product deposition in large HTGRs. Design and install coolant samplers up and downstream of more than one steam generator. Determination of amounts and distributions of all γ emitting nuclides in primary circuit plus ^{129}I , ^{89}Sr , and ^{90}Sr as feasible. Characterization of dust and deposits in steam generator, circulator, ducting, and hot valve. Determination of distributions of activity between deposited dust, surface films, and base metal.	0.5	1						1	1	1
Task group 2.6 totals	7	11.5	5.5	5	2	2	2.5	2.5	2.5	2.5
Task group 2.7 Tritium behavior in HTGRs										
2.7.1 Source tasks.										
2.7.1.1 Tritium adsorption on graphite. Measurement of ^3H adsorption and/or exchange with ^3H on graphite.			1	1						
2.7.1.2 Tritium permeation through metals. Measurement of ^3H permeation through steam-generator materials	0.5	0.5								
2.7.1.3 Tritium release from fuels, recoil sites in graphite and boron carbide. Determination of source functions describing release of ^3H from bare kernels, coated particles, and recoil sites in graphite and boron carbide for use in TRITGO code.	1	1								

Table 1.10 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
2.7.2 Synthesis task.										
2.7.2.1 Codes describing tritium behavior in HTGR. Continued development of code describing ^3H formation, distribution, and release in HTGR.	0.5									
2.7.3 Validation tasks.										
2.7.3.1 Tritium in Peach Bottom coolants. Determination of steady state concentrations of ^3H and $^3\text{H}/^1\text{H}$ ratio in primary and secondary coolants. Determination of sizes of hydrogen sinks and effects of $^3\text{H}/^1\text{H}$ ratio on steam generator permeability. Measurement of Peach Bottom steam generator average permeability. Measurement of ^3H concentrations in waste streams, purge stream, and inlet to purification system.	1.5									
2.7.3.2 Tritium in Peach Bottom fuel, graphite, and control rods. Measurements of ^3H concentrations in fuel, fuel element, and moveable reflector graphite and control rod materials.	0.5	0.5								
2.7.3.3 Tritium in Fort St. Vrain coolant. Determination of ^3H concentrations in primary and secondary coolants, in all waste streams, and in entry and exit of purification system.	1	0.5	0.5							
2.7.3.4 Tritium in Fort St. Vrain fuel, graphite, and control materials. Measurement of ^3H concentration relative to initial ^{235}U , ^6Li , and ^{10}B concentrations in fuel, fuel element, and moveable reflector graphite and control rod materials.			1		1		1			
2.7.3.5 Tritium in large HTGR coolant. Determination of ^3H concentrations in primary and secondary coolants, in all waste streams, and in entry and exit of purification system.								1	0.5	0.5
2.7.3.6 Tritium in large HTGR fuel, graphite, and control materials. Measurement of ^3H concentration relative to initial ^{235}U , ^6Li , and ^{10}B concentrations in fuel, fuel element, and moveable reflector graphite and control rod materials.										0.5

Table 1.10 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 2.7 totals	5	2.5	2.5	1	1		1	1	0.5	1
Task group 2.8 Fuel particle failure mechanisms						2.5 ^a	2.5 ^a	2.5 ^a	2.5 ^a	2.5 ^a
2.8.1 Source tasks.										
2.8.1.1 Mass transport in temperature gradients. Identification and description of carbon, silicon, and carbide coated particles. Determination of P_{CO} and $P_{(Kr+Xe)}$. Determination of parameters needed to calculate core thermal limits.	2	2	2	1	1					
2.8.1.2 High temperature isothermal failure mechanisms. Determination of conditions required to fail various types of coated fuel particles by high-temperature annealing following irradiation.	2	2	2	1	1					
2.8.1.3 Chemical compatibility. Determination of factors governing compatibility of HTGR fuels with coating materials after extensive burnup. Derivation of limiting conditions.	3	4	2	2	1					
2.8.2 Synthesis task.										
2.8.2.1 Formulation of fuel-failure criteria. Formulation of mathematical descriptions of chemical failure processes in oxide and carbide fuels for calculation of core working limits. Incorporate CO production terms into STRESS II and other similar codes. Update as needed.	0.5	0.5	0.5	0.5						
2.8.3 Validation tasks.										
2.8.3.1 Postirradiation examination of fuels from capsules. Measurements of kernel migration rates, internal gas pressures, and compositions and observation of kernel-coating compatibility in fuels from capsules.	0.5	0.5	0.5	0.5	0.5					
2.8.3.2 Special irradiation capsule for mass transport studies. Design, construction, operation, and examination of capsule for "amoeba effect" mechanism studies.	0.5	0.5	2	2	2					

Table 1.10 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
2.8.3.3 Postirradiation examination of Fort St. Vrain fuel Determination of kernel migration coefficients, fraction of failed SiC layers, and composition and pressure of gas in intact particles. Observation of kernel-fuel compatibility. Scale of examination to be re-assessed in light of realistic objectives. Provision of data for revision of core operating limits and fuel specifications.		1	2	2.5	2.5	1	1	1	1	
2.8.3.4 Postirradiation examination of large HTGR fuel. Determination of kernel migration coefficients, fraction of exposed fuel, fraction of failed SiC layers, and composition and pressure of gas in intact particles. Observation of kernel-fuel compatibility. Scale of examination to be re-assessed in light of realistic objective. Provision of data for revision of core operating limits and fuel specifications.									1	1
Task group 2.8 totals	8.5	10.5	11	9.5	8	3.5	3.5	3.5	4.5	3.5
Task area 2 totals	49	60.5	47.5	51	39.5	30	34.5	32.5	26	19.5

^aMinimal effort to maintain nominal capability.

Because of its unique behavior characteristics, such as diffusion through hot metal walls, tritium (^3H) is treated separately (task group 2.7). While some ^3H is formed by ternary fission, the principal source in HTGRs is neutron reaction in boron. The distribution of ^3H throughout HTGR systems should be studied to support the development of improved models to describe its behavior.

The tasks under task group 2.8 concern fuel-failure mechanisms; these studies are closely related to those on fission-product behavior and distribution, since fuel failure leads to fission-product release from coated particles. The specific tasks concern: (1) evaluation of basic compatibilities and transport processes in HTGR fuels at very high burnup and at high temperatures, (2) prediction of fuel behavior under various conditions, and (3) determination of those conditions that lead to fuel failure so as to estimate the failures which might occur under accident conditions.

1.4.2 Primary coolant technology (Task area 3)

This task area studies the chemical and mass transport reactions that may occur between the impurities in the primary coolant and primary circuit materials such as core graphite, fuel, and structural metals. Table 1.11 gives a brief description of the tasks that are identified.

Task group 3.0 provides for a detailed and continuing assessment of the technology as it relates to safety and the definition of possible needs for further data to describe the relevant processes more accurately.

Under task group 3.1 (steam-core reactions and coolant composition), the tasks determine: (1) steam-graphite-fuel reactions and associated reaction rates under various conditions, (2) the coolant composition as influenced by steam-graphite reactions and reactor conditions, (3) the influence of radiolytic reactions on coolant composition, (4) the influence of coolant impurities on fuel kernel hydrolysis and associated reaction rates (considering failed coatings on fuel), and (5) graphite strength as a function of exposure to oxidation conditions. Also involved in these tasks are: development of calculational programs that will describe steam-core reactions, estimate coolant compositions, and estimate associated fuel and graphite behavior; evaluation of gas compositions which might develop within the secondary containment vessel and their flammability; surveillance of coolant compositions occurring in operating HTGRs; and post-power-generation testing and examination in the Peach Bottom HTGR.

Table 1.11. Summary of identified safety and safety-related work in task area 3

Task groups or tasks marked with an asterisk (*) are included in the safety planning guide (Part II of this document).

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 3.0 Need assessment*	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5	0.5
3.0.1 Source task										
3.0.1.1 Assessment of the adequacy of primary coolant technology as regards reactor safety.										
Task group 3.0 totals	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5	0.5
Task group 3.1 Steam-core reactions and coolant composition										
3.1.1 Source tasks										
3.1.1.1 Kinetics of graphite oxidation.										
3.1.1.1a Estimate of graphite variability. Determination of surface areas, permeability, interdiffusion coefficient values, and porosities of reference graphites.	1.0	1.0	1.0	1.0						
3.1.1.1b Effects of elevated pressure. Determination of effect of total pressure and of high steam partial pressure on rate of reaction.	1.0	1.0	0.5	0.5						
3.1.1.1c Identification of reaction regimes. Determination of pressure and temperature ranges over which chemical reaction control, in-pore diffusion control, and gas-phase diffusion control are operative.	(included in tasks 3.1.1.1b and 3.1.1.1d)									
3.1.1.1d Determination of rate constants. Includes burnoff and catalysis effects of all pertinent reference graphites.	2.0	3.0	3.0	3.0	3.0					
3.1.1.1e Variation of geometric parameters with extent of reaction. Determination of some parameters as in task 3.1.1.1a but as a function of burnoff.	(included in tasks 3.1.1.1d and 3.1.1.3)									
3.1.1.1f Transverse flow effects. Determination of steam graphite corrosion characteristics under conditions that simulate a single coolant channel and surrounding fuel compartments.		2.0	6.0 ^{b,c}	3.0	0.5 ^d	0.5				

Table 1.11 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
3.1.1.1g Catalysis studies. Determination of catalytic effect of impurities and fission products on graphite corrosion by steam.	(included in task 3.1.1.1d)									
3.1.1.1h Measurement of yields of radiolytic reactions. Determination of appropriate G values as required.	(additional work conditionally unwarranted)									
3.1.1.1i Air-graphite reaction measurements. Determination of rate constants for graphites of interest.	(additional work conditionally unwarranted)									
Task 3.1.1.1 total	4.0	7.0	10.5	7.5	2.5	0.5				
3.1.1.2 Reactions involving fuel kernels. Determination of rate of carbide fuel hydrolysis and passivation mechanisms.	0.5	1.0	1.0	1.0						
3.1.1.3 Structural strength of graphite. Determination of graphite strength as a function of burnoff.		1.0	1.0	1.0						
3.1.1.4 Reactions involving fission products. Determination of likelihood of formation of volatile forms by interactions with steam or carbon monoxide.		1.0	1.0							
3.1.1.5 Flammability studies. Determination of flammability limits of H ₂ -CO-He-H ₂ O-air mixtures.	(additional work conditionally unwarranted)									
Task 3.1.1 total	4.5	10.0	13.5	9.5	2.5	0.5				
3.1.2 Synthesis task										
3.1.2.1 Modeling studies of effects of steam ingress. Comparison of current codes, use of GOP in Fort St. Vrain startup, perform parameter sensitivity analyses.	2.0	2.5	2.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5
3.1.3 Validation task										
3.1.3.1 Validation of steam ingress computer code. Intercomparison of code results with large-scale and inpile loop experiments.			0.5	1.0	7.0	3.0	3.0	1.0		
Task group 3.1 total	6.5	12.5	16.0	11.5	10.5	4.5	3.5	1.5	0.5	0.5

Table 1.11 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 3.2 Primary coolant-component compatibility										
3.2.1 Source task										
3.2.1.1 Reactions of coolant impurities with HTGR structural materials. Determination of rates of oxidation, carburization, etc., of primary circuit materials in impure helium. Study of microstructural stability and surface films.	0.5	3.0	3.0	3.0	3.0	2.0				
3.2.3 Validation tasks										
3.2.3.1 Examination of components from Peach Bottom HTGR. Examination of components from primary circuit for evidence of oxidation, etc.	0.2	1.0	4.0	1.3						
3.2.3.2 Examination of specimens from Fort St. Vrain HTGR. Examination of metallic components (e.g., control rods).		0.1	0.1	0.2	0.2	0.2	0.1	0.1		
Task group 3.2 total	0.7	4.1	7.1	4.5	3.2	2.2	0.1	0.1		
Task area 3 total	8.2	17.6	24.1	17.0	14.2	7.7	4.1	2.1	1.0	1.0

^aIncludes construction of a large-scale loop that will also be employed in validation experiments.

^bNecessity of experimental loops and/or extent of experimental work depends upon the nature and the availability of Pegase loop data. Estimates assume that the Pegase data are unavailable or unacceptable for AEC use.

^cIncludes construction of an in-pile loop.

The tasks under task group 3.2 (primary coolant-component compatibility) involve: determination of the influence of coolant impurities on HTGR structural-materials properties and on corrosion behavior; acquisition of data on the oxidation and carbonization of Incoloy 800, other HTGR structural metals and their weldments when exposed to primary coolant conditions and the examination of surveillance specimens and components in the Peach Bottom and other HTGRs. The metals of interest are those associated with the steam generator, the circuit liner, and other components exposed to the coolant.

1.4.3 Seismic and vibration technology (Task area 4)

The effort in this task area is directed toward a more complete understanding of all aspects of the physical disturbance of HTGR structural and operational components. Such disturbances may result from external causes, e.g., earthquakes, or from internal processes, e.g., gas flow. Brief descriptions of the identified tasks are given in Table 1.12.

Task group 4.0 involves an independent review and assessment of the technology in this task area and the recommendation of possible further work that could lead to more clearly defined margins of safety.

Task groups 4.1 through 4.4 involve evaluation of seismic events and determination of the influence such events have on HTGR structural behavior. Task group 4.1 (core seismic response) concerns: experimental and theoretical studies of core response (including response of fuel blocks, support floor, and graphite support posts) to mechanical shaking motions; evaluation of seismic motions to be expected; and analysis of core response and integrity under such conditions. Tasks under group 4.2 (component seismic response) are to perform analytical and experimental studies of component motions induced by assumed seismic motions and to evaluate component integrity under postulated seismic conditions. The work in task group 4.3 (soil-structure dynamic interaction) examines soil-structure dynamic motions which result from seismic events; both analytical and experimental studies are involved. Tasks under group 4.4 (PCRV support) are to perform theoretical and experimental studies concerning the response of the PCRV support system to mechanical motions and to evaluate the integrity of the support structure when exposed to postulated seismic disturbances.

Tasks in group 4.5 (reactor internals vibration) determine component responses to possible flow-induced or circulator-induced

Table 1.12. Summary of identified safety and safety-related work in task area 4

Task groups or tasks marked with an asterisk (*) are included
in the safety planning guide (Part II of this document).

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 4.0 Need assessment*	1.0	1.0	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5
4.0.1.1 Need assessment and technology review										
Task group 4.0 totals	1.0	1.0	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5
Task group 4.1 Core seismic response*										
4.1.1 Source tasks										
4.1.1.1 Seismic motions and effects		1.0	2.0	2.0	2.0	2.0	1.3	0.7		
4.1.1.2 Core scale model seismic tests	8	12	16	20	13	9	4			
4.1.1.3 Fuel element impact tests	1	2	2	1.5						
4.1.2 Synthesis tasks										
4.1.2.1 Analytical model of core seismic response	1	4	4	4	4	2.5	1.5			
Task group 4.1 total	10	19	24	27.5	19	13.5	6.8	0.7		
Task group 4.2 Component seismic response*	0.7	3.3	6.7	10.7	8	4				
4.2.1 Source tasks										
4.2.1.1 Seismic induced component motions and their effects										
4.2.1.2 Component seismic response tests										
4.2.2 Synthesis task										
4.2.2.1 Analytical models of component seismic response										
Task group 4.2 total	0.7	3.3	6.7	10.7	8	4				
Task group 4.3 Soil-structure dynamic interaction*	1	2	3.3	3.3	1.4					
4.3.1 Source tasks										
4.3.1.1 Soil-structure interaction										
4.3.1.2 Soil-structure seismic model tests										
4.3.2 Synthesis task										
4.3.2.1 Analytical models of soil-structure seismic response										
Task group 4.3 total	1	2	3.3	3.3	1.4					
Task group 4.4 PCRV support*		2.4	5.3	6.7	5.3	5.3	4	2.7	1.3	
4.4.1 Source tasks										
4.4.1.1 PCRV support system concepts										
4.4.1.2 PCRV support model tests										

Table 1.12 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
4.4.2 Synthesis task										
4.4.2.1 Analytical models of PCRV seismic response										
Task group 4.4 total		2.4	5.3	6.7	5.3	5.3	4	2.7	1.3	
Task group 4.5 Reactor internals vibration	1.3	1.3	2.7	2.7	2.7	1.3	1.3			
4.5.1 Source tasks										
4.5.1.1 Vibration excitation and component response										
4.5.1.2 Component vibration response tests										
4.5.2 Synthesis task										
4.5.2.1 Analytical methods for predicting component vibration										
Task group 4.5 total	1.3	1.3	2.7	2.7	2.7	1.3	1.3			
Task area 4 total	14	29	43	51.9	37.4	25.1	12.6	3.9	1.8	

vibrations and estimate the performance of equipment when exposed to various vibrations. Structures involved in these studies include thermal-barrier cover plates, steam generator tubing, seals, and components of the core.

1.4.4 Concrete reactor vessels and containment structures (Task area 5)

The work in this area examines all aspects of the safety and safety-related behavior of PCRVs, including their component parts, and of concrete reactor-containment structures. Eleven task groups are identified that would ultimately permit accurate evaluation of the long-term safety of HTGR pressure vessel and containment systems. The tasks to be performed are listed in Table 1.13, which also briefly describes the work.

The initial effort in this task area (task group 5.0) calls for a comprehensive assessment of the technology developments in recent years to permit more accurate definition of the needs and goals in the remaining task groups. The effort summarized below may be subject to some modification as a consequence of this review. This task group also provides for continuing assessment of the technology and identification of needs.

Task group 5.1 (failure analysis) has as its goal the development of reliable capability for predicting the modes of failure of PCRVs. The work involves failure testing of PCRV models and refinement of the analytical techniques for describing the kinds of failures that are observed. Task group 5.2 (design and analysis methods) is similar to 5.1 but deals with the description of long-term safety of PCRVs under normal, as well as off-normal, conditions.

Both of the above task groups will draw extensively on the information developed in task group 5.3 (concrete properties). The tasks in group 5.3 would generate data on the strength, long-term creep, and creep recovery of ordinary concretes as well as any special formulations that may be developed for PCRV applications. Tests involve thermal and stress cycling as well as failure testing, with temperature (to 600°F) and moisture content as parameters. Multi-axial tests should be included to study shear effects. The work in this task group is applicable to concrete containment structures and PCRVs.

Task group 5.4 would develop information about the properties and failure modes of prestressing components; both axial tendons and circumferential wires and strips are considered. Of particular interest are effects of environmental factors (moisture, temperature,

Table 1.13. Summary of identified safety and safety-related work in task area 5

Task groups or tasks marked with an asterisk (*) are included
in the safety planning guide (Part II of this document).

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task area 5 Confinement components										
Task group 5.0 Need assessment for confinement components*										
5.0.1.1 Initial technology assessment	1.0									
5.0.1.2 Continuing need assessment study		1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5
Task group 5.0 total	1.0	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5
Task group 5.1 Failure analysis										
5.1.2.1 Analytical PCRV failure studies*	0.5	0.5	0.5	0.5	1.0	1.0	1.0	1.0	1.0	0.5
5.1.2.2 Failure analysis methods development	0.5	0.5	0.5	0.5						
5.1.2.3 Three-dimensional finite element analysis improvement and evaluation	1.6	1.6	1.6	1.0	1.0	0.5				
Task group 5.1 total	2.6	2.6	2.6	2.0	2.0	1.5	1.0	1.0	1.0	0.5
Task group 5.2 PCRV head failure studies										
5.2.2.1 Concrete failure criterion development		1.0	1.0	1.0						
5.2.2.2 Simplified design methods development							0.4	0.4	0.6	
5.2.3.1 PCRV model failure tests		0.2	2.5	2.5	2.3					
Task group 5.2 total		1.2	3.5	3.5	2.3		0.4	0.4	0.6	
Task group 5.3 Concrete properties										
5.3.1.1 High-temperature strength		1.0	2.0	3.0	3.0	3.0	2.0	1.0		
5.3.1.2 Long-term creep and creep recovery		1.0	2.0	3.0	3.0	3.0	3.0	2.0	1.0	0.5
5.3.1.3 Reinforced concrete shear strength			1.0	2.0	2.0	2.0	0.5			
5.3.2.1 New concrete materials development		0.5	1.0	2.0	2.0	1.0	0.5	0.5	0.5	0.5
Task group 5.3 total		2.5	6.0	10.0	10.0	9.0	6.0	3.5	1.5	1.0
Task group 5.4 Prestressing components and materials										
5.4.1.1 Tendon corrosion and failure mechanisms		1.0	2.0	2.0	2.0	2.0	0.5	0.5	0.5	
5.4.1.2 Strip-wound prestressing systems		2.0	2.0							
5.4.2.1 Tendon installation criteria development					1.0			1.0	1.0	
Task group 5.4 total		3.0	4.0	2.0	3.0	2.0	0.5	1.5	1.5	

Table 1.13 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 5.5 Liners and penetrations										
5.5.1.1 Liner and penetration materials and fabrication techniques		2.0	2.0	1.0						
5.5.1.2 Liner and penetration failure modes and criteria		1.0	3.0	3.0	2.0	1.0				
5.5.2.1 Liner and penetration design analysis methods		2.0	2.0	1.6	1.6	1.6	0.8	0.8		
Task group 5.5 total		5.0	7.0	5.6	3.6	2.6	0.8	0.8		
Task group 5.6 Thermal barrier										
5.6.1.1 Thermal barrier testing	1.5	5.0	4.0	1.0						
5.6.2.1 Thermal barrier materials development		2.5	4.0	1.5	0.5	0.5				
Task group 5.6 total	1.5	7.5	8.0	2.5	0.5	0.5				
Task group 5.7 Instrumentation										
5.7.2.1 Concrete instrumentation development		1.0	2.0	2.0	2.0	2.0	1.0	0.5	0.5	
5.7.2.2 PCRV component instrumentation development		1.0	3.0	3.0	3.0	2.0	1.0	1.0	0.5	
Task group 5.7 total		2.0	5.0	5.0	5.0	4.0	2.0	1.5	1.0	
Task group 5.8 Model studies and PCRV surveillance										
5.8.1.1 Simple concrete structure tests		0.5	4.0	2.0	2.0	0.5				
5.8.1.2 Complex concrete structure tests			0.5	6.0	4.0	2.0	0.5			
5.8.2.1 PCRV scale-model tests				1.0	6.0	6.0	6.0	2.0	1.0	
5.8.3.1 PCRV surveillance										
Task group 5.8 total		0.5	4.5	9.0	12.0	8.5	6.5	2.0	1.0	
Task group 5.9 Containment convection studies										
5.9.2.1 Containment forced-convection study	1.0	3.0	3.0	1.0						
5.9.2.2 PCRV natural convection study		1.0	1.0	2.0	2.0					
Task group 5.9 total	1.0	4.0	4.0	3.0	2.0					
Task group 5.10 Code rule development										
5.10.1.1 Data collection and interpretation				0.5	0.5	1.0	1.0	1.0	1.0	1.0
5.10.2.1 Data evaluation and code rule development					0.5		0.5	0.5	0.5	0.5
Task group 5.10 total				0.5	1.0	1.0	1.5	1.5	1.5	1.5
Task area 5 total	6.1	29.3	45.6	44.1	42.4	29.6	19.2	12.7	8.6	3.5

radiation levels) on the prestressing materials themselves and on the grouts and protective coatings in contact with the prestressing devices.

The PCRV liner system and penetrations are studied in task group 5.5; the liner system includes the anchors and cooling coils as well as the liner membrane. Liner failure modes (as a consequence of environmental effects, interactions with the PCRV, and stress conditions) should be determined as a basis for assessing reliability and safety margins. Similar investigations should be applied to penetrations.

The safety-related performance of the PCRV thermal barrier is the subject of task group 5.6. Tests should be performed to define the adequacy of candidate materials (fibrous and solid) for service and resistance to thermal-cycling effects at temperatures associated with accident conditions.

The special instrumentation needs for monitoring and testing PCRVs and containment structures are addressed in task group 5.7. Specific development efforts should be directed toward devices to monitor concrete moisture content, to detect local concrete cracking and prestress failures, and to detect and locate cooling-coil leaks or blockages and separations of the liner from the PCRV.

Task group 5.8 is concerned with model studies and surveillance of actual PCRVs. Measurements of stress, strain or creep, and water content should be made in various models and PCRVs (as opportunities occur) to provide guidance for further basic studies and to validate analytical techniques that are developed.

Task group 5.9 would evaluate the effects and consequences of heat transfer to the internal surfaces of PCRVs and containment structures due to thermal convection effects under accident conditions such as loss of coolant flow (for PCRVs) or depressurization (for containments). The work includes accurate determination of the rates of heat transfer and the thermal stresses and thermal shocks that result therefrom.

Task group 5.10 provides for reducing the technology developed in this area to actual practice through contributions to the evolution of code rules which govern the design of PCRVs and reactor containment structures.

1.4.5 Primary system materials technology (Task area 6)

This task area is concerned with the safety and safety-related aspects of the materials used in the primary systems of HTGRs. Brief descriptions of the tasks are given in Tables 1.14.

The work in task group 6.0 is associated with assessing the technology and identifying additional safety needs with respect to the three classes of primary-system materials: control materials, graphite components, and metallic materials.

Task group 6.1 (control materials) is concerned with the behavior of neutron-absorbing materials in HTGR core environments. The reference material, B_4C , should be examined to determine potential limitations due to chemical, nuclear, thermal, and aging effects particularly under accident conditions.

Under task group 6.2 (graphite components), the specific tasks involve: graphite creep determinations as a function of temperature and of fast neutron fluence; obtaining data on the dimensional behavior, heat transfer behavior, and mechanical properties of graphite under HTGR conditions; development and evaluation of analytical models for estimating graphite behavior; and surveillance of graphite behavior in operating HTGRs.

Task group 6.3 deals with metallic materials to provide: evaluation of mechanical property data presently available; additional mechanical property data as needed and failure criteria for HTGR structural metals; analytical methods for estimating mechanical performance of equipment; design analysis of stress conditions in specific equipment to estimate where failures might occur; experimental testing of materials and equipment to determine response and failure characteristics; examination of components from the Peach Bottom HTGR to determine the influence of environment on mechanical properties; and corrosion testing of steam generator materials to determine material corrosion behavior and to evaluate failures that might be expected as a result of corrosion.

1.4.6 Safety instrumentation (Task area 7)

The subject of this task area is the assessment, development, and testing of instruments and instrument systems required to assure the safety of HTGR plants. The initial work (task group 7.0) deals with technology assessment and need identification with respect to four specific classes of instrumentation and safety instrumentation as a whole. Eight additional task groups are defined to develop technology in particular categories of safety and safety-related instrumentation. These activities are summarized in Table 1.15.

Table 1.14. Summary of identified safety and safety-related work in task area 6
Task groups or tasks marked with an asterisk (*) are included in the safety planning guide (Part II of this document).

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 6.0 – Need assessment*										
6.0.1 Source tasks	1	1	1	1	1	1	1	1	1	1
6.0.1.1 Control materials safety technology needs										
6.0.1.2 Graphite materials safety technology needs										
6.0.1.3 Metallic materials safety technology needs										
Task group 6.0 totals	1	1	1	1	1	1	1	1	1	1
Task group 6.1 – Control materials										
6.1.1 Source tasks										
6.1.1.1 Reactions of coolant impurities with HTGR control materials. Assessment of potential for boron movement due to water vapor, etc. Effect of temperature, impurity concentrations, B ₄ C content and distribution, and irradiation.	4	4.5	4.5	0.5						
6.1.1.2 Structural stability of HTGR control materials. Irradiation stability of full-sized compacts and effects of thermal transients.	2	2	2	1						
6.1.1.3 Integrity of control-rod cladding materials. Determination of compatibility effects on mechanical properties of cladding and thermal aging on shock absorber capabilities.	2	2	2	1						
6.1.2 Synthesis task										
6.1.2.1 Control materials modeling. Development of model for control material performance including structural stability and boron movement.	1	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
6.1.3 Validation task										
6.1.3.1 Fort St. Vrain control-rod materials surveillance tests. Examination and testing of control-rod absorbers and cladding, reserve shutdown system, and burnable poison.	0.2	0.5	1.5	1	1	1	1			
Task group 6.1 totals	9.2	9.5	10.5	4	1.5	1.5	1.5	0.5	0.5	0.5

Table 1.14 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 6.2 – Graphite components										
6.2.1 Source tasks										
6.2.1.1 Effect of temperature, fluence, and materials properties on creep coefficients.	4	2	2	2						
6.2.1.2 Determination of physical properties above the normal operating temperatures of an HTGR.	4	3	3							
6.2.1.3 Effect of thermal history on irradiation behavior.		3	2	2	2					
6.2.1.4 Annealing effects on graphite properties.	1	2	2	2						
6.2.1.5 Effect of cyclic loading on the stress-strain curve.			1	1	1					
6.2.1.6 Determination of thermomechanical properties under reactor operating conditions.				2	2	1				
6.2.2 Synthesis task										
6.2.2.1 Development of time-dependent constitutive equations.	1	1	1	1	1	1				
6.2.3 Validation task										
6.2.3.1 Reactor surveillance.					2	2	2	2	2	
Task group 6.2 totals	9	10	11	10	10	4	2	2	2	
Task group 6.3 – Primary loop metallic components										
6.3.1 Source tasks										
6.3.1.1 Technology status report. Evaluation of the status of technology of HTGR metallic structural materials.	10	15	20	20	15	10	5	2		
6.3.1.2 Mechanical properties and failure criteria for HTGR structural materials. Mechanical property information needed to develop safety evaluations for HTGR primary circuit structural materials and weldments will be obtained.										
6.3.2 Synthesis task										
6.3.2.1 Design analysis methods for HTGR safety evaluation. Formulation of models of material behavior to permit safety analyses under normal and accident conditions.	Effort in 6.3.1.1									

Table 1.14 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
6.3.3 Validation tasks										
6.3.3.1 Validation of design analysis methods (simple geometries). Simple structural tests to verify the capability of analysis methods for material behavior.	Effort in 6.3.1.1									
6.3.3.2 Design analysis and failure-mode validation (complex tests). Large-scale tests to examine the response of HTGR primary circuit components to service and accident conditions.				2	6	8	4	3		
6.3.3.3 Examination of components from Peach Bottom HTGR. Removal of primary circuit components from Peach Bottom HTGR. Conduct of complete examination including mechanical property determinations.	1	5	5	2						
6.3.3.4 Validation of alternative steam generator materials. Demonstration of the feasibility of employing high alloy materials in the low-temperature regions of the HTGR steam generator.	2	3	4	4	2					
Task group 6.3 totals	13	23	29	28	23	18	9	5		
Task area 6 totals	32.2	43.5	51.5	43	35.5	24.5	13.5	8.5	3.5	1.5

Table 1.15. Summary of identified safety and safety-related work in task area 7

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 7.0 – Safety instrumentation assessment										
7.0.1.1 HTGR thermometry. A study to determine thermometry needs for safety and the consequences and probabilities of thermocouple failure adversely affecting the safety status of an HTGR. Includes identifying needs for on-line surveillance.	2	2								
7.0.1.2 In-core flux monitoring. Assess HTGR safety needs for high-temperature in-core neutron detectors; this also involves assessing the state of the art of existing detectors.	0.5									
7.0.1.3 Surveillance and diagnostics by noise analysis and dynamic testing. Identify incipient failures that require early detection and process variables that might be monitored or dynamically perturbed for analyses related to adequate safety assessment.	1									
7.0.1.4 Reactor coolant moisture monitoring. A detailed study of safety requirements.	1	1								
7.0.1.5 Study to identify additional safety instrumentation needs. A comprehensive assessment of status and requirements of all instrumentation related to the safety of the plant.	3	2								
Task group 7.0 total	7.5	5								
Task group 7.1 – Development of failed-fuel location										
7.1.1.1 FAT-P system development. Continue development of system based on flux and temperature peaking.	3	2	2	2	1	1	1			
7.1.1.2 Sniffle system development. Continue study of gas sampling scheme to detect activity from specific regions.										
7.1.1.3 Source-induced perturbation development. Evaluate use of Cf-252 source to perturb local core regions.										
7.1.1.4 Delayed-neutron triangulation. Investigate feasibility for HTGRs.										
Task group 7.1 total	3	2	2	2	1	1	1			
Task group 7.2 – HTGR thermometry development										
7.2.1.1 Testing of sheathed Chromel/Alumel thermocouples. Test to 1100°C for times of more than 10,000 hr in fluxes to 10^{14} n/cm ² ; evaluate service life.	2	2	2	4	4	2	2			
7.2.1.2 Qualification of improved thermocouples. Test Geminol, Nicrosil/Nisil, and Platinel under severe conditions.										

Table 1.15 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
7.2.1.3 Development of advanced temperature sensors. Develop single-point devices and sensors using gas as the working substance.										
7.2.1.4 Methods for assuring accurate and reliable thermometry. Develop methods for on-line testing, surveillance, diagnosis, and restoration.										
Task group 7.2 total	2	2	2	4	4	2	2			
Task group 7.3 – Development of in-core flux monitoring	2	3	2	1	1					
7.3.1.1 Testing of in-core flux monitors. Test detectors in simulated HTGR environments.										
7.3.1.2 Application of in-core flux monitors. Observe performance in operating systems.										
7.3.1.3 Development of in-core flux monitors. Extend current technology for fission chambers and self-powered detectors.										
Task group 7.3 total	2	3	2	1	1					
Task group 7.4 – Development of surveillance and diagnostics by noise analysis and dynamic testing	2	3	3	3	2	1	1	1		
7.4.1.1 Noise-analysis methods development. Feasibility studies followed by in-plant tests to demonstrate applicability to HTGRs.										
7.4.1.2 Dynamic-testing method development. Identify subsystems to be monitored and perform in-plant tests.										
7.4.1.3 Overall surveillance and diagnostic system design. Design a system to be integrated into HTGR plants.										
Task group 7.4 total	2	3	3	3	2	1	1	1		
Task group 7.5 – Development of reactor coolant moisture monitoring	2	3	1							
7.5.1.1 Evaluation of current capability. Compile existing information and identify other potentially applicable methods.										
7.5.1.2 Qualification of monitors for HTGRs. Develop systems that offer greatest promise.										
Task group 7.5 total	2	3	1							
Task group 7.6 – Development of data acquisition and processing (DAP) systems	1	2	2	1						
7.6.1.1 Identification of the status and completeness of a DAP system. Establish criteria for evaluating the system in the context of safety.										

Table 1.15 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 7.6 total	1	2	2	1						
Task group 7.7 – Development of post-accident monitoring systems	1	2	2							
7.7.1.1 Range of accidents for consideration. Determine accidents and resultant conditions for monitoring.										
7.7.1.2 Identification of needed information. Define plant status data required for safety at all stages of accidents.										
7.7.1.3 Identify measures to obtain information. Define measurements to be made and development needs to ensure availability of data.										
Task group 7.7 total	1	2	2							
Task group 7.8 – Development of plant controllability	1	2	2	2						
7.8.1.1 Identify desirable plant control features. Identify features needed for power-plant control.										
7.8.1.2 Establish reasonable control schemes. Determine feasible and reasonable modes of control.										
7.8.1.3 Investigate effects on plant safety. Relate desired operating methods to safety.										
7.8.1.4 Investigate changes required to protection system. Determine needs to effect improvements.										
7.8.1.5 Develop performance criteria. Evolve criteria for optimum plant control.										
Task group 7.8 total	1	2	2	2						
Task area 7 total	21.5	24	16	13	8	4	4	1		

Task group 7.1 (failed-fuel location) studies the requirements for and the adequacy of various techniques for locating failed fuel within an HTGR core. Possible approaches include: flux-and-temperature peaking (reference method), fission-product sniffing, source-induced perturbations, and delayed-neutron triangulation.

Task group 7.2 is concerned with the demonstration, and development as required, of thermometers that will perform reliably under HTGR conditions. The work includes qualification of thermocouples, development of advanced thermometers, and development of methods to verify performance, anticipate failures, diagnose failures, repair failures in situ, and improve reliability.

Studies of in-core flux monitors (task group 7.3) would include evaluation of available devices for currently planned applications and development of improved devices for use in normal power operation along with determination of performance criteria.

Task group 7.4 would begin with studies of the applicability of noise analysis and dynamic testing techniques to HTGR systems and components and then develop those approaches that are capable of providing useful surveillance and diagnostic information.

Task group 7.5 is concerned with the demonstration of adequate moisture-monitoring capability for HTGR primary systems. Both high-sensitivity continuous systems and less sensitive devices should be examined.

Task group 7.6 is concerned with (1) the establishment of criteria for evaluating computer-based data acquisition and processing systems for HTGRs in the context of safety and (2) a study of the potential for such systems to give diagnoses of plant conditions and guidance in safety-significant situations.

The subject of study in task group 7.7 is post-accident monitoring in HTGR plants. This work would help to establish information needs in post-accident situations and approaches to ensure the availability of the needed data.

Task group 7.8 would examine the effects of plant control schemes on system safety and contribute to the evolution of performance criteria for control systems that provide optimum control with the required safety characteristics.

1.4.7 Peach Bottom HTGR end-of-life program

The previous task areas include studies that utilize the Peach Bottom HTGR in end-of-life tests and studies. Those studies are summarized here so as to present them in an overall context. The support for the Peach Bottom work is included in the other task areas.

The proposed Peach Bottom testing program involves postpower tests and postirradiation examinations of fuel and of reactor components. The post-power tests include observation of system behavior following injection of hydrogen, deuterium, and tritium into various parts of the system and some studies of source-perturbation techniques for detecting failed fuel.

Postirradiation examination of fuel and reflector graphite elements will be performed to give information on fission-product and element behavior; a control rod will also be examined, and tritium loss from that rod evaluated.

Certain components of the reactor circuit will be examined to determine component behavior, corrosion and/or erosion effects, and fission-product behavior. Detailed examinations will be made of ducting sections, hot valve internals, a circulator impeller, seals, a steam-generator tube sheet and tube bundle, and control and shut-down rod drives. These studies will provide performance information under HTGR operating conditions pertinent to safety evaluation.

PART I

BACKGROUND

2. INTRODUCTION

High-temperature gas-cooled reactors (HTGRs) have been developed to the stage where they are to be used in central station power plants for the generation of electricity. Thus far, in the USA, a small plant¹ has been operated, one commercial-size plant has been licensed,² and several applications have been filed for construction permits for large-scale plants.³ It is therefore, necessary that the safety technology of HTGRs be sufficiently well understood and that the information be widely enough promulgated to assure adequate capability for safety assessment at all stages of the design, construction, and operation of the plants. Information for licensing requirements is presently being supplied by the vendor. However, there is a need to obtain further information so that systems behavior, under both normal and abnormal conditions, can be more precisely predicted. Information developed in a national HTGR safety program will aid and expedite reactor safety design and will aid the regulatory agencies in establishing a capability for independent assessment. More precise predictions of the effects of postulated accidents on the environment should also facilitate site selection. This document describes research and development work that can contribute to the safety technology of HTGRs, as well as to the safety-related technology in the broader context of the development of this reactor type.

The planning guide is divided into three parts. Part I (Sections 2 through 6) provides background information of a generic nature that is pertinent to both of the succeeding parts. It includes a discussion of the currently perceived needs, objectives, and scope of the safety planning guide; a summary description of a large HTGR nuclear steam power plant (based on the 3000 Mwt and 2000 Mwt systems being offered for commercial applications by General Atomic Company); a discussion of the features and characteristics of HTGRs that are of particular interest from the standpoint of safety; and a brief summary of postulated accidents that have been identified by the USAEC⁴ and are treated in the Preliminary Safety Analysis Report for the Summit Station⁵ by Delmarva Power and Light Company. This material is intended to provide sufficient background for a clear understanding of the subsequent parts of the document; however, liberal use is made of references to sources of additional detailed information.

Part II of this document (Sections 7 and 8) comprises a safety planning guide that identifies work to be considered for support by the Division of Reactor Research and Development in HTGR safety. To the extent that safety is a major consideration in all stages of the development of a reactor concept, a large development program could, in principle, be identified with a safety program.

However, the relevance of much of the work to other considerations as well as to safety suggests that such work could be supported by a safety-related effort within an engineering and technology program. Consequently, Part II includes, primarily, selected tasks that are clearly associated with safety considerations. In some cases, where clear distinctions could not be made between direct safety technology and safety-related engineering and technology, specific tasks or task groups were included (or excluded) in the safety guide based on AEC guidance.

Part III of the document (Sections 9 through 14) is primarily a planning guide for a safety-related engineering and technology effort. Its principal purpose is to identify those safety-related technology needs that could reasonably be expected to be fulfilled by an engineering and technology program. However, in order to provide the comprehensive treatment believed to be necessary for an in-depth appreciation of the complex interrelationships among the various tasks, this part includes the work identified in Part II.

In both Parts II and III of the document, the work has been divided into seven major sections, termed task areas, each oriented toward a particular technological aspect of safety and each presented separately. The treatment of each task area begins with a general discussion to define the scope, objectives, justification, and status for the entire area and then proceeds into detailed discussions of the specific tasks. The seven major areas are identified as follows:

Task area 1	Systems and safety analysis*
Task area 2	Fission product technology
Task area 3	Primary coolant technology
Task area 4	Seismic and vibration technology
Task area 5	Concrete reactor vessels and containment structures
Task area 6	Primary system materials technology
Task area 7	Safety instrumentation and control

All of the areas are interrelated, since information from one area may be used in any of the others. Consequently, extensive interaction is to be anticipated, not only among the various task areas within the safety guide, but also with their counterparts in the engineering and technology guide.

*Within the safety program planning guide, this task area also provides the coordination of the effort among all task areas to assure the attainment of the overall objectives.

Within each task area, efforts directed toward given objectives are organized under task groups; further, tasks are arranged in order according to progression from surveys of existing information through reduction to engineering practice. Thus within each task group, tasks are identified by type as source tasks (accumulation of basic data), synthesis tasks (integration of data and models), or validation tasks (experimental confirmation and reduction to practice) and finally by a task number. This method of ordering areas, groups, types, and tasks leads to a four-digit number for each task which is used in the various tables and task-area diagrams for task identification. The following diagram illustrates the basis for the four-digit number for each task:

REFERENCES FOR SECTION 2

1. Philadelphia Electric Company, *Peach Bottom Atomic Power Station*, USAEC Docket No. 50-171, Provisional Operating License (DPR-12) (Jan. 24, 1966).
2. Public Service Company of Colorado, *Fort St. Vrain Nuclear Generating Station*, USAEC Docket No. 50-267, Facility Operating License DPR-34 (Dec. 21, 1973).
3. See, for example, *Summit Power Station Unit 1 and Unit 2, Application for Construction Permit*, Delmarva Power and Light Company, USAEC Docket Nos. 50-450 and 50-451, April 30, 1973.
4. *HTGR Edition of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants*, prepared by the Regulatory Staff, USAEC, July 1973.
5. Delmarva Power and Light Company, *Preliminary Safety Analysis Report for Summit Station Unit 1 and Unit 2*, Docket Nos. 50-450 and 50-451, 1973.

3. NEED, OBJECTIVES, AND SCOPE

Gas-cooled reactors have been under development in the United States and other parts of the world since the earliest days* of the nuclear energy industry. This development has led to the construction and operation of a number of nuclear-electric power plants (mostly, but not exclusively, outside the U.S.) based on gas-cooled reactors utilizing various configurations of fuel, moderator, and gaseous coolant. At the present time, a number of construction-permit applications are being considered within the USAEC for central-station nuclear power plants based on a particular HTGR concept being offered commercially by General Atomic Company (GAC).

3.1 NEED FOR A SAFETY PROGRAM

The above action on the part of the utility companies implies a conviction that HTGRs can be operated not only with the high degree of reliability that is required for central-station plants but also without undue risk to public health and safety. The latter conviction is based on extensive studies and assessments of HTGR safety by the vendor and the prospective operators, performed, to some extent, in support of the licensing applications.

A thorough understanding of all aspects of the safety of HTGRs must, however, also be publicly available to permit the exercise of informed independent judgments regarding such systems. One view of this need was expressed in a letter¹ from the 1972 chairman of the Advisory Committee on Reactor Safeguards (ACRS) to the chairman of the USAEC. This letter stated, in part:

"It also becomes important that the Regulatory Staff develop an independent capability for accident analysis, and that the Staff and the ACRS have available independent expert consultants in various major aspects of HTGR design.

The Committee believes that the necessary source of independent expert consultants to the regulatory groups can best be obtained by an adequate, continuing, federally-supported program of safety research and development which is organized and directed in a manner to meet the regulatory needs. The ACRS recommends that the AEC substantially augment its current modest program on HTGR

*The Oak Ridge Graphite Reactor, which began operation in 1943, was gas (air) cooled.

safety so as to accomplish this function, as well as to provide independent resolution of safety questions and to probe for unidentified safety matters."

This letter also listed several "Examples of HTGR safety research areas which warrant augmentation of new studies ..."

In response to the above recommendation, a letter² from the USAEC General Manager indicated that "the AEC will continue to independently fund the safety research and development efforts necessary to fulfill its own program responsibilities and to assist in providing Regulatory with an independent safety assessment capability."

Another indication of the need for HTGR safety information was expressed in a letter³ from the USAEC Directorate of Licensing (DOL) to GAC in December, 1972. In addition, an attachment to that letter identified five safety issues for which there was an apparent need for research and development to provide additional information relevant to licensing requirements. The above letter from DOL, with attachment, was subsequently transmitted⁴ to the Director of the Division of Reactor Development and Technology (DRDT, now DRRD), stating "It is expected that GAC will accommodate these identified safety R&D needs as well as others that may be identified."² The letter also indicated a "need for assistance in order to implement our assessment of certain areas (of HTGR safety)." A second attachment to the letter to DRDT lists six technical assistance contracts that were implemented by Regulatory and three major areas of HTGR safety research funded by DRDT.

In response to the above correspondence, a letter from GAC⁵ indicated that GAC is continuing to pursue a program of safety research and development related to HTGRs. The attachment to this letter includes summary descriptions of the state of safety technology at GAC with regard to each of the five safety issues identified by Regulatory. In addition it provides a list of 20 topical reports to be published to describe various aspects of HTGR safety technology; three of these have been published as of October 31, 1973. On the basis of this response, it appears reasonable to expect that timely safety technology required to design, construct, and operate HTGRs in a safe and reliable manner will exist, at least within the organization of the principal vendor of such systems. However, it does not necessarily follow that the capability for objective and independent assessment of that technology will be concurrently developed. Neither can it be assumed that the technology developed independently by GAC will, in all cases, meet explicit needs that may be identified by various federal agencies.

In recent testimony before the Joint Committee on Atomic Energy of the Congress of the United States,⁶ past chairmen of the ACRS presented the following statement with regard to HTGR safety research:

"C. HTGR-Temperature Gas-Cooled Reactors (HTGRs)

In a report on safety research dated October 12, 1966 (Appendix F), the ACRS recommendations included one that the nuclear industry and the AEC should promptly initiate a very active safety research program on prestressed concrete reactor vessels (PCRVs). It was recommended that the program include research into anomalous failure modes of such vessels, particularly under pneumatic loading, and that the work should encompass effects of potential structural defects or overloads and problems associated with closures, penetrations, and anchors. Following this letter the AEC did institute a modest safety research program on PCRVs, particularly on load behavior, but this program has since been terminated.

Most recently, in a report dated July 21, 1972, the ACRS stated its belief that it was increasingly important that appropriate HTGR safety research programs be defined and implemented in timely fashion by both industry and the AEC. The ACRS also stated that it was important that the AEC Regulatory Staff develop an independent capability for HTGR accident analysis, and that the Staff and the ACRS must have available independent expert consultants on various major aspects of HTGR design.

The ACRS stated its belief that the necessary source of independent expert consultants can best be obtained by an adequate, continuing, federally-supported program of safety research and development which is organized and directed in a manner to meet the regulatory needs.

The ACRS reiterates its recommendation that the AEC substantially augment its current modest program on HTGR safety so as to accomplish the above functions, as well as to provide independent resolution of safety questions and to probe for unidentified safety matters. The ACRS also reiterates that the necessary industrially supported safety-related HTGR research and development should be implemented on a time scale commensurate with the anticipated commercial application of this reactor type.

Examples of HTGR safety research areas which warrant augmentation or new studies include the following: failure modes of multi-cavity PCRVs, including pneumatic effects; irradiation and structural behavior of reactor graphite; fission product release characteristics as a function of fuel design; safety-related instrumentation and other means or measures to reduce the probability of accidents or detect incipient difficulties; temperature effects on concrete and other important structural materials; potential common failure modes; and dynamic response of the PCRV, the core, and other internals to both seismic and accident excitation."

The testimony⁷ of the Director of DRRD at these hearings included the following statements regarding HTGR safety research:

"The major areas of emphasis planned in the current HTGR Safety Program include preparation of a HTGR Safety Research Program Plan which is to be a document outlining the safety research and development believed necessary to support this concept; fission product technology work including fission product release measurements from fuels and related analytical methods improvement, as well as measurements of fission product release in the Peach Bottom Reactor; primary coolant technology studies including measuring of coolant impurity composition during the start-up of the Fort St. Vrain Reactor; and critical reviews of high temperature mechanical properties of HTGR component materials and of PCRV technology."

The program plan for HTGR safety research and development presented in this document was devised to be responsive to the various needs expressed in this section. In summary, these needs are (1) additional research and development work and (2) independent technical review of postulated accident sequences, their likelihood, and their consequences.

3.2 OBJECTIVES

The needs for an HTGR Safety Program, discussed earlier in this section, establish a basis for defining the fundamental goals of such a program, and, hence, the objectives of this safety and safety-related planning guide. The following subsections present the overall objectives of the planning guide, along with brief summaries of the more detailed objectives associated with the work identified in each of the various task areas.

3.2.1 Overall objective

The primary objective of the safety program is perceived to be the assessment, development, and/or validation of technology that will ensure appropriate levels of safety under postulated accident conditions that could conceivably affect the health and safety of either on-site or off-site personnel. Pursuant to this objective, the safety planning guide identifies research and development work that will contribute to the ability to evaluate HTGR safety under such conditions. The identified work involves the assembly, critical evaluation, and codification of all pertinent technical data and of the methods for using those data in safety assessments. It also includes the acquisition of additional information to verify, supplement, or improve the accuracy of existing data and methods.

The pertinent studies include evaluations and experiments relative to abnormal reactor operation and potential accidents and to safety systems designed to prevent accidents and to limit their consequences. The emphasis in the safety planning guide (Part II of this report) is on abnormal and accident phenomena; however a large body of technology related to near-normal and normal operating conditions is known to be essential for comprehensive safety studies, because such technology establishes the basis from which consequences and likelihood of accidents are predicted. This technology development is treated in Part III of this report. Efforts that fit into this category include studies providing physical property values of materials over long periods of time; improved calculational models for the description of fission product, material, and component behavior; and large-scale tests that may be required to confirm the expected behavior of specific components or systems. Also identified are tasks that use the Peach Bottom and Fort St. Vrain plants for surveillance studies to determine fission product and material behavior under actual HTGR conditions. Thus, the overall safety effort is likely to receive substantial support from the development of safety-related technology; further it is anticipated that the safety program will make use of relevant safety technology that is developed in and made available from other federally funded programs (including those supported by other divisions of the USAEC, such as RSR) as well as from industrial and foreign programs.

An important byproduct of the detailed safety studies identified in this planning guide will be a more thorough understanding of the performance requirements that are imposed on materials, components, and systems in safety-significant situations. This understanding can be of substantial value in the development of safety standards and design and performance criteria for such items. Hence, the

provision of information to support the development of standards and criteria, or in some cases, to contribute directly to their development, is an additional objective of the program.

3.2.2 Task area objectives

As was indicated in Sect. 2, the work identified in this planning guide is divided into seven task areas, each dealing with particular safety aspects. Consequently, the objectives of activities in each area, while ultimately directed toward the overall safety objectives, are more narrowly defined. The following paragraphs provide brief summaries of the principal technical objectives of the work identified in each task area; these are described more extensively in later sections which treat the task areas in detail.

Task area 1, in addition to contributing to the capability for detailed analysis of postulated HTGR accidents, provides a coordinating function among the various task areas to ensure the attainment of the overall objectives. Narrower goals within this task area include: the detailed analysis of specific accident sequences and their consequences; the development and/or evaluation of data, techniques, models, and other information required for such analyses; and the development of models, methods, and data to define more clearly the event sequences that may initiate or develop from postulated events.

Task area 2 addresses the behavior of fission products in an HTGR system. Both normal and abnormal conditions are considered because of the importance of the normal behavior in defining the conditions at the start of an abnormal sequence of events. However, most of the associated technology is placed in the category of safety-related engineering and technology (Part III) and omitted from the safety planning guide (Part II). The objectives of the safety work include a comprehensive evaluation of the available fission-product technology (to help define more explicitly the needs for further research and development) and the refinement of models to describe fission-product behavior in portions of the reactor system and in containment systems (including concrete).

Task area 3 is concerned with the chemical reactions that may occur between impurities (e.g., H_2O) in the primary coolant and the materials which form the primary system, including fuel, graphite, and structural metals. The objective is the development of data and methods to describe such reactions under all conditions of safety interest. The emphasis in the safety planning guide is on the assessment of safety needs in this task area. Technical tasks involving structural materials and components are included in the engineering and technology guide.

Task area 4 deals primarily with the subject of seismology and the effects of seismic events on various portions of HTGR systems, including the core itself, other essential components, and the PCRV support structure. In addition, consideration is given to the subject of soil-structure dynamic interactions. The objective of this work is to develop further the capability for describing the effects and consequences of postulated seismic events. To avoid potential inconsistencies in the division of identified work between safety and safety-related engineering and technology, all of the tasks dealing with seismic events are included in the safety guide. Other, somewhat related work, dealing with the vibration of reactor internal structures and components, is included under task area⁴ in the technology planning guide. The objective of the studies in this task group is to improve the capability for describing gas-flow induced vibrations and their effects in HTGRs.

Task area 5 is directed toward the safety aspects of prestressed concrete reactor vessels (PCRVs) and the containment structures of HTGRs. The objectives are to consolidate and apply, as required, information about the safety-related properties and behavior modes of vital items such as the concrete structures *per se*, the prestressing tendons and wire wrappings, the vessel liner and penetrations, and the thermal barriers. Studies related to containment structures will address those areas of particular interest to HTGRs. Since most of the safety-related research and development work identified in this task area is treated in the technology planning guide, the safety studies depend heavily on the work performed in the technology program.

Task area 6 is concerned with the technology of the materials that are used in fabricating the primary-system components in an HTGR — metals, graphite, and special neutron-absorbing materials. The objective of the work identified in the safety planning guide is to assess the adequacy for safety of the development of additional information in the engineering and technology program that will permit improved definition of the response characteristics of the materials, particularly under accident conditions. The majority of the actual technology development, which is closely related to the needs for normal operation, is identified in the technology planning guide. Throughout this task area, the interest is directed primarily toward mechanical properties and effects, with appropriate consideration of the thermal and irradiation environment; but other phenomena such as corrosion and erosion must also be included. Consequently, this task area is also supported by those portions of task areas 2 and 3 which address materials compatibility topics.

Task area 7 deals with the subject of safety instrumentation. The principal objective of this task is to ensure that the required technology exists or is developed to provide adequate safety instrumentation systems for HTGRs. Within the safety planning guide, this involves primarily the review, analysis, and assessment of safety system requirements that are instrument related. Where additional needed system safety requirements are identified, they will be provided to the engineering and technology program for the development of instruments.

3.3 SCOPE OF THE SAFETY AND SAFETY-RELATED PROGRAM

Although several configurations of gas-cooled reactors may be considered, and ultimately used, for central-station power plants, all the commercial plants to be built in the U. S. in the immediately foreseeable future will probably make use of a single basic configuration. Among the salient features of this configuration are: (1) the use of an all-ceramic fuel in a graphite-moderated, helium-cooled core and (2) the implementation of a more-or-less conventional steam cycle for conversion of the heat energy to electricity. Consequently the scope of the identified activities is currently restricted to considerations that are appropriate to such systems. It must be recognized, however, that as other configurations (such as direct-cycle gas-turbine systems) achieve commercial status, it may be desirable to adapt the program toward any special safety considerations that may be associated with them.

It is generally recognized and accepted that nuclear power stations must provide "defense in depth" to provide maximum assurance that no aspects of their operation present an undue risk to the public health and safety. It is also generally recognized that the required degree of protection can be attained through appropriate application of the three levels of safety as expounded by various agencies involved in the evolution of nuclear energy technology. Since the philosophy of these three levels of safety has been extensively discussed in other documents (see, for example, Ref. 8), they will be only briefly summarized here. The three levels, as defined in ref. 8, are:

1. "Design for maximum safety in normal operation and maximum tolerance for system malfunctions. Use design features inherently favorable to safety operation; emphasize quality, redundancy, inspectability and testability prior to acceptance for sustained commercial operation and over the plant lifetime."

2. "Assume incidents will occur in spite of care in design, construction and operation. Provide safety systems to protect operators and the public and to prevent or minimize damage when such incidents occur."
3. "Provide additional safety systems as appropriate, based on evaluation of effects of hypothetical accidents, where some protective systems are assumed to fail simultaneously with the accident they are intended to control."

There is, of course, significant overlap between these three levels (especially 2 and 3), and it is not always possible to associate a given safety consideration with one, and only one, particular level. Nevertheless, it is believed that adherence to this kind of overall approach will provide the required degree of safety assurance for nuclear power plants.

Since the three levels of safety are closely coupled, any detailed study of reactor safety must consider all three levels, and the HTGR safety planning guide does address all three. The first level is frequently considered to be primarily within the domain of the reactor designer (or vendor). However, to the extent that safety technology must be developed to permit assessment of how well a given design meets these requirements and to provide a basis for the establishment of design criteria and standards, a comprehensive safety and safety-related program must address the first level. In addition, the technology that is developed to permit the evaluation of postulated accidents and the systems which provide accident protection (second and third levels of safety) may produce results that are of value in better achieving the first level.

Because of the close, and often complex, interrelation between safety and the basic design and operation of a nuclear system, it is necessary to establish some basis for determining what work should be included in a safety-oriented research and development planning guide. Potential tasks were first identified on the basis of relation to safety and perceived need. Relationship of a proposed activity to safety and safety-related efforts was presumed to exist if the information to be developed could be associated with one or more of the following categories:

1. Establishment of steady-state conditions, properties, or behavior modes which establish conditions of safety significance at the beginning of a postulated accident.

2. Determination of properties, behavior modes, and/or failure modes of materials, components, and systems under abnormal conditions.
3. Determination of accident sequences and consequences, quantitative as well as qualitative.
4. Identification and evaluation of safety margins through reduction of uncertainties in data and calculational methods used in accident evaluations.
5. Development of design bases, properties, criteria, and standards for materials, components, and systems that go beyond those needed for normal design and operation but are needed for safety assessment. In all cases, consideration was also given to the question of whether or not the needed information was under development and might be made publicly available from other known programs, and only those tasks were included that appeared to require additional support.

The above criteria served to identify both safety and safety-related activities; so an attempt was then made to identify separately the tasks for the safety planning guide through their association with postulated abnormal or accident conditions that may involve risk to personnel. In general, activities related to the fundamental design of the nuclear power plant were placed in the technology planning guide; these included most of the activities associated with item 1, above, and those included most of the activities associated with item 1, above, and those identified with items 3 and 5 that could be related to anticipated plant transient behavior. In addition, in some cases the specific inclusion or exclusion of tasks from the safety planning guide was based on AEC guidance. At the same time, the safety-related tasks were retained to help provide overall guidance for planning purposes and are given in Part III.

In addition to the identification of possible activities, the safety planning guide presents qualitative designations of the relative priorities of the tasks that are listed. This priority rating is intended primarily to indicate areas

in which additional support for safety research and development could significantly enhance the evolution of HTGR safety technology. The priority designations are A, B, C, or D, based on current judgments and the following definitions:

- A Tasks in this category should be supported on the suggested time schedule and at essentially the suggested level if undesirable delays in meeting program goals are to be avoided. Assignment to this category implies that the importance of the task and the amount of work required are such that additional support could be accommodated to expedite the attainment of goals.
- B Tasks in this category should be supported on the suggested time schedule, but lower levels of support may be acceptable for a limited period without undesirable delays in overall program goals. Benefits of increased support are likely to be marginal in most cases.
- C Tasks in this category are desirable, but either their effect on other tasks is relatively small or time is not a critical consideration for development of the information. These tasks could be delayed or supported at lower levels temporarily. Increased support is not warranted at present.
- D Tasks in this category are identified for consideration, but support does not appear warranted at this time. In those cases where insufficient information is currently available for reasonable assignment of priorities, no assignments were made and the reason indicated.

REFERENCES FOR SECTION 3

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3. A. Giambusso, Deputy Director for Reactor Projects, USAEC-DOL, letter (with enclosure) to J. Landis, President, Gulf General Atomic Company, December 4, 1972.
4. L. Rogers, Director, Directorate of Regulatory Standards, USAEC, letter (with two enclosures) to M. Shaw, Director, DRDT, "HTGR Safety Research - REG:RSR-84," December 15, 1972. (Reference 4, which includes all of ref. 3, is reproduced in its entirety in Appendix A of this Program Plan.)
5. J. M. Waage, Director, Engineering Division, Gulf General Atomic, letter (with enclosure) to A. Giambusso, USAEC-DOL, April 19, 1973. (Reference 5 is reproduced in its entirety in Appendix A of this Program Plan.)
6. H. G. Mangelsdorf (1973) and C. P. Siess (1972), Chairman, ACRS, "Testimony before the Joint Committee on Atomic Energy, Congress of the United States, Hearings on Nuclear Reactor Safety," September 27, 1973, pp. 24-25.
7. T. A. Nemzek, "Statement of Mr. T. A. Nemzek, Director, Division of Reactor Research and Development, Hearings Before Joint Committee on Atomic Energy on Nuclear Reactor Safety," Sept. 25, 1973.
8. USAEC, *The Safety of Nuclear Power Reactors (Light-Water-Cooled) and Related Facilities*, WASH-1250, Final Draft (July 1973), pp. 2-2 to 2-5.

4. REPRESENTATIVE PLANT DESCRIPTION

This section describes a large commercial HTGR of a type presently being offered by General Atomic Company. GAC has orders for 3000 MWt and 2000 MWt reactors. The 3000-MWt reactor is described and illustrated, and where there are differences between the two sizes of reactors, the information given first is for the larger reactor with that for the smaller reactor following in parentheses. A brief overall description of the reactor is followed by somewhat more detailed descriptions of the nuclear steam system and of the balance of plant. A short section on the evolution of HTGR design in the U.S. concludes the section.

4.1 OVERALL PLANT DESCRIPTION

The HTGR is a thermal reactor using helium gas as the reactor coolant and graphite as the moderator and core structural material.^{1,3} The initial fuel loadings will consist of a mixture of enriched uranium dicarbide and thorium dioxide in the form of spherical particles individually clad with multilayer ceramic coatings.

A typical plant arrangement for an 1160-MWe (3000 MWt) HTGR is shown in Fig. 4.1. The heart of the plant is the prestressed concrete reactor vessel (PCRVR) that encloses the core, the entire primary coolant system (including the steam generators), and other on-line auxiliary systems where radioactivity is involved. The PCRVR is housed in a reinforced concrete containment building that provides a final fission-product barrier. This building is sized so that it can sustain the transient pressures during the postulated design basis depressurization accident without exceeding its pressure limits.

The steam generators transfer the heat from the circulating helium to a circulating-water-steam system that delivers steam to a standard turbine-generator unit at 2400 psig and 950° to 1000°F. Due to the temperatures attainable in the HTGR system, overall thermal efficiency approaches 40%. A core auxiliary cooling system (CACS) is provided to remove heat from the reactor core when the main coolant

Fig. 4.1. 1160-MWe HTGR plant layout.

4.2 NUCLEAR STEAM SYSTEM

The nuclear steam system consists of the active core and reflector regions, the main helium circulators, the auxiliary helium circulators, the associated steam generators and heat exchangers, and the helium purification system, all of which are contained in the PCRV as shown in Fig. 4.2. The primary coolant — helium — flows from the circulators downward through the core, through the steam generators, and back to the circulators. Only steam from the steam generators leaves the PCRV during normal operation. Each circulator is coupled with one steam generator to constitute a cooling loop, and there are a number of identical loops in each reactor. Reactor sizes are increased by adding more loops for larger cores.

4.2.1 Core

The reactor core of the 3000-MWt HTGR will be made up of about 4000 hexagonal fuel elements. Each element is approximately 14 in. wide across the flats and 31 in. high. The fuel, in the form of coated particles bonded into sticks, is located in blind axial holes in the hexagonal graphite blocks. Axial coolant channels are provided for helium flow. Figure 4.3 shows part of the core of the Fort St. Vrain reactor to illustrate the way blocks are stacked, and Fig. 4.4 shows one control-rod core block. This figure also illustrates the dowel pins and dowel sockets used to align the blocks and to prevent horizontal movement.

In the current design for all large HTGRs succeeding the Fort St. Vrain reactor, each regular fuel block will have 72 coolant holes 0.826 in. in diam and 132 fuel holes 0.624 in. in diam. The 3000-MWt reactor will contain 493 blocks in cross-sectional layer, and the 2000-MWt reactor will be 343 blocks in cross-sectional layer. Both active cores are eight blocks deep.

A group of seven columns of fuel and associated movable reflector blocks constitutes a refueling region. A central column of blocks in each group of seven contains two channels for control-rod pairs and a channel for boron-carbide containing balls that are released by the reserve shutdown system (Fig. 4.4). The active core is surrounded by graphite reflector elements, and the entire assembly is mounted on a graphite block floor, which in turn is supported by graphite posts.

The two types of particles, fissile and fertile, that are used in all fuel elements are shown in Fig. 4.5. The fissile particles contain enriched uranium (93% U-235), or U-233 bred in prior cycles, and thorium, if specified. The fissile particles have Triso coatings

Fig. 4.2 Nuclear steam supply - 1160 MWe HTGR designed by General Atomic Company.

ORNL-DWG 71-10131R

Fig. 4.3. Core block arrangement used in Fort St. Vrain HTGR.

ORNL - DWG 74 - 10133R

Fig. 4.4. HTGR fuel element, showing holes for control rods. General Atomic Company design used in Fort St. Vrain reactor; similar designs are proposed for use in larger plants.

Fig. 4.5. HTGR fuel components.

which have a SiC layer between two low-temperature isotropic (LTI) pyrocarbon coats. The fertile particles contain only thorium and are given a Biso cladding that includes two LTI coats but no SiC layer. Both types of particles have an initial coating of very low density pyrocarbon. This buffer coating absorbs the energy from fission recoil and provides void volume for expansion and gas collection. The outer coatings act as individual pressure vessels, and when intact they contain virtually all the fission-product gases and metals that are generated by fission as well as the fissionable or fertile metal oxide or carbide.

The expected burnup in fissile particles is approximately 70% fissions per initial metal atom (FIMA) for fuel containing U-235. Dilution with thorium reduces this value. Fuel containing only thorium will experience about 5% FIMA. Fuel and core graphite will be irradiated for four years and will receive up to 8×10^{21} nvt of neutrons with energy greater than 0.18 MeV.

The coated particles are bonded into rods using a carbon pitch mixture as a matrix. The rods are carbonized, and in addition to serving as a binder, the matrix improves fuel thermal conductivity and fission-product retention. The components of the fuel assembly are shown in Fig. 4.5. The structural graphite between the fuel sticks and the coolant channels provides neutron moderation and gives the core structural integrity. It also serves to reduce diffusion into the coolant of any metallic fission products released from the fuel.

4.2.2 PCRV

The PCRV^{4,5} provides the primary coolant boundary, encloses all the major nuclear steam system components, and provides a biological radiation shield. As illustrated in Fig. 4.2, the core and the circulator-steam generator loops make up the primary heat generation and transport facility with only steam leaving the PCRV under normal conditions. The concrete vessel has an outside diameter of 106 ft and a core diameter of 37 ft. The vessel is prestressed with interior longitudinal tendons and external circumferential wire-wrap that can be monitored and adjusted if necessary. An interior steel liner forms the primary coolant pressure boundary. This liner is protected from the hot gas on the helium side by an insulating layer termed a thermal barrier, and both the liner and the concrete of the PCRV are water-cooled (Fig. 4.6).

Fig. 4.6. PCR V liner and class A thermal barrier.

4.2.3 Main coolant loops

The schematic flow diagram of primary coolant in the PCRV is shown in Fig. 4.7. Each loop consists of helium circulator and a steam generator. The helium is picked up and returned to plenums that are common for all the loops of a reactor. The steam from all the steam generators joins a common header that feeds the turbine.

Helium enters the core at 640°F and exits at 1434°F. Nominal helium pressure is ~725 psi, and the difference in pressure throughout the coolant loop is ~25 psi.

The main coolant loops remove core heat during all operating conditions including power operation, normal shutdown, refueling operations, and anticipated transient conditions.

4.2.4 Steam generator

Figure 4.8 shows a steam generator as positioned in a PCRV cavity. The different sections and the helium flow paths are indicated. The unit can be removed from the top of the PCRV for maintenance. As indicated on Fig. 4.7, the main steam pressure is 2500 psig at 955°F, and the reheat pressure is 575 psig at 1000°F.

4.2.5 Main loop helium circulators

The helium in each loop is circulated by means of a vertical-axis, steam-driven circulator. These circulators are larger improved models of the type currently installed in the Fort St. Vrain reactor. They are single-stage, axial-flow compressors and have water-lubricated bearings.

4.2.6 Helium purification system

A helium purification system consisting of delay beds and cold traps is installed in each PCRV. This system processes ~10% (13%) of the helium inventory each hour and removes all radioactive materials and impurity gases from the side stream before returning it to the reactor.

Fig. 4.7. Schematic flow diagram of main heat-removal system of 1160-MWe HTGR designed by General Atomic Company.

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Fig. 4.8. Steam generator arrangement.

4.2.7 Auxiliary cooling loops

Three (two) auxiliary cooling loops, each consisting of an electric-motor-driven auxiliary circulator, a helium check valve, and a water-cooled heat exchanger, are installed within the PCRV to cool the core after reactor shutdown when the main loops are taken out of service for any reason. Figure 4.9 gives a schematic of an auxiliary cooling loop. Any two (one) of these loops are capable of cooling the core with the PCRV either pressurized or at containment atmosphere pressure. The operating conditions and design bases of these loops differ from those of the main heat removal system to ensure diversity of cooling as well as redundancy. The diversity stems from the differences in drive (electrical vs mechanical) and in heat sink (cooling towers vs steam turbines and condensers).

4.2.8 Instrumentation

Signal information for neutron flux monitoring, core protection, and plant control is supplied from neutron detectors located within the core and in wells in the PCRV wall outside the liner. There are thermocouples in all coolant outlet channels and in the coolant, but not in the fuel elements themselves. These thermocouples sense overheated fuel indirectly. Failed fuel is detected by analysis of the coolant and located by flux tilting operations and coolant analysis. Moisture sensors are installed in the primary loop to detect steam leakage conditions and to initiate necessary and timely plant protection actions. Pressure is also monitored to detect abnormal conditions.

4.3 SECONDARY SYSTEM AND BALANCE OF PLANT

A remote fuel-handling system and storage facilities for spent fuel are provided. The spent fuel is stored in water-cooled containers with four elements to a container and four containers to a well. The fuel storage wells are located in the reactor service building.

The turbine-generator system operates at 3600 rpm and is of the most advanced design, as the temperature attainable in the HTGR gives steam conditions comparable to those of the most efficient fossil fuel stations.

Fig. 4.9. Core auxiliary cooling system.

4.4 NORMAL EFFLUENTS

There is no liquid waste stream containing fission products leaving the HTGR under normal operating conditions. Only tritium, which could diffuse from the primary coolant to the secondary coolant through the metal tubes of the steam generator, could leave the reactor boundaries through secondary coolant leakage. Nevertheless, an allowance is made for release of fission-product gases associated with an assigned primary coolant leak of 1% per year. The fission products — whether solid, liquid, or gas — can be packaged for removal from the site and stored elsewhere.

One effect of the high thermal efficiency is the small amount of heat that must be dissipated from the HTGR compared with that from water-moderated reactors.

4.5 SUMMARY OF FISSION-PRODUCT CONTAINMENT

Figure 4.10 summarizes, by box diagram, the barriers and sinks provided in the HTGR to prevent fission products from being transported to the environment from the fuel kernels where they are born. Thus, a fission product must first escape from the fuel kernel; it is then barred by the particle coatings. The fuel rod matrix and fuel element graphite next offer large-surface-area sinks upon which fission products may be sorbed. The interior surfaces of the helium circulating system offer the next retention sink, and any fission products carried by the gas can be diverted to and removed by the helium purification system. Tritium can penetrate some metals and pass from the primary coolant to the secondary coolant. Other fission products are contained in the primary coolant system by the liner. The liner is backed up by the PCRV concrete that can act as a delayed bed and sink for certain fission products. Finally, the containment building contains a gas clean-up system that will remove particulate fission products and iodine.

4.6 EVOLUTION OF DESIGN

The design of the 1160-MWe and 770-MWe HTGRs has drawn upon information developed in the course of designing and constructing the Peach Bottom Atomic Power Station and the Fort St. Vrain Nuclear Generating Station. Peach Bottom No. 1, a 40-MWe prototype plant owned and operated by Philadelphia Electric Company, was the first HTGR constructed in the United States. Since starting commercial operation in June 1967, the plant has operated at design steam

ORNL-DWG 73-12620A

Fig. 4.10. Barriers and sinks in HTGR preventing fission-product transport to environment.

conditions of 1000°F and 1450 psig with an efficiency of about 35%. The first core produced 467,000 MWe-hr of electrical energy. The second core has produced over 560,000 MWe-hr and is still in operation (July 1973).

The second HTGR in the United States is Fort St. Vrain, a 330-MWe unit built for the Public Service Company of Colorado. The operating license for this reactor was issued on Dec. 21, 1973, and initial criticality was attained in January 1974. Commercial power operation is currently (March 1974) planned for July 1974. This plant is the first in the United States to incorporate a PCRV. The concept also includes once-through helical-coil steam generators and advanced HTGR fuel elements.

Virtually all of the technology developed for the prototype Peach Bottom and the Fort St. Vrain plants has been used in the design of the large 1160-MWe (770 MWe) HTGR plants. Changes from the earlier two plants have resulted primarily from the increase in plant size and optimization of the steam cycle. However, several design improvements have also been incorporated in the large HTGRs.

The design, construction, and preoperational testing of the Fort St. Vrain plant has already provided much information helpful in the design of the large HTGR. Data from the operation of Fort St. Vrain will be used to check previous design calculations and test results.

REFERENCES FOR SECTION 4

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5. SAFETY CONSIDERATIONS

From the description of a representative HTGR system in the preceding section, substantial differences are obvious between these plants and other nuclear plants based on light-water reactors (LWRs) or the liquid-metal-cooled fast breeder reactor (LMFBR). While these differences relate primarily to the details inherent in the normal implementation of different systems, some have a direct bearing on nuclear reactor safety. The purpose of this section (and the next one, which describes accidents that have been postulated for HTGRs) is to focus attention on those features and considerations of HTGRs that are of interest from the standpoint of safety without expressing any judgments regarding the overall safety of the system.

5.1 PLANT FEATURES

A number of the physical features of HTGRs combine to give these reactors safety characteristics different from those of other reactors. With respect to the reactor core itself, the fuel and moderator in an HTGR are effectively inseparable, with the fuel being completely enclosed by the moderator mass. This arrangement helps to confine fission products that may escape from the fuel and also helps to maintain stable nuclear conditions in accident situations. The cores of gas-cooled reactors are, typically, much larger than those of liquid-cooled systems, and core power densities are correspondingly lower; a typical average power density in an HTGR is ~ 8 kW/liter, as opposed to ~ 50 kW/liter in BWRs and ~ 90 kW/liter in PWRs. This means that HTGR heat sources, even under accident conditions, are more diffuse and, therefore, are less likely to produce extremely rapid temperature excursions.

Since essentially all of the primary system of an HTGR is enclosed within the prestressed concrete reactor vessel (PCRV), the system is closely and rigidly coupled. Thus breaches of the primary coolant pressure boundary (PCPB) are unlikely to result from relative motion of components or failure of large interconnecting pipes. However, this close coupling also leads to relatively complex flow paths for the primary coolant, especially in the steam generators, and tends to eliminate natural convection as a significant mechanism for coolant circulation. Consequently, forced-circulation capability must be provided to achieve substantial heat removal from the core. Under some accident conditions, this implies a need for forced auxiliary cooling to limit the ultimate effects of shutdown decay heat.

The reheat steam cycle used in HTGRs gives pressure differences in both directions between the primary-and secondary-coolant systems. Leaks in the boiler-superheater section of the primary heat exchanger can allow secondary coolant (water) to enter the primary circuit, while leaks in the reheater section can allow primary coolant (helium) to enter the secondary circuit. Potential consequences of leaks in both directions must, therefore, be considered in safety evaluations.

The PCPB in an HTGR is provided by the PCRV with its metal liner and penetration closures. However, this arrangement also has some characteristics of a containment envelope. That is, failures in the liner do not lead to catastrophic losses of primary coolant because the concrete, itself, serves as a backup container. Conversely, concrete failures, which appear as cracks if they occur, do not automatically induce liner failures. Penetrations through the PCRV and liner are necessary, but these can be and are designed to limit the consequences of failure.

The PCRV concept is such that the finished product contains a large number of prestressing members which impart a high degree of redundancy to the structural strength of the vessel; i.e., multiple failures would have to occur to produce significant weakening of the vessel. In addition, the location of these members (outside of the liner) protects them from the effects of nuclear radiation and makes them accessible for continuous surveillance and even periodic testing and replacement if necessary. This type of arrangement eliminates, for all practical purposes, the possibility of a sudden, massive failure of the primary vessel.

Since concrete may be subject to degradation at very high temperatures, the PCRV is protected from the temperatures of the helium primary coolant. This requires that insulation be provided between the helium and the vessel and that heat transmitted through the insulation, however slowly, be removed continuously. A redundant water-cooling system is provided for the PCRV liner, and this system must remain in service under essentially all conditions.

The containment envelope for the HTGR is a reinforced concrete structure which completely encloses the primary system and those parts of the secondary system that are in direct communication with the primary. The purpose of this containment is the same as for any other reactor system — to limit the release of radioactive materials in certain accident situations. However, the fission-product retention characteristics of the primary system, including the PCRV, and the low heat capacity of the primary coolant generally tend to reduce the severity of the service conditions imposed on the HTGR containment structure compared with those for other systems.

5.2 REACTOR MATERIALS

The materials which constitute the reactor system of an HTGR include several which differ from those in other systems and, therefore, impart to this system a different set of safety considerations.

The basic fuel entities in HTGRs are small spherical particles, made up entirely of ceramic materials. Because of the high degree of thermal stability of these materials, HTGR fissile and fertile particles can withstand temperatures several hundred degrees higher than their normal operating temperatures without experiencing a high rate of failure. While there are physical and chemical processes that can produce particle failures, failure rates are predictable on the basis of service conditions. Furthermore, because large numbers of particles are involved (a large reactor core may contain of the order of 10^9 such particles), the failure processes are well suited to the kinds of statistical treatments used for quantitative assessments. The failure mechanisms that do affect the coated particles act only on individual particles so that there is no general tendency for failure to propagate to other particles.

The intact fuel particles, even though they are not metal clad, have excellent fission-product retention characteristics. The Triso-coated particles retain all fission products including the rare gases as well as Cs and Sr. However, the latter two elements tend to diffuse very slowly through the Biso coatings. Thus, while the fission-product release from such particles is not identically zero,* the very low rate of release can be described mathematically, and it is much less than the release from failed particles, even when the fraction of failed particles is very low. Because of the release of fission products, primarily from the small fraction of failed particles that will be present in every reactor, the presence of fission products in the primary circuit must be included in HTGR safety considerations.

The individual fuel particles are embedded in a carbonized matrix to form fuel sticks which, in turn, are completely enclosed within the large graphite moderator blocks that are used to build up the reactor core. Thus, even if fission products are released from

* At least part of the fission-product release from intact particles is due to fissions in uranium that is present in the outer particle-coating layers as a result of very low-level contamination.

the fuel particles, they do not immediately appear in the coolant because they must still diffuse through the matrix material and the graphite moderator to adjacent coolant channels and then evaporate into the coolant stream. Since both the matrix and the moderator materials have substantial sorptive capacities for many fission products (noble gases and iodine are exceptions), they provide important delays in the release of those fission products. The sorption and diffusion of fission products in graphite can be sufficiently well described that the effects of these processes can be realistically factored into HTGR safety evaluations when diffusion is the transfer mechanism.

In addition to its influence on fission-product behavior in HTGRs, the graphite moderator contributes to several other safety considerations. Since graphite is the only structural material in the core, its stability characteristics are an important safety issue. The mechanical strength of graphite increases with increasing temperature up to values substantially higher than those encountered even under accident conditions. Graphite is also relatively insensitive to thermal shocks and is chemically inert in the normal primary-system environment. However, graphite does react endothermically with water at high temperature to produce combustible reaction products; so the effects of water entry into the primary system must be considered in safety assessments. The large mass of material present in an HTGR core tends to minimize the effects on the graphite structure itself. Graphite does undergo damage, in the form of dimensional changes, due to irradiation with fast neutrons; based on present understanding of this process, the effects are small for the exposures anticipated in current reactor designs. As a consequence of all of these factors, loss of core structural integrity can effectively be eliminated from consideration under all credible situations.

The fuel-graphite arrangement in the core of an HTGR inseparably couples a mass with a large amount of thermal inertia to the heat source. This is particularly significant under certain accident conditions where fission-product afterheat increases core temperatures in the temporary absence of adequate cooling. As a consequence, rates of core-temperature rise for loss-of-coolant accidents in HTGRs are about 20°F/min immediately after reactor trip; these rates are much lower than those in liquid-cooled systems (where a substantial fraction of the thermal inertia may be lost with the coolant). This, in turn, allows more time for recovery of cooling capability before other unfavorable conditions are reached. In addition, this high heat capacity permits adequate heat removal at low coolant pressures under shutdown conditions.

The helium coolant in an HTGR remains in the gaseous state under all normal and abnormal conditions. Consequently, its behavior is easily predicted, and the mathematical expressions for such things as equations of state and heat-transfer coefficients are simple. (Phase changes between liquid and gaseous states add substantial complexity to the analysis of liquid-cooled systems.) However, helium is a low-density, low-heat-capacity material, compared with liquid coolants, so that large masses and even larger volumes must be moved to transport a given amount of heat. In addition, thermal convection is relatively ineffective as a method of heat transport. This combination tends to impose relatively stringent requirements on heat removal systems for abnormal conditions. On the other hand, the low heat capacity imposes relatively mild thermal loads on containment components in case of a loss of coolant event.

The primary coolant system operates at moderate pressure (~50 atm) so that only moderate mechanical stresses are imposed on the system. Furthermore, since helium is itself chemically inert, corrosion by the coolant in the absence of impurities is not a consideration. Both of these factors are significant in considering mechanisms that could lead to breaches of the PCPB.

The structural metals of the primary system of an HTGR are exposed to service conditions that differ substantially from those of other reactors. Although mechanical stresses can be kept low, the average core-outlet helium temperature in normal operation is typically around 750°C, and substantially higher temperatures can be experienced in off-normal situations. Therefore, extensive use is made of high-strength high-temperature alloys (like Incoloy 800), and auxiliary cooling is provided to some components, e.g., the PCRV liner. Even so, the possibility of exceeding acceptable service conditions is an important consideration in the safety of these reactors along with the long-term behavior of structural metals under normal conditions. In addition, possible adverse effects of coolant impurities on structural metals must be considered.

Owing to the presence of a thick graphite reflector around the core, the neutron irradiation environment to which most of the metallic structures are exposed is relatively mild. Therefore, neutron irradiation damage to the metal components is not a major consideration. An exception to this situation is the structural material in the control rods. However, they are designed for replacement several times during the life of the reactor so that long-term irradiation damage is not incurred.

The neutron poison used in both the control rods and the backup system is boron carbide, a material with good high-temperature stability. In addition neither system depends upon the integrity of the primary vessel or the presence or absence of the primary coolant to remain functional. Consequently, potential loss of neutronic shutdown capability is not a major consideration in HTGRs.

5.3 NEUTRONIC FEATURES

The neutronic characteristics of HTGRs do not differ dramatically from those of other thermal reactors, but these features are closely related to the safety of the systems. The use of graphite as a moderator endows these reactors with a relatively long prompt-neutron generation time. Although this time is not a major consideration in most nuclear events (because prompt criticality is difficult to achieve), it does tend to slow the response of the reactors to some reactivity perturbations. After an initial period of operation with ^{235}U fuel, a major fraction of the power generated in HTGRs is a consequence of ^{233}U fission. Therefore, the reactors will normally operate with lower delayed-neutron fractions and correspondingly faster response to reactivity disturbances. However, this effect is probably of minor importance from the standpoint of nuclear stability.

Gas-cooled reactors containing substantial amounts of fertile material characteristically exhibit moderate, but negative, temperature coefficients of reactivity over the entire temperature range of interest. Small coefficients limit the temperature defect and, hence, the amount of reactivity that must be provided by the control rods, between cold and operating conditions. On the other hand, they also limit the system's capability to compensate for reactivity changes without large variations in average temperature. This limitation is partially compensated for by the fact that there is little reactivity associated with the primary coolant. That is, variations in coolant flow or condition will not induce significant reactivity effects that must be checked by the system.

Compared with other commercial power reactor systems, HTGRs are physically very large. While large reactors may be susceptible to local power variations such as spatial xenon oscillations, physical size is not the only contributing factor. It is, in fact, possible to design large cores that are inherently stable with respect to xenon oscillations, and the current generation of HTGRs apparently is stable. The presence or absence of such inherent stability can significantly influence both reactor-control and safety considerations.

6. HTGR ACCIDENT ANALYSES

Providing maximum assurance of nuclear power plant safety through application of the three levels of safety discussed in Sect. 3 gains practical expression in the detailed design of the plant. Backup protective systems are provided whose actions are invoked if priority defenses fail, the ultimate defense for the public being the plant containment. In the course of the plant design, therefore, it is necessary to examine the full spectrum of defense situations and the full ranges of transient conditions that could be encountered by systems participating at each level of defense. The practical means for such accounting is to postulate and evaluate various incidents. This section provides a summary of the analyses that have been performed for postulated accidents in large HTGRs along with some discussion of the computer codes that have been used. In addition, several highly simplified accident sequence diagrams are presented to couple the work proposed in this program plan to identified safety considerations.

6.1 POSTULATED ACCIDENTS

6.1.1 Selection and grouping of incidents

The major incident categories of interest to HTGRs were generally identified during the early conceptual design of these plants. They followed from consideration of the obvious fission-product escape mechanisms and routes and provided an initial basis for deciding what kinds of protection were necessary and what considerations would determine plant operating limits. Specific incidents were subsequently identified within the major categories with due recognition of the possibilities of deficient performance by the various operating and safety systems proposed for actual plants. Such incidents are continually being reviewed, redefined, and re-evaluated with increasing realism as plant design evolves.

Typical incidents chosen for evaluation of defense response stem from postulated credible failures of operating systems or externally caused disruptions of plant operations. In many cases, the operating systems themselves are able to cope with the initial contingencies; hence, failures must be compounded before any protection systems are even challenged. Other highly improbable event combinations or sequences must then be envisioned in order to assess effectiveness of the ultimate lines of defense. Two other considerations influence the selection of incidents or their associated parameter values: (1) events, event sequences, and some of the parameter values should be realistic (albeit unlikely) in order for

the response assessments to be valid and (2) certain conditions or parameter values may be exaggerated "conservatively" for purposes of encouraging the large design margins appropriate to potential large releases.

Natural phenomena are considered because of their potential for inducing incidents via component damage or disruption of desired operation sequences. The design bases of Class 1 structures and equipment, therefore, include capability to withstand such events as earthquakes, tornados, and floods. Any operating contingency or system failure induced by a natural event should invoke protective actions in their regular order and hierarchy.

A number of potential incidents of interest to HTGRs have been studied or identified for study thus far.¹⁻³ In general, the greatest attention has been given to those for which the defense provisions govern principal aspects of the plant design or on which the HTGR safety concepts are based. Accident considerations and their consequences are continually under review as the plant design details evolve, and the advancement of this review capability is one of the purposes of the safety program.

Incidents of safety interest with regard to plant, system, and/or component design are discussed below in three categories: (1) anticipated plant contingencies, (2) incidents that are credible but avoidable through good design and operating practice, and (3) extreme situations which are not expected to occur to the postulated extent in a well-engineered plant, but which provide the means to define the large safety margins needed for low release risks. These categories correspond generally to the plant conditions A, B, and C defined in Sect. 2.1.1 of ANS-23 (ref. 4), the proposed standard for GCR safety criteria. The categories include essentially the same incidents discussed in the Delmarva Summit Station PSAR¹ and the AEC format guide,³ ordered here by increasing potential to cause significant radioactivity release. The AEC guide does not group the incidents, while the PSAR simply classifies them according to whether or not any release is indicated by the reactor manufacturer's analysis.

6.1.2 Anticipated plant occurrences

This category groups incidents that could reasonably be expected to occur over the combined lifetimes of actual HTGR power plants. Their statistical frequencies are estimated individually to be in the range of one per plant-year to one per 100 plant-years.

Power generation disruptions

Two distinct types of operating contingencies are considered to be ordinary power generation disruptions: (1) the plant forced outage that involves neither separation from the main distribution grid nor loss of main circulator core cooling capability and (2) the plant electrical separation from the main grid where the main generator is able to continue supplying the plant auxiliary loads. In both instances, the preferred response is to run back the reactor power to a standby level, nominally 25% of rated power. Where a plant forced outage is a consequence of or otherwise requires reactor shutdown, the core is subsequently cooled down with the main circulators and steam generators; an auxiliary fossil-fueled boiler makes up the steam supply deficiency that develops as the core cools.

Loss of offsite power (LOSP) and concurrent turbine trip (TT)

If a forced plant outage is accompanied by a loss of normal station power (main turbine generator and off-site power), a different shutdown sequence is followed. However, that portion of the sequence that releases secondary coolant and activity to the environment is the same as in the previous case.

Given such a total loss of auxiliary power (i.e., LOSP and TT), the HTGR main circulators can still function while there is a supply of water to the main boiler feed pumps. Such reserve is available from the deaerators to support reactor shutdown cooling for about 15 min. During that interval, the on-site emergency engine generators are started and extended cooldown by the electrically powered CACS is initiated.

Any deficiency of the interim cooling by the main circulators increases the urgency of prompt startup of the CACS, as discussed below. Excessive delay in starting the CACS, or reduced capability of that system, is considered to be a "cooling interruption accident," as discussed in Sect. 6.1.4.

Loss of main loop cooling (LOMLC)

Of the several ways in which the nuclear steam supply can become disabled, only loss of main condenser vacuum ranks as an "anticipated plant occurrence" in terms of ordinary power plant experience. Otherwise it is necessary to postulate rare events such as failure of steam, feedwater, or condensate lines common to all steam loops.

Any major failure of the nuclear steam supply requires shutdown of the reactor — immediately in the event of major line ruptures — and places upon the CACS the entire burden of shutdown core cooling. Alleviating the situation is the small likelihood that the steam-system failure will occur so abruptly as to disrupt the external power network; that is because the onset of condenser difficulties is usually gradual enough to permit orderly shutdown of the plant.

As in the case above, LOSP with TT, CACS deficiencies are considered as the "coolant interruption accident."

Tube rupture in main steam generator

Among the hundreds of tubes, header welds, etc., in the HTGR steam generators, occasional failures may be expected. These may be due to a variety of causes including initial defects, fatigue, or corrosion, possibly aggravated by operating transients. Other sources of potential water inleakage to the primary coolant exist; however, the quantities of water involved are comparatively small.

Elaborate provisions are made to detect moisture in each steam generator loop of the helium circuit. Protective actions based on this information are appropriate to the water inleakage rate. A small leak could continue until the moisture concentration actuated an alarm, presumably on the source loop; the operator would first monitor moisture buildup rate of all loops to confirm that the leak was located in the loop that gave the alarm and would then manually shutdown and isolate the offending loop and dump its contents to a dump tank. Should this procedure halt the moisture concentration buildup, the reactor could remain in operation with the coolant cleanup system eventually removing the moisture and steam-carbon reaction products.

A larger leak might raise the source-loop moisture concentration to an unacceptable operational level before manual intervention could prevent it. Protective-system logic would then automatically initiate loop shutdown, isolation, and dump and would also scram the reactor. The core would then decrease rapidly in temperature, cooled by the operating main loops, thus inhibiting the steam-graphite reaction and preventing significant attack on the core structure or fuel coatings. Should the primary logic for any reason select the wrong loop for shutdown, backup logic is available to make a second selection, subject to operator confirmation; shutdown, isolation, and dump of that loop would then be manually initiated.

Any water introduced to the primary coolant circuit occasions some steam-graphite reaction, with possible helium entrainment of fission products previously deposited on surfaces. Other reactions may tend to release fission products plated onto metal surfaces. Moreover, the steam-carbon reaction products - CO , CO_2 , and H_2 - and the steam itself add their partial pressures to that of the initial helium. None of these effects, however, become serious until the quantity of water introduced is very large and unless core temperatures remain at or above normal operating levels. The fission products that become entrained remain in the coolant loop until removed by the cleanup system. A hypothetical major inleakage accident, resulting from failure to isolate and dump a faulty loop, is discussed in Sect. 6.1.4.

Primary coolant helium leak to containment building

With helium contained at about 700 psi in the PCRv, a rather small breach of the liner or penetration welds or the rupture of an instrument or process line can cause coolant leakage at a significant rate. Some of the lines are monitored for radioactivity and could be valved off when activity was detected due to leakage flow; depending on the function of the isolated line, the plant might then continue normal operation. Other leaks would be detected by the containment activity monitors, which initiate containment isolation. Reactor shutdown would become necessary for any significant leak that could not be isolated or plugged, so that the helium inventory could be transferred via the cleanup system to storage.

The primary coolant always contains small amounts of radioactive material, mainly iodine and noble gases. During a leak, some portion of the iodine and metals might adsorb onto the cooler surfaces of the leak path; however, this process is not currently given much weight in safety analyses. Of the residual materials introduced to the containment, most would be removed by the containment atmosphere cleanup system, and the remaining condensibles would collect on cool surfaces.

Larger leaks introduce consideration of rapid depressurization effects, including flow forces on components and flow entrainment of radioactive materials plated on primary circuit surfaces. These are discussed in Sect. 6.1.4 as part of the "design basis depressurization" incident.

Power-cooling mismatch

Any dynamic control situation implies continual small mismatches, or "errors," among the control parameters which induce responses in direction to reduce the errors. But if independent variables

change by large amounts or more rapidly than the controlled system responses can follow, if they oscillate at critical frequencies so as to induce out-of-phase controlled responses, or if controlled responses are inadvertently in the wrong direction, serious instabilities may result. Of safety interest in this situation are conditions in which the reactor power could substantially exceed the heat removal rate, causing the reactor to store excess heat and reach high temperatures.

One of the more obvious and severe control transients occurs when one boiler feed pump is lost, i.e., when half of the feedwater supply to the boilers is lost. Similarly, various boiler-control malfunctions could change the heat removal rate without corresponding change in reactor power. More difficult to predict are the possible cyclic demands associated with power network disturbances; although the negative temperature coefficient in an HTGR generally enhances favorable plant response to such demands, inadequate imposed controls intended to supplement the inherent response could aggravate the transient differences between power and cooling. In addition to the "direct" effects noted, it is also conceivable that transient excess cooling could add reactivity and so induce a power excursion. However, the inherent features of HTGRs tend to minimize the consequences of such events.

Ability to handle disturbances externally induced or resulting from failures of components such as the boiler feed pump is a major criterion for the design of the plant control system. Failures of the control system itself may require operator intervention, e.g., to override faulty automatic features, to reduce plant loads temporarily until malfunctions can be corrected, or to shut the plant down orderly. The plant protective system will scram the reactor automatically if any of the safety parameters are violated.

Inadvertent addition of reactivity

One of the classical accidents postulated for all reactors is the inadvertent addition of nuclear reactivity. This could occur in an HTGR by control-rod withdrawal due to operator error or control malfunction, by a rod breaking so that one portion falls out of a core region of high importance to one of lower importance, by error or accident during fuel handling, or by a rapid decrease in core temperature.

Various design features of the HTGR plant are intended to limit the amount of reactivity that can be added with the control rods. These include positive prevention of gang-rod withdrawal, limited

reactivity investment in a rod pair, control insert-command override with withdraw-command, positive prevention of rods dropping through the core bottom, and prohibition of all rod withdrawal when safety conditions are not satisfied. Should excess reactivity be added despite the restrictions, a controlled rod rundown might be attempted. The protection system would ultimately scram the reactor if power increased or an excursion developed beyond prescribed safety limits.

The most serious event in this category is the uncontrolled withdrawal of a single rod pair during full-power operation accompanied by failure to scram on high neutron flux. In this case, reactor shutdown is delayed until the high temperature of the helium entering the steam generators actuates a trip. This sequence results in some local overheating and failure of fuel, but the activity released from the fuel is retained in the primary system.

Small spills or leaks of radioactive material outside of containment

Despite all precautions in the management of radioactive material, incidents can occur which allow small discharges of activity. These may involve mishandling of fuel, waste, or source containers; improper venting of facilities; improper discharge from hold tanks, etc. Monitoring systems are provided to detect such releases, and plant ventilation is arranged to isolate their effects. Such incidents represent primarily a local hazard.

6.1.3 Unlikely serious incidents

This category contains incidents which are credible but largely avoidable through good design, quality assurance, and operating practices; several have the potential to severely disrupt plant operations and require repairs that would keep the plant out of service for extended periods. Very few occurrences representative of this entire category are expected over the cumulative service life of many HTGR plants, e.g., perhaps altogether one per 100 to 1000 plant years.

Large additions of reactivity

Various accidents have been postulated whereby larger reactivity excursions than considered in Sect. 6.1.2 might be produced. Among those originally considered were rapid ejection of a rod pair, gang withdrawal of control rods, massive inleakage of water, and rearrangement of core components due to earthquakes. The rod-pair ejection and gang withdrawal accidents are positively prevented by

system design features, and other analyses that were performed concluded that the credible processes introduce much less reactivity than the amounts which can safely be countered by control-rod scrams. Further, protection against rod-insertion failure is afforded by a backup scram system that allows boron carbide pellets to fall into core passages.

Coolant channel blockage

With HTGR cores of the order of 30 ft in diameter, there appears to be no credible way that the entire top surface could be blanketed to obstruct coolant flow. The possibilities that are seriously considered concern partial obstruction of one refueling region or a few coolant channels. Since the coolant flow for each region passes through a single control orifice into a distribution plenum, the externally induced blockage incident must postulate (1) an obstruction of the orifice inlet ports, possibly by pieces of thermal barrier cover plate or pieces of thermal barrier material, (2) failure of the orifice assembly, or (3) penetration through the orifice of materials such as small pieces of metal, graphite, or thermal barrier, which then block individual passages rather than an entire region. The other class of incidents involves failure of the core graphite in such a way as to block coolant flow. Total loss of flow through an entire refueling region is not considered credible.

Depending on the extent and degree of flow blockage, very high local or region temperatures could be reached during reactor operation and possibly after shutdown for large blockages. With significantly restricted flow down through a refueling region, the region exit coolant temperatures will indicate that high temperatures are developing in the core above. Because the exit coolant temperatures are monitored by thermocouples, the temperature anomaly can be observed and the appropriate corrective action taken. An alternate means of detection is the observation of increasing coolant radioactivity due to fuel particle failures. Obstruction of one or two coolant passages might not be clearly indicated by the exit temperature monitors because of the small decrement in helium flow, and the resulting coolant activity rise may be small enough that no positive identification of the blockage is made.

The possible very high local temperatures could cause damage to fuel particles and control elements, even to the extent of melting and redistributing. However, such effects do not tend to be self-propagating in this system, so no threat to the heat removal facilities or the PCRV integrity is presented. The limiting direct consequence, therefore, could be contamination of the PCRV internals with core materials.

Fuel handling or storage accidents

Beyond the minor incidents previously discussed, several radioactive materials management accidents having potentially greater release consequences are identified. They are (1) loss of cooling of the fuel handling machine, (2) loss of fuel storage cooling, and (3) fuel container drop.

Incidents (1) and (2) occur inside the containment structure; hence any radioactivity release invokes containment isolation and cleanup actions. The fuel container drop could occur outside of the containment boundaries; in which case other cleanup procedures might be required. High-efficiency particulate filters in the main plant ventilation system reduce the spread of radioactivity.

Gas waste tank rupture

Low-level contaminated waste gases are stored outside of containment in the service building and could escape in the event of a pipe, valve, or tank rupture. When such escape is detected by local radioactivity monitors, affected areas of the service building can be isolated and vented through HEPA and activated charcoal filters to limit releases to the environment.

Control room uninhabitable

The HTGR plant control room, with its adjacent instrument rooms, cable trays, etc., is the "nerve center" of the plant. However, it must be considered as subject to evacuation should smoke or other disruptions make it uninhabitable. Such conditions may require the plant to be shut down, and provisions are made to perform all necessary shutdown operations and monitoring from other locations.

The objectives in analyzing this incident parallel those of the plant design, namely, to assure that all necessary functions for safe shutdown can indeed be performed remotely. Moreover, no event so affecting the control room can also be allowed to drive the plant toward an unsafe condition or to disable the remote control facilities; accordingly, the analysis considers the potential for such effects.

Major secondary coolant leakage

One aspect of a major steam line rupture, the possible LOMLC, was discussed in Sect. 6.1.2. There remains the additional prospect of discharging large amounts of steam inside or outside of containment. As indicated previously, such releases can also occur in load rejection incidents. Since the steam contains very little radioactivity, the release consequence is negligible. The heat load imposed on the containment structure by the steam is well within its capability to dissipate.

Primary coolant leakage into secondary coolant system

The reheat sections of the HTGR steam generators operate at pressures lower than the primary coolant helium; hence, a tube rupture or leak allows helium to enter the secondary coolant. Radiation monitors on the reheat steam lines are intended to detect helium and to initiate isolation of the faulty steam generator and shutdown of the reactor.

If the leak were very large, sufficient helium might pass to the main condenser to raise the pressure there considerably. The rupture disc provided for condenser overpressure protection would then operate to permit a steam and helium release along with some radioactivity. Prompt isolation of the leaking steam generator would limit the associated radioactivity discharge to a small amount.

6.1.4 Extreme situations

This category postulates events or event combinations of such safety significance that (1) principal design and development efforts are devoted to precluding their occurrence and (2) design of the plant and facilities to withstand or counter their effects assures adequate protection margins for all less severe incidents. Incidents or conditions as severe as postulated here are not expected to occur.

"Design basis" depressurization accident (DBDA)

A sudden depressurization of the PCRV through a 100-in.² opening is assumed for each of several locations corresponding to major top penetrations; the escape cross-section area is established by flow restrictors installed in each penetration to limit flow should a closure fail. The blowdown flows may be sufficient to entrain portions of the radioactive materials deposited on loop surfaces

or present in dust accumulations. Actions immediately following the accident would include reactor scram, turbine-generator shutdown, and containment isolation. Long-term core cooldown can proceed with the main steam generators and circulators, assisted later by the auxiliary boiler. The resulting radioactivity releases from the containment are indicated to be well below the AEC limits. The containment design basis is such that the transient pressures and heat loads can be accommodated.

Other phases of the accident analysis establish that the blowdown pressure differential forces should not damage the PCRV internals, in particular the components of the loop in which the penetration failure occurs. Neither do these forces cause core levitation, which could have serious consequences. Analyses have demonstrated also that addition of LOMLC, aggravated by delayed CACS startup, failure of one CACS loop, and failure of one main-loop helium reverse-flow-prevention valve, can be tolerated without significant damage to the PCRV internals from high temperature.

During the long-term cooldown following the DBDA, some air-helium mixture from the containment will enter the PCRV. The air reacts with core graphite to produce CO, but the indicated concentrations are well below flammability limits.

"Design basis" inleakage accident

In current HTGR safety analyses, a steam-generator tubesheet failure is postulated to occur in such a way that the initial rate of water intrusion into the primary system is higher than could be achieved through offset failure of a single tube. In the worst case it is further postulated that the moisture monitors on the affected loop fail and that dispersion of the moisture in the system causes automatic isolation and dump of another loop. Since this action is allowed for only one loop, the plant operator must then manually initiate isolation and dump of the faulted loop. It is assumed, however, that the operator also selects a wrong loop and that water ingress continues until shutdown of all the main loops terminates the leak. Main-loop cooling of the core is assumed to continue up to this point, followed by manual actuation of the CACS.

The substantial moisture inleakage raises the primary loop pressure to the level of the pressure relief valve setting, causing the valves to open and discharge into the containment; the valves subsequently reseal. Manual shutdown and isolation of all main loops, including the leak source, is assumed to be completed 30 sec after the relief valve closure.

The most significant safety aspect of the accident is that hydrolysis of carbide fuel particles with failed coatings releases substantial quantities of radioactive materials that could become entrained in the primary coolant; operation of the pressure relief valve then permits some of this activity to enter the containment. Effective operation of the containment, however, keeps the environmental release well within tolerable limits. The moisture remaining in the PCRVR after the relief valve reseats continues to circulate with the primary coolant; portions diffuse into the core graphite to react with carbon in the hottest regions. Reaction rates, however, decrease as the core is cooled, becoming negligible after about 1 hr following plant shutdown. The small amount of CO and H₂ discharged to containment dilutes to concentrations well below flammability limits.

Cooling interruption accident

Extrapolations of the LOSP and TT and LOMLC (see Sect. 6.1.2) accidents are considered with respect to delays in placing the CACS in service (e.g., due perhaps to emergency power being unavailable), CACS loop failures, bypass of CACS loop flow due to failure of main-loop backflow prevention valves, and combinations of the foregoing. Salient conclusions from these analyses are (1) CACS startup delays on the order of 1 hr may be tolerated without damage to the most vulnerable of the PCRVR internals — the top plenum thermal barrier cover plate; (2) operation of two CACS loops is required for satisfactory shutdown cooling of a 3000-MWt HTGR or one loop for a 2000-MWt plant; (3) failure of two main-loop backflow prevention valves can be tolerated without damage to core or PCRVR internals; and (4) extended delay of CACS startup allows natural convection loops to become established among core regions, which must be overcome by subsequent two-loop CACS operation in the 3000-MWt plant or one loop in the 2000-MWt plant.

6.2 COMPUTER PROGRAMS

Owing to the wide variety of accidents considered for these reactors, several computer programs or codes have been used by the HTGR vendor and license applicants in the U.S. to analyze the system behavior. Some of these codes are generally applicable to nuclear systems, while others are adaptations of general codes and still others were especially developed for these systems. The purpose of this section is to identify and briefly describe the computer programs that have been applied to the safety analysis of HTGRs by the principal U.S. vendor and his customers. Where appropriate, comments will be made regarding other codes with comparable capabilities that are known to exist, either in the U.S. or abroad.

6.2.1 Reactor statics codes

Reactor statics calculations provide much of the basic data that is essential to safety evaluations inasmuch as they supply core power distributions, various nuclear reactivity coefficients, control-rod worths, and other neutronic properties.

The procedure for computing the basic neutronic properties of a reactor begins with the processing of raw nuclear data from sources such as the ENDF library⁵ and proceeds through spectrum calculations, to full reactor calculations in one, two, or three dimensions, as required, to produce the desired end product. The multiplicity of codes used for this process is comprehensively summarized in ref. 6, and no attempt will be made here to describe, or even mention, all of the codes and combinations thereof that are applicable.

The code GAND2⁷ is typical of several that may be used to reduce ENDF/B data to "fine-group" average cross sections suitable for use in spectrum and cell calculations. This process must, in principle, be carried out at several temperatures in the range of interest for all of the nuclides to be considered. The resultant cross-section sets are then processed by spectrum codes, of which MICROX⁸ is an example, to produce "broad-group" cross sections. These calculations must consider specific nuclide mixtures as well as geometric factors to account for heterogeneities in the reactor core. Both of the above steps are required for any nuclear core calculation, and many other codes are available throughout the world to perform the computations. However, HTGRs add some complexity to the second step. The coated fuel particles themselves lead to heterogeneities in the fuel rods which, in turn, are heterogeneously mixed with the moderator graphite. Accurate spectrum averaging, in principle, requires consideration of both levels of heterogeneity, and MICROX provides for this treatment.

Once the appropriate sets of broad-group cross sections are available, a large number of few-group codes may be used to calculate the reactor characteristics of interest. Although either diffusion-theory or transport-theory codes may be employed, the properties of large HTGRs are generally such that the diffusion-theory treatment is adequate. Representative codes used by OAC are GAZE-2⁹ and FEVER¹⁰ for one-dimensional and GAMBLE-5¹¹ and GATT-2X¹² for two and three dimensions, respectively. Other applicable codes of this type include CITATION¹³ and PDQ-7.¹⁴ In general, these codes may also be used for perturbation theory calculations and for fuel depletion and management. Thus, in appropriate combinations, they provide most of the required reactor statics information.

6.2.2 Reactor kinetics codes

Also in the area of reactor physics, nuclear kinetics codes are generally available throughout the nuclear community, ranging from simple point-model treatments to detailed space-time kinetics and coupled neutronics-hydrodynamics-thermodynamics codes. Again, HTGRs require some peculiar considerations, in this case, because of heat-transfer time lags associated with the fuel particles and the solid moderator between the fuel and coolant. At the same time, the lack of a phase change in the coolant eliminates the need to consider some other complex phenomena. The BLOOST codes, of which BLOOST-7 is the most recent version, were developed by GAC to supply the required reactor kinetics analysis capability for HTGRs. While this particular version is not now generally available, the basic capabilities also exist in BLOOST-5¹⁵ and BLOOST-6.¹⁶ These codes are based on space-independent kinetics with two-dimensional treatment of the heat-transfer considerations. One other code that has at least potential applicability in this area is COSTANZA,¹⁷ which is nominally available from the NEA library. This program considers variations in the spatial flux distribution during transient calculations.

6.2.3 System kinetics codes

Detailed accident analyses require evaluation of the transient response of the entire plant to upset conditions. The analytical tool developed by GAC to study the transient behavior of HTGRs is the Transient Analysis Program (TAP),¹⁸ 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 balance of the plant and auxiliary equipment having a significant effect on nuclear steam supply 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 and temperature and pressure at numerous points throughout the nuclear steam system. A second program, also available in this general area, is the Reactor Emergency Cooling Analysis (RECA) program¹⁹ which deals with a particular class of events involving the CACS.

6.2.4 Core thermal transients

While the system kinetics codes, discussed above, provide information about the plant as a whole, many situations require more detailed analyses in particular areas, and such analyses can often be separated

from the more "global" treatments. One of these areas deals with thermal transients in the core, particularly under conditions of either local or general impairment of coolant circulation. The general heat conduction code, CINDA,²⁰ has been used to analyze local temperature effects for such conditions. Another heat-transport code, HEATING3,²¹ is currently being adapted, under USAEC sponsorship, to treat problems of this nature.

6.2.5 Steam-graphite reaction

Because of its complexity, the steam-graphite reaction is usually treated separately in safety studies relating to HTGRs. To date, at least six computer codes have been produced to treat this process. The OXIDE code, developed by GAC, now exists as OXIDE-3.²² For defined (input) primary-system transients and steam and/or air ingress rates, the code solves mass, continuity, and state equations for all gaseous species assuming perfect mixing in core inlet and outlet plenums. The core is modeled by up to 3400 nodes for regional orificed coolant flow, heat generation and heat transport, in-pore diffusion and chemical reaction of oxidant gases (H_2O and O_2) and generated products (CO and H_2) and potential fission-product release due to graphite oxidation and fuel hydrolysis.

In addition, particular actions in the remainder of an HTGR plant can be simulated. STEAMCAR,²³ an ORNL code, provides a steady-state analysis of graphite corrosion by steam. It includes consideration of the rate of steam-graphite reaction, diffusion of steam through the graphite, and steam-graphite contacting through turbulent convection in the coolant channel. The GOP²⁴ code, Battelle Northwest Laboratories, is the only one of the six that includes radiolytic, as well as thermally induced, reactions. The code considers multiple core regions, but diffusion is described in steady-state form and no provision is made for burnoff effects. In addition, the code does not appear to allow variable catalyst concentrations, and flat temperature profiles are assumed in the graphite. RICHTER,²⁵ developed at the United Kingdom Atomic Energy Research Establishment at Harwell, treats the following aspects of the problem: (1) reaction rates for the steam-graphite, carbon dioxide-graphite, and the water-gas shift reactions; (2) steady-state diffusion of the steam through the graphite; (3) inclusion of a bulk flow mechanism in addition to diffusion; and (4) variation of the steady-state diffusion parameters to simulate effects of burnoff. LIGHTER,²⁶ also a UK AERE code, provides for a transient solution in cylindrical geometry with diffusion as the sole mechanism for transport in the graphite. However, unlike RICHTER, attack of the graphite by CO_2 and the effects of the water-gas shift reaction are not included. TUBER²⁷ is a Dragon Project code for transient analyses. Some of

the noteworthy aspects of this code include the use of a two-radial-zone heat transfer model that divides the core thermal capacity between the bulk moderator and the coolant channel walls, the use of an Arrhenius-type chemical reaction model to integrate the corrosion rate over the entire core (with reactivity factors that can vary from coolant channel to coolant channel), and a bulk transport model to describe steam penetration into the core graphite, but with no provision for Fick's Law type of diffusion.

6.2.6 Fission-product behavior

Fission-product behavior is treated in detail in all reactors, but the physical characteristics of HTGRs impose some different considerations on this subject area. The fact that the PCPB is the first absolute barrier to fission products means that the primary system will always contain some fission-product activity. However, the release of fission products from the coated fuel particles and their transport, through the matrix and moderator graphites, to the coolant stream are complex processes which strongly affect the available inventory. It is, therefore, necessary to understand and account for these processes. The first consideration involving fission products is their release from the coated particles which is heavily dependent on particle behavior under irradiation. Models have been developed at ORNL,^{28,29} GAC,³⁰ and the DRAGON project³¹ in Europe to evaluate stresses in particle coatings. The ORNL model is incorporated in the computer program STRETCH;³² the GAC code is TRISO, and the DRAGON code is STRESS-2.^{33,34} It is also necessary to describe the actual escape of the fission products and their transport to the coolant stream. For diffusional transfer, this part of the problem is treated by FIPER^{35,36} in the U S. and also by a British code, FIPDIG.³⁷ The transport and plateout behavior of fission products in the primary circuit is treated by another GAC code, PAD.³⁸

Although more tritium is produced in HTGRs from incidental reactions than from fission, tritium may be regarded as a "fission product" with special characteristics. A portion of the tritium that is formed in solids or fuel may be released to the coolant. This tritium mingles with the hydrogen in the coolant, and any tritium formed there will, in part, chemisorb on graphite. In addition, it may leak with the coolant from the reactor vessel, be removed in the helium purification system, or permeate system walls into the steam generator, where as a result of removal (blowdown) or losses it is subject to release to the environment. This behavior is evaluated, under the assumption of steady conditions, by TRITGO,³⁹ a computer program developed at ORNL.

6.2.7 Containment pressurization

Two types of events which lead to pressurization of the containment structure are treated for HTGRs. The first of these is depressurization of the primary system which is treated by an apparently undocumented GAC code, PCRVDPEP. Results have been compared with a modified version of CONTEMPT.⁴⁰ The second event involves failure of the secondary, or steam, system leading to pressurization of the containment. Another modified version of CONTEMPT has been used to study this process.

6.2.8 Activity releases into and from containment

The ultimate consideration in any accident analysis is the release of radioactivity into areas where the result is exposure of personnel. A currently undocumented program, RDOSE, was developed by GAC to provide analysis capability in this area.

6.2.9 Seismic effects

Two core seismic computer programs currently under development at GAC are COSAM⁴¹ and CRUNCH.⁴² These codes can model a horizontal array (strip) of the core; a second COSAM version can model a full horizontal section of the core but is still in the early development stages. These programs are special purpose codes and were developed only for analytical prediction of dynamic response of the nuclear reactor core. Available general purpose structural codes (NASTRAN, etc.) can not be used for the following reasons:

- (1) The reactor core is nonlinear. It is composed of several thousand stacked graphite blocks in columns that will impact during a seismic disturbance.
- (2) Graphite characteristically responds to impact in a non-elastic fashion.

COSAM and CRUNCH employ sinusoidal or earthquake time-history boundary-forcing functions as input.

Basically COSAM and CRUNCH will provide the following design information: (1) forces on the boundary, (2) impact forces between graphite elements (magnitude and frequency), (3) relative velocities between adjacent elements and between the elements and the boundary, and (4) nature of response, i.e., significant rattling, degree of "lumping", etc.

NASTRAN⁴³ is a large digital computer program for static and dynamic structural analysis by the finite element approach. The program is intended primarily for large problems with hundreds or thousands of degrees of freedom. All mathematical subroutines are designed for efficient solution of large problems by taking maximum advantage of matrix sparsity and bandedness. A flexible executive system facilitates addition of new functional capabilities.

SHAKE⁴⁴ is 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. The capacity of SHAKE is 132 dynamic degrees of freedom for a 64K-word core storage.

DSAP⁴⁵ is 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 it must 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. The program can handle a maximum of 10 force tables and 25 modes. When the constraint option is exercised, the maximum rank of constraint coefficient arrays is limited to 50.

SOSDIT⁴⁶ is a program to study the soil-structure dynamic interaction effects on both the steady-state and time-history response of several arbitrary three-dimensional, linearly elastic structures under seismic excitation. The foundation medium is represented by a three-dimensional elastic half-space.

The structures, which are allowed to be either similar or dissimilar, are modeled as conventional discrete systems mounted on separate base slabs of close proximity. It is assumed that the stiffness of any structure, such as piping connecting the adjacent buildings, is negligible. The solution technique uses the normal mode method; for the steady state case, complex matrix inversion is involved while the time-history version includes Fourier synthesis. With a 64K-word

core storage, the only limitation of the program is that the product of the total number of degrees of freedom in a model and the number of structural modes of vibration considered cannot exceed 50,000.

6.3 ACCIDENT DIAGRAMS

One objective of the safety program is to achieve a thorough understanding of the processes involved in accidents as a basis for safety technology advancement. Typical relationships between accident processes and the program task groups that concern the associated technologies are illustrated in Figs. 6.1 through 6.3. These diagrams show event sequences representative of the most serious of potential HTGR accidents discussed in this section, along with certain conditions that are precursors of some events. The diagrams are not intended to be either comprehensive or consistent in level of detail; they do, however, show at least some of the correlations between every task group and HTGR safety. (Although task area 1 is not represented in these diagrams, it is included implicitly because of its very nature.)

POSTULATED ACCIDENT SEQUENCE

DPAWG 73-1265

Fig. 6.1. Relation of safety-program task groups to postulated accidents: local core overheating.

ORNL-DMG-73-126524

Fig. 6.2. Relation of safety-program task groups to postulated accidents: moisture leakage and loss of primary coolant.

Fig. 6.3. Relation of safety-program task groups to postulated accidents: PCRV depressurization and post-shutdown core cooling deficiencies.

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PART II

SAFETY PLANNING GUIDE

7. SYSTEMS AND SAFETY ANALYSIS (TASK AREA 1)

The preceding chapter indicates the wide variety of phenomena that influence HTGR safety and the need to perform comprehensive sets of accident analyses. Such analyses are central to an understanding of reactor safety; they also confer perspective to the diverse technical aspects of safety and have a role in reactor plant design for defense-in-depth.

The analysis of accident sequences and their consequences is the primary function of task area 1. In performing that work, information is obtained which helps provide guidance to work carried out in the other task areas.

7.1 OBJECTIVES OF TASK AREA 1

Essentially, accident analysis describes plant responses to postulated events or conditions. It may be done with more or less formality and sophistication, depending on the individual problem requirements, modeling capability, and extent of knowledge about the mechanisms involved. Often the process is iterative, beginning with fairly crude assessments of isolated effects and proceeding to rigorous studies that account for complex interactions. More advanced analyses formally describe plant transient behavior in successive increments, each beginning with a discrete event and following a logical event progression.

The primary goal of task area 1 is to provide a detailed study of potential accidents and their consequences. To this end, it is necessary to perform accident analyses that combine logical development of accident event sequences with computations of plant transients and fission-product behavior. Initially required are methodical procedures for identifying "generalized" events, i.e., events applicable to HTGRs in general, and for selecting credible combinations of events. Involved are identification of event combinations and the probabilities for the occurrence of these events, as well as estimates of associated consequences. Generalized mathematical models of HTGR component and process behavior are required to cover the ranges of interest. Because most of these models individually will represent only particular aspects of transient behavior, they must be applied in sets for overall analyses; practical methods for integrating the models, therefore, demand a significant portion of the development effort.

In addition to the plant and process descriptive information used directly in formulating analytical models, technology information developed in the other task areas is needed in a form suitable for task area 1 needs. This includes data for use in transient behavior analyses and information pertinent to the reliability and behavior of HTGR components.

The accident analysis work carried out in task area 1 also provides program guidance by evaluation of technological needs in the context of behavior response of components and processes and their influence on accident sequences and consequences. Initially, accident evaluations will be based on GAC calculations and will include critical reviews of analyses that have been performed to date. Where needed, assessments of, and calculations with, GAC transient response codes will be performed. As the program capability grows more comprehensive, analyses will be performed as required, including independent calculations.

Practical program guidance will be obtained during the process of performing accident analyses. For example, information will be obtained relative to the magnitude or frequency of failures or processes which can be tolerated. Transient response requirements of systems or components can be deduced from studies of the effects of changes in component characteristics on accident progression. The influence of behavior processes, such as fission-product transport and steam-graphite reactions on accident consequences help determine the need for that information. Additional direction to the program efforts evolves from sensitivity analyses performed for various accident routes.

To assimilate the large amount of information needed, an up-to-date record of the safety issues that have been studied and the status of each will be maintained. This will help focus program resources on needs and provide the background needed to extend previous work in response to new questions, problem insights, or plant design changes.

7.2 SCOPE

The above objectives require that task area 1 scope cover the entire spectrum of safety-related conditions and events. Initially, broad safety and topical reviews will be performed to help identify significant safety studies and analyses. Such reviews will serve generally to interface task area 1 with all of the safety and related technology areas.

Development of analytical capability to carry out pertinent plant transient-behavior studies is an important part of task area 1. Specifically included are development of: (1) methods for describing logically the accident event sequences; (2) models for HTGR plant behavior which combine thermodynamic, hydrodynamic, and controls transient effects; (3) HTGR core neutronic steady-state and transient response calculations; and (4) methods for assessment of accident consequences by probabilistic models. The analytical models will then be used to determine the overall system response to postulated accident sequences.

Task area 1 activities will also include studies to provide program guidance; these include performance of generalized safety-analysis case studies and the assembly and organization of information from all sources so as to obtain overall safety perspective. Again, the broad scope of interest extends over such diverse aspects as the assessment of accident processes that determine the release and transport of fission products, and the environmental and long-term operating conditions that affect component integrity. These program guidance studies will also provide some degree of general safety assessment.

7.3 STATUS AND NEEDS

There are broad program analytical requirements associated with HTGR safety technology. Development of the needed analytical capability builds to some extent upon existing models, with adoption of those models found acceptable; at the same time, it is expected that some new analysis models will need development. In particular, safety considerations not treated by previous analyses, or situations in which confirmation of prior results is sought via alternative methods, may require new analytical tools.

The existing analysis capability is largely that provided by the principal HTGR vendor in support of plant design and licensing. That capability is inferred in Sect. 6 and permits evaluation of plant responses to postulated major initiating events such as primary coolant depressurization, water inleakage to primary coolant, and interruption of core cooling as well as other less severe contingencies. An early need from the viewpoint of independent assessment capability is to review these analyses to gain initial insights for program steering purposes and to assess what supplementary analytical capability is needed to satisfy the program requirements. An in-depth review of the existing analyses should consider the following aspects:

- (1) Ident sequences. The existing choice of event sequences emphasized "design basis" accidents. There is need to evaluate the entire range of credible accidents to see if accidents other than the currently defined ones should be included.
- (2) Accident extent. The present analyses primarily consider the behavior when defined damage or safety limits are not exceeded as a result of particular event combinations. A broad safety perspective requires a further awareness of the probability, however low, of violating such limits and the associated consequences. Moreover, event progressions into "unsafe" domains must be considered in order to define the safety margins associated with inherent and engineered safeguards. This indicates the need to consider pursuit of some accident event sequences beyond the limits of prior studies.
- (3) Validity and depth of methods. The existing analyses embody primarily the vendor's interpretations of test data and other information in mathematical models of various accident processes or in design criteria. How well such models or criteria represent the processes is in many cases crucial to safety conclusions and, hence, a legitimate subject for independent verification. Such verification may initially require detailed reviews of analytical procedures, assumptions, calculation methods, and data; where anomalies are found, it may be desirable for the safety program to duplicate analyses independently with different models or perhaps seek confirmation through sensitivity or parametric studies.

There is a need for an independent evaluation of safety analyses. In this regard, maintaining up-to-date and logical documentation of safety analyses and of related topical studies should be performed to promote systematic comparisons of corresponding analyses from within and outside the program.

Some aspects of the analytical capability proposed for task area 1 are currently being developed within the AEC by the Directorate of Licensing. It is assumed that such capabilities will be available to this program and that appropriate coordination will serve to avoid duplication of effort.

7.4 TASK AREA EFFORT AND SCHEDULE

This area is divided into four task groups which address the following four general subjects: (1) component response evaluation in accident situations, (2) logical analysis of accident sequences, (3) physical-system modeling to facilitate analysis of specific accidents, and (4) quantitative analyses of defined accident sequences. Of these, task group 1.4 performs the essential safety analyses, while others provide input information and analytical tools.

Task group 1.1 provides information on component response behavior; this work is the basis for quantifying the analyses in task group 1.2. Therefore, the results must become available before the quantitative tasks in group 1.2 are undertaken.

The work in task group 1.2 (accident sequences) will ultimately become the basis for identifying the events that are subjected to detailed transient analysis. The first results of such analyses should appear early in the program to ensure that transient-analysis work is properly oriented.

Task group 1.3 involves developing the mathematical models required to carry out safety analyses. This work should receive early and substantial attention to ensure that the techniques employed in accident analyses will produce results that accurately reflect the transient behavior of an HTGR plant.

Task group 1.4 (safety analysis and evaluation) provides for the review and analysis of identified accident sequences. Several sequences have been established (by HTGR vendors and license applicants in preparing safety analysis reports), as being important to the demonstration of reactor plant safety. Initially, this review and analysis effort will depend heavily on previously established techniques and computer programs. Improved tools will be applied as they are developed. Another important aspect of the work in this task group is the provision of guidance for other task and task area efforts.

The proposed tasks, and the effort to be expended on them, are summarized in Table 7.1. This table is based on currently recognized needs. The logical progression of the effort in the time domain is shown graphically in Fig. 7.1.

Table 7.1. Summary of identified safety work in task area 1

Task description	Task priority	Costs (\$1000) and manpower (man-years in parenthesis) by fiscal year									10-year total	
		1975	1976	1977	1978	1979	1980	1981	1982	1983		1984
Task area 1—Systems and safety analysis												
Task group 1.1 Component qualification		60 (1.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	60 (1.0)	60 (1.0)	1020 (17.0)
1.1.1 Source tasks												
1.1.1.1 Component reliability experience	B											
1.1.1.2 Component design and duty qualification	B											
1.1.1.3 Component application in systems	B											
1.1.2 Synthesis task												
1.1.2.1 Component response evaluation	B											
Task group 1.1 total		60 (1.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	60 (1.0)	60 (1.0)	1020 (17.0)
Task group 1.2. Analysis of event sequences		60 (1.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	60 (1.0)	60 (1.0)	30 (0.5)	30 (0.5)	30 (0.5)	750 (12.5)
1.2.1 Source tasks												
1.2.1.1 Systems functional logic	A											
1.2.1.2 Accident-initiating situations	B											
1.2.1.3 Confidence analysis methods	C											
1.2.1.4 Common mode failure studies	A											
1.2.2 Synthesis task												
1.2.2.1 Accident logical sequence models	A											
Task group 1.2 total		60 (1.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	60 (1.0)	60 (1.0)	30 (0.5)	30 (0.5)	30 (0.5)	750 (12.5)
Task group 1.3. Modeling of plant transient behavior												
1.3.1 Source tasks												
1.3.1.1 Neutronic characteristics	A	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	90 (1.5)	60 (1.0)	1110 (18.5)
1.3.1.2 Component and subsystem characteristics	B	60 (1.0)	90 (1.5)	120 (2.0)	120 (2.0)	120 (2.0)	90 (1.5)	60 (1.0)	60 (1.0)	30 (0.5)	30 (0.5)	780 (13.0)

Table 7.1 (continued)

Task description	Task priority	Costs (\$1000) and manpower (man-years in parentheses) by fiscal year										10-year total
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
1.3.1.3 Physical and chemical process models	A	60 (1.0)	60 (1.0)	60 (1.0)	60 (1.0)	60 (1.0)	60 (1.0)	30 (0.5)	30 (0.5)	30 (0.5)	30 (0.5)	480 (8.0)
1.3.1.4 Consolidated plant models review	A	60 (1.0)	180 (3.0)	180 (3.0)	180 (3.0)	180 (3.0)	180 (3.0)	180 (3.0)	120 (2.0)	120 (2.0)	120 (2.0)	1500 (25.0)
1.3.2 Synthesis tasks												
1.3.2.1 Neutronic behavior models	B	Effort included in source tasks										
1.3.2.2 Component and subsystem behavior models	B											
1.3.2.3 Physical and chemical process models	A											
1.3.2.4 Consolidated plant models	B											
1.3.3 Validation tasks												
1.3.3.1 Neutronic transient models	B	Effort included in source tasks										
1.3.3.2 Component and subsystem transient behavior	B											
1.3.3.3 Physical and chemical process transient behavior	A											
1.3.3.4 Plant transient behavior	B											
Task group 1.3 total		300 (5.0)	450 (7.5)	480 (8.0)	480 (8.0)	480 (8.0)	450 (7.5)	390 (6.5)	330 (5.5)	270 (4.5)	240 (4.0)	3870 (64.5)
Task group 1.4 Safety analyses and evaluation												
1.4.1 Source task												
1.4.1.1 Review of available analyses and results	A	60 (1.0)	180 (3.0)	180 (3.0)	120 (2.0)	120 (2.0)	120 (2.0)	120 (2.0)	60 (1.0)	60 (1.0)	60 (1.0)	1080 (18.0)
1.4.1 Synthesis tasks												
1.4.2.1 Analyses and sensitivity studies	B	30 (0.5)	120 (2.0)	180 (3.0)	180 (3.0)	180 (3.0)	180 (3.0)	180 (3.0)	120 (2.0)	120 (2.0)	60 (1.0)	1350 (22.5)
1.4.2.2 Definition of additional needs	B	Included in task 1.4.1.1										
Task group 1.4 total		90 (1.5)	300 (5.0)	360 (6.0)	300 (5.0)	300 (5.0)	300 (5.0)	300 (5.0)	180 (3.0)	180 (3.0)	120 (2.0)	2430 (40.5)
Task area 1 total		510 (8.5)	990 (16.5)	1080 (18.0)	1020 (17.0)	1020 (17.0)	930 (15.5)	870 (14.5)	660 (11.0)	540 (9.0)	450 (7.5)	8070 (134.5)

Fig. 7.1. Work progression diagram for task area 1.

7.5 COMPONENT QUALIFICATION (TASK GROUP 1.1)

This task group is to collect and evaluate information concerning the prospects for adequate response by individual components under abnormal plant conditions, component reliability, and component failure modes, particularly those that could affect accident sequences in ways other than by simple loss of function.

7.5.1 Scope and justification

Most of the critical events in accident sequences are, or stem from, component malfunctions and failures. Thus, accident initiations may be characterized as failures of components in operating systems or as their inability to cope with externally imposed severe conditions. Later in an accident progression, the reliability of protection system components is challenged, possibly under unusual conditions generated by the accident itself. Beyond the mere fact of a component failure is its failure mode; this may range from simple loss of function to possibly unanticipated malfunctions representing safety challenges in themselves, such as reactivity additions.

A methodical analysis of accidents must, therefore, proceed from a thorough understanding of the serviceability of components with regard to accident conditions and of their potential to create or aggravate plant incidents. The probabilities of component survival or failure under all relevant conditions, to the extent they can be properly estimated, provide a viable basis for judging the relative likelihood of various accidents.

Initially emphasized by task 1.1.1.1 is the systematic gathering of experience observations which are directly or indirectly related to the operation of HTGR plant components and which concern their demonstrated capabilities. For some classes of conventional plant equipment, industry-wide experience may be represented by generic failure statistics, available from established data banks. Regarding novel components, in particular those developed specifically for HTGR service, records of qualification tests may disclose practical operating limits and other indications of failure modes or probabilities. Operation of test or demonstration HTGR plants and other gas-cooled reactors in Europe may also provide insights to potential problems in components.

Information regarding the component specifications and the actual designs is the focus of task 1.1.1.2. Particularly with regard to the novel HTGR equipment, such design information may delineate inherent limitations of components which can be related to probable failures when these limits are exceeded. Moreover, prior to qualification testing, the design information may be the only available basis for evaluating potential component serviceability. Information sources to be explored include the plant technical specifications, purchase specifications, descriptions of parts and materials typical of equipment classes, application data for commercially available equipment, and design features claimed by manufacturers in support of component qualifications for intended service.

Task 1.1.1.3 deals with the conditions encountered by components, mainly during the accident transients. This task is therefore dependent on results from task group 1.4. Tentative descriptions of accident conditions will be derived initially from prior analyses.

Formal organization and interpretation of the above information to evaluate component response or failure behavior is assigned to task 1.1.2.1. For each of the more vital components or subsystems involved in plant protection, a study is projected to provide an engineering basis for describing possible response and failure modes and for synthesizing the behavior of each with respect to the conditions that may be encountered.

7.5.2 Status

Much of the information called for in task 1.1.1.2 (i.e., equipment design basis, specifications, and actual design data) has been developed to various degrees by the reactor and equipment manufacturers. Although very little of this data has been published, significant portions are intended to be made available in support of plant license applications.

Regarding operating experience, the subject matter of task 1.1.1.1, some statistics are available for European gas-cooled reactor components which resemble counterpart U.S. equipment. Qualification test information on Fort St. Vrain plant major components is also available. Peach Bottom plant operating experience is partially applicable to large plants.

Safety analyses of the proposed large HTGR plants have developed some information of the type called for in task 1.1.1.3. The range of situations called for and the resolution required by task area 1 are much greater than represented by existing studies.

Substantial information input is available from completed or current work in the HTGR safety program and from other RDT programs (e.g., seismic, PCRV, pipe fatigue, and LMFBR).

7.5.3 Identified work

All four of the tasks in task group 1.1 will consist initially of reviewing current literature pertinent to GCR component experience, the vendor's safety analyses, and other HTGR transient analysis studies. The reviews will extend to new information as it is developed.

Source tasks 1.1.1.1, 1.1.1.2, and 1.1.1.3 develop the information basis for predicting whether, or to what extent, individual components and subsystems will function correctly under anticipated transient conditions. Task 1.1.2.1 assembles the diverse information from the source tasks and synthesizes the response behavior. Initial component evaluations will be made and later updated to reflect additional insight or modified to represent revised operating and transient conditions.

The basic information developed by task group 1.1 will contribute to the task area 1 safety analyses, reviews, and evaluations. Specific response probability estimates will be considered in system analyses. Timely component critiques may be issued where the need is apparent.

Information on the generic reliability experience with subsystems and major components of conventional classes is to be gathered in FY-1975. Data that are more specific to HTGRs will be obtained as it becomes available; that from the Peach Bottom plant will be reviewed during FY-1975, while information from the Fort St. Vrain and commercial plants will be gathered in succeeding years. Design and duty qualification data for safety-related components and systems is to be assembled in FY-1975, with emphasis on equipment in the HTGR primary loop. A preliminary assessment of the transient stresses imposed on HTGR components by upset and accident conditions is proposed for FY-1975; the assessment will be refined and updated as detailed accident descriptions are developed through analysis. Tentative evaluation of component and system probabilities for adequate response under accident conditions, or for

failure in modes so as to induce accidents or promote their progression, will be initiated in FY-1975 and completed the following year; thereafter the estimates will be upgraded as further information becomes available.

7.6 ANALYSIS OF EVENT SEQUENCES (TASK GROUP 1.2)

This task group is concerned with the description in logical terms of credible accident courses and the development of program capabilities to perform probabilistic analyses.

7.6.1 Scope and justification

Central to the assessment of reactor safety is consideration of the potential accident. Two general pieces of information are required to describe an accident satisfactorily; the logical ordering of discrete event sequences and the mathematical model of plant transient behavior. Further, the conditions that occur during a transient may influence subsequent equipment response due to component failure under unusual stress.

Task group 1.2 is concerned primarily with the ordering of accident sequences, while task group 1.4 is associated with determining transient system response behavior. Because the objective of this work is to understand the consequences that might occur under various accident sequences, this work is fundamental to safety analysis and planning.

7.6.2 Status

The methodologies of event tree diagramming and probabilistic analysis are well developed. They have been applied widely in aerospace projects and to a more limited extent in AEC-funded projects. Examples of the latter are the SNAP reactor and radioactive heat source systems and, currently, the LMFBR.

System probabilistic analysis error evaluation techniques have been reduced to practical computer codes which are available from Aerojet Nuclear to federal agency users. Such programs are useful primarily when a great deal of information on component responses behavior under various circumstances is available.

7.6.3 Identified work

Task group 1.2 orders accident sequence information so that the chronology, causes, and effects of the events in the sequence can be readily understood. The group also includes the acquisition of methods for calculating event probabilities and for incorporating probability distribution. Within the group, task 1.2.1.1 develops the ordering system; task 1.2.1.2 lists accident-initiating events, for example, forced shutdown accompanied by loss of power. Task 1.2.1.3 provides means for estimating variances in the probabilities of events in the accident sequences; task 1.2.1.4 addresses common-mode failures and their effects on accident sequences. Task 1.2.2.1 coordinates the information generated in the above subtasks for use by task group 1.4.

Some specific topics for early investigation under task group 1.2 are the following types of common-mode failures: (1) failures of moisture monitors in one or all loops, (2) time-dependent interruption of offsite and onsite electrical power, (3) loss of adjacent PCRV-liner cooling channels, and (4) loss of all of the main helium circulators. Common-mode failure events that could defeat system redundancy will be studied. In succeeding years, extension of this study to other critical areas is planned.

These investigations are to (1) identify common-mode failure causes, (2) predict occurrence probabilities, and (3) recommend, where necessary, further design studies aimed at reducing failure probabilities to satisfactory levels.

Event sequence diagrams representing the major classes of accidents postulated for HTGRs will be executed in FY-1975. These sequences will be expanded by sets of fault-tree diagrams to describe systems involvement at critical stages in each accident; these will be at the level of major systems functions and degradation states in FY-1975, and in greater detail subsequently. The common-mode failure analyses described earlier are for execution in FY-1975 and 1976. Adaptation of fault-tree probabilistic and error analysis computer programs for use in accident analyses is projected for FY-1976.

7.7 MODELING OF PLANT TRANSIENT BEHAVIOR (TASK GROUP 1.3)

Accurate description of the transient response of an HTGR system to upset conditions requires the availability of appropriate mathematical models. In principle, these models must be applicable to the entire spectrum of upsets, whether or not they may have potentially serious safety consequences. In practice, the upsets

of greatest interest are those with the most severe potential consequences, and these also provide the most severe tests of the models. The effort in this task group is directed toward the development of appropriately validated mathematical models and associated computer programs that will provide the required descriptions of plant transient response in safety-related situations.

The models will be applied in task group 1.4 to acquire definitive information on conditions that (1) stress components and, hence, affect their response modes or failure probabilities; (2) affect the accident course, e.g., PCRV pressurization due to steam-carbon reaction; and (3) control the fission product release and transport mechanisms. Accordingly, strong interaction between the logic and dynamic behavior models will be required to define the possible accident sequences.

7.7.1 Scope

Initial phases of this task group are the review of existing modeling capability and definition of the model requirements for accident analysis. The review work will be in conjunction with that undertaken in the other area 1 task groups and will emphasize model fundamentals as well as the parameter values embodied. Specific model requirements arise from the accident studies and must be defined with respect to the basic effects or effect combinations that govern the accident progression and the ranges of plant conditions over which the models must represent processes faithfully. Eventually, the accident logic developed in task group 1.2 provides important input in defining the model requirements.

Independent construction or review of models must proceed from basic knowledge of the plant and process characteristics. Accordingly, information is to be acquired concerning the operating characteristics of mechanical components, behavior-determining properties of the primary system, core neutronic behavior, control-process interactions, and characteristics of secondary processes such as the power generation and heat rejection systems.

The description of transient behavior in a system as complex as an HTGR nuclear electric station requires a number of models which may vary greatly in both detail and extent. Often models that treat a given process or phenomenon in great detail are limited to that one process and consider the remainder of the system only to the extent required to avoid invalidation of the results. On the other hand, the more "global" models usually avoid highly detailed treatments of the most complex processes.

In general, four types of models will be treated in this task group - neutronic models, component and subsystem models, physical and chemical process models, and complete or "global" system models. The neutronic models are aimed at descriptions of the full range of core transients and will include thermal effects as required. Another subject of interest in this area is the magnitude and time dependence of decay heat after reactor shutdown. Incorporation in comprehensive system models will not be a primary consideration in the development of these models.

Separate component and subsystem models will be developed for items of particular safety significance in the system. Potential candidates for such modeling include the steam generators, the helium circulators, and the core auxiliary cooling system. Although these models may be used for studies of the individual items or systems, they will more commonly be incorporated into more comprehensive system models. Highly specialized models, such as detailed models of instrument safety and control subsystems, will be developed in other task areas where appropriate capabilities exist to assess the dynamic interplay between instrumentation and reactor under accident conditions.

Several types of physical and chemical processes must be included in the modeling of HTGRs. The most prominent among these are the steam-graphite reaction and fuel and fission-product behavior. The development of steam-graphite reaction models is treated in task area 3 (Sect. 10), but the results of that effort will be factored into the modeling work in this task group. Modeling of fission-product behavior will proceed as a joint effort between task areas 1 and 2, while fuel-behavior models will be produced in task area 2. Where process modelling is assigned to other task areas, it is because such models consolidate and express the results of investigations in those areas and are best understood by specialists in the technologies involved.

The effort related to developing overall system models will be concentrated in task group 1.3. Ultimately, the models in this category should include as much of the plant as possible with as much detail in individual areas as is practicable.

7.7.2 Status and needs

A substantial effort has already been expended on the development of transient models by the principal vendor of HTGRs in the U.S. (GAC) and by other organizations interested in gas-cooled reactors, e.g., the DRAGON project in the United Kingdom. The discussion of some of the existing computer codes in Sect. 6 gives some

indication of the extent of that effort. In addition, GAC has published one report¹ describing their nuclear design methods, including nuclear transient codes, and another report² that describes some thermal and plant transient-analysis codes as part of a digital analysis of core cooling capability. Several reports describing other models and codes are also planned. However, most of the models and codes that have been developed especially for HTGRs are not thoroughly understood outside the organizations in which they evolved. Furthermore, many of the codes have not been critically compared against other codes with comparable capabilities or against other methods of analysis, and relatively few comparisons with real systems are available.

Detailed independent assessments of the vendor models and digital codes should be made, and comparisons of their results with those obtained through other approaches are highly desirable. Additional comparisons with experiments and operating systems should also be made. In some areas independent and diverse analysis capability should be developed, and as the technology of HTGRs is developed and further refined, it will be necessary to improve the models, codes and methods to take advantage of the available information.

The proposed work, to be described in the following subsections, is separated according to the four general classes of models that have been identified. Use is also made of the source-synthesis-validation concept to help define the logical progression of the effort. However, to avoid excessive proliferation of task numbers, individual tasks will frequently cover several different models of the same general class.

7.7.3 Identified work (source tasks)

The source tasks within this group will fulfill several common needs. First, they are intended to assemble and codify the information that is nominally available from HTGR vendors and license applicants regarding component, system, and material properties and transient characteristics. This information will be reviewed, evaluated, and in some cases independently verified prior to its actual use. Often a major fraction of this work will be done under other task areas and will simply be adapted to the specific needs of this task group. These tasks also provide for review and assessment of the models and calculational techniques that are used both to develop and to use the basic information. The reviews will help to better define the specific requirements imposed on the synthesis tasks. Again, information developed in other task areas will be adopted where possible. The work described below relates initially to data, models, and codes that are currently available. Since the

models and codes at least are likely to subject to continuing refinement over a substantial period, these review and assessment activities could be expected to continue for a comparable period.

Neutronic characteristics (Task 1.3.1.1)

The work in this task begins with the assembly and classification of the basic neutronic data and properties of HTGRs built and being built in the U.S. This information includes, but is not limited to, core material concentrations; flux, power, and temperature distributions; reactivity coefficients of various kinds; and control-rod and other material reactivity worths. Although most of this information currently exists, it is often not readily available in the form or in the detail required for accident-analysis studies by organizations other than the one that generated. Some of these data will be independently verified through the use of equivalent, established computational techniques and comparisons with available experimental results. Should any differences become apparent, they will be resolved; but this will not affect the progress of other tasks. Information of the type described above will be accumulated initially for the Fort St. Vrain and Summit reactor stations. Usable compilations for Fort St. Vrain are anticipated to be generally available by the middle of FY-1975, and the Summit station data will appear in similar form in the third quarter of FY-1975.

An important nuclear consideration in safety analyses is the release of decay energy from fission products. A comprehensive review of fission-product-afterheat measurements from thermal-neutron fission of ^{235}U was recently performed.³ This work should be extended to fuel mixtures of interest to HTGRs in an effort to improve the confidence in evaluations involving afterheat. Approximately 1/2 year of effort would be required to complete this activity.

This task also includes the review and assessment of nuclear kinetics models and codes that have been applied to HTGRs. A thorough understanding of the several items in this category is essential if meaningful safety analyses are to be conducted. Among the specific objectives of these reviews are the development of independent and diverse application capability, confirmation of model validity where appropriate, and identification of needs for further development and/or refinement. The initial work associated with this activity is expected to require 6 to 12 months. If a need is identified for the performance of substantially more complicated analyses (see also task 1.4.2.2.), such as transients associated with core or core-material rearrangement, additional models and codes will have to be reviewed and evaluated, requiring an additional year of effort.

Component and subsystem characteristics (Task 1.3.1.2)

The work in this task roughly parallels that in task 1.1.3.1. However, much of the basic data will be developed in task group 1.1 and simply used in this task. Additional compilations and reviews of transient response characteristics will be developed as required. Of particular interest are data on the thermal, hydraulic, and mechanical properties of components and subsystems over the entire range from normal to highly off-normal conditions.

Several models will be reviewed, assessed, and/or developed including those for steam generators, helium circulators, and other primary-system and BOP components. Much of the work being done for the AEC Directorate of Licensing with respect to core thermal transients under various flow conditions will be adopted. Steam generator modeling studies carried out under the GCFR program will be adopted and altered to represent an HTGR steam generator. The objectives of the model review aspects of this task are the same as for task 1.3.1.1. It is anticipated that the initial reviews will be completed by the fourth quarter of FY-1975.

Physical and chemical process models (Task 1.3.1.3)

Information regarding the transient effects associated with various physical and chemical processes will be compiled and adapted to system models through work on this task. The incorporation of results from the steam-graphite reaction modeling effort in task area 3 has already been mentioned. Data and modeling information will also be obtained from other task areas to facilitate description of fuel-particle failure processes and structural materials behavior. Work will be done in this task and in task area 2 on programs like FIPER-Q and PAD to establish the modeling and data requirements with respect to fission-product behavior. These efforts in other task areas provide the highly specialized information that must be embodied in the models. Since the reviews and assessments are to be performed in other task areas, the time schedules associated with them are treated there also (see Sect. 8).

Consolidated plant models (Task 1.3.1.4)

At the present time there appears to be only one comprehensive model to describe the transient behavior of an entire HTGR plant. This is the TAP program developed by GAC over a period of several years. A detailed review and assessment of this program will be made to provide a basis for deciding whether or not a federally funded effort should be launched to develop an independent calculational program

with comparable capabilities. If a comparable program is not developed, this study will establish the capability for understanding TAP analyses and may suggest refinement possibilities. Approximately 1 to 1-1/2 years would be required to carefully review and understand TAP.

7.7.4 Identified work (synthesis tasks)

The four synthesis tasks (1.3.2.1 through 1.3.2.4) treat the same four classes of analytical models described in the previous section. Since the efforts for these tasks are essentially parallel, the four tasks will be discussed collectively. The first step for each class of models will be the adoption or adaptation of existing models for "first-round" analyses of HTGR transients. This step, as well as the actual performance of such analyses in task group 1.4, is clearly one that has long since been taken by the principal HTGR vendor. It is, therefore, of interest only to new entrants in the field of HTGR safety analysis as a means of gaining familiarity and competence. The next step, which is of interest to all participants, involves the development of new models, as required, and the improvement of existing models. This updating and refinement of the models, as well as adaptation to new sets of conditions should continue for the duration of the safety program. Factors influencing such changes are: (1) the development of new basic information by other task areas, (2) improved definition of contingency conditions through accident analysis, (3) additional significant process interactions through model-based sensitivity studies, and (4) determination of significant disagreement between analysis predictions and observed test loop or plant behavior. The effort projected for the model-upgrading activities is substantial; however, the establishment of specific goals is dependent on the results of the source-task activities.

7.7.5 Identified work (validation tasks)

As with the other task classes in this group, four separate tasks (1.3.3.1 through 1.3.3.4) are proposed. However, relatively little effort will be devoted to task 1.3.3.3 because the validation of these highly specialized models will generally be undertaken in the task areas primarily responsible for their development.

The validation efforts will initially involve the comparison of calculated transient behavior with results obtained through the use of other models, codes, and methods with comparable capabilities. The transients used for such comparisons may be, but are not required to be, representative of potential accident situations in reactor systems. Initial comparisons of this kind will become available

in selected areas during the first year of effort. These will include fission-product release effects associated with flow aberrations in the core and core thermal transients following interruption of coolant circulation. Subsequently, where appropriate and practical, especially calculated results will also be compared with data from laboratory or scale-model tests and experiments such as critical experiments or in-pile tests. An important consideration in this regard is whether or not special tests should be performed to provide a basis for other comparisons. Since tests of this kind that adequately challenge the analysis capability are difficult, expensive, and time consuming (they may involve special experimental reactors, for example), it is currently assumed that they will not be performed.

Other opportunities to validate the transient analysis capability may develop from the operation of HTGR systems. Among the possibilities are planned tests during the startup and commissioning of new plants and any unplanned transients that are experienced. While none of these are likely to provide tests of the models in the regions of safety interest, they may permit comparisons in the near-steady-state regime which will help to establish confidence in the models.

7.8 SAFETY ANALYSES AND EVALUATION (TASK GROUP 1.4)

Task group 1.4 provides for the analysis of specific event sequences that are or may be significant from the standpoint of HTGR safety. The initial studies will examine currently identified events with the aid of available models and computer program. Eventually, however, results from all areas of the safety program will be factored into these analyses to ensure that the most accurate conclusions are reached. The purpose of these analyses is to determine the response of the reactor system to the various accident sequences under consideration; in performing that work, information on the safety features and margins of HTGRs can be obtained.

7.8.1 Scope and justification

The list of event sequences that may be examined in this task group includes all the accidents that have been or may be recognized to involve legitimate safety considerations in HTGRs. This does not imply that all accidents will be evaluated to the same level of detail or even that all postulated accidents will be treated. However, enough different accidents must be considered to demonstrate the applicability and adequacy of the safety technology that is developed. In addition, since these analyses will help to define

quantitatively the goals and requirements for other tasks throughout the safety program, it is important that the analyses that are performed include events which consider the important processes.

The first situations to be examined include events associated with the interruption of primary-system cooling, accidents involving depressurization of the primary system, and water-ingress accidents. These accidents will require consideration of a major fraction of the plant safety systems, as well as the most important safety-related physical and chemical processes. Additional accidents and combinations of events will be identified for study as the safety program progresses.

7.8.2 Status

A large number of postulated accident situations in HTGR systems have been investigated by the reactor vendor and the license applicants. These accidents cover the entire spectrum of readily identifiable events from minor disturbances to sequences that challenge the important engineered safeguards. To date, there is no evidence of unacceptable consequences from any of these studies. While event combinations that may be unacceptable can be conceived, these combinations require the assumption of coincident multiple failures which has been regarded as unrealistic.

In addition to the vendor-applicant studies, some limited studies have been undertaken in programs funded by the AEC Directorate of Licensing. To date these studies have been focused on the interruption of core cooling and the attendant thermal transients. This work includes the development of a computer program to describe the primary-system thermal behavior. Other studies for DOL have examined the fission-product transport code (FIPER-Q), various fuel-particle behavior codes, and fuel-failure processes. Substantial work is also in progress among European agencies interested in gas-cooled reactor concepts. All of this work will be of value for the safety analyses to be performed in this program.

7.8.3 Identified work

While all of the work in this task group is, in effect, application of the safety technology as it exists at any given time, specific source and synthesis tasks may be identified. The final category in this task group is an application task which provides for some management of the program effort.

Review of available analyses and results (Task 1.4.1.1)

This source task is initially of interest only to organizations that have not been heavily involved in the development of HTGRs; it should be recognized as a continuing, iterative effort. As additional analyses are performed and more highly refined results become available, they must be subjected to critical reviews to search for possible inadequacies or discrepancies. The results of such reviews provide the basis for the synthesis tasks which lead to the next round of results.

The initial independent reviews will examine the loss-of-cooling, depressurization, and steam-ingress accidents. The initial round of reviews will be made during the first year of activity under this task. Eventually, all of the event sequences with safety significance will be reviewed in detail.

A complementary feature of the review task involves the consolidation of results from various analyses of comparable events. Since organizations throughout the world are active in the area of gas-cooled reactor safety, it is to be expected that results of interest to this program will be developed elsewhere. Many of these will become available to AEC contractors through various international exchange agreements. These, as well as internally generated results, must be appropriately consolidated and catalogued for maximum utility. Although only a minor effort within task 1.4.1.1 is required for this work, a continuing effort will be devoted to maintaining a current "library" of accident analyses and results.

Analyses and sensitivity studies (Task 1.4.2.1)

This task, which encompasses the actual performance of accident analysis calculations, represents the major effort in this task group. By its nature, it is very closely coupled to the tasks that generate the analysis programs and to the tasks that define the accident sequences to be evaluated. Analyses will be performed for a wide variety of events which may consider only portions of a system or an entire reactor plant. Sequences to be studied initially again include the interruption-of-core-cooling, depressurization, and water-ingress cases.

In addition to specific accident-sequence analyses, sensitivity studies will be performed to help elucidate the effects of various parameters. Results from such calculations will be of value in defining accuracy needs for data and special modeling and computational requirements.

Effort on this task will continue for the duration of the safety program to provide for testing new codes or methods as they evolve and to examine any new event sequences that may be identified.

Definition of additional needs (Task 1.4.2.2)

By formalizing the establishment of specific needs for data and/or analytical capability, this task assures that requirements are clearly defined and appropriately justified. The two synthesis tasks described here represent the other side of the iterative process mentioned in connection with task 1.4.1.1 between development of analytical capability and performance of specific analyses. The needs identified here will be translated into results through activities in task group 1.3 and task 1.4.2.1.

REFERENCES FOR SECTION 7

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3. A. M. Perry, F. C. Maienschein, and D. R. Vondy, *Fission-Product Afterheat - A Review of Experiments Pertinent to the Thermal-Neutron Fission of ^{235}U* , USAEC Report ORNL-TM-4197, Oak Ridge National Laboratory, October 1973.

8. SAFETY RESEARCH AND DEVELOPMENT

The primary goal of all the safety work discussed in this planning guide is closely related to the objectives of the tasks identified in the previous section, that is, the performance of systems and safety analyses to demonstrate as quantitatively as possible the safety characteristics and safety margins of HTGRs under accident conditions. Such studies require the availability of large amounts of technological data in many areas ranging from the fundamental properties of all materials in the system to detailed behavior characteristics of components under conditions of interest. Data may be required for the entire spectrum of plant situations from shutdown, through normal operation, to faulted conditions.

Much of the technology required for detailed safety studies is also applicable to other aspects of the development, design, construction, and operation of HTGRs. Consequently, it has been under development for a number of years by GAC and by a variety of programs funded by the U.S. and numerous foreign governments. It is anticipated that these development efforts will continue and that additional safety-related technical information will become available. In conjunction with these efforts, and partly because of them, work can be identified within every area of HTGR technology development that could reasonably be performed under a program directed explicitly toward the safety aspects of HTGRs. Many of the tasks that are thus identifiable deal with specific technical information that must be fully developed to permit the detailed analysis of postulated high-risk events which could otherwise be treated by the use of design conservatism or by highly conservative, but less precise, analysis methods. However, a significant amount of work is also concerned with maintaining surveillance of developmental activities to ensure that all applicable safety-related technology is critically reviewed and adapted to the safety studies as it becomes available and to ensure that any needs for additional technical information are recognized and acted upon.

The ensuing portions of this section discuss currently identifiable safety work associated with both the development and surveillance activities within each of the six technology-oriented task areas treated in this planning guide. Each task or group of tasks is presented in the context of current status of the technology, perceived needs, and explicit safety justification of the work that is identified. Because this work is closely related to and dependent on technology developments within other HTGR programs, these tasks are also treated in Part III of this document, which

deals with all of the safety and safety-related work that can be identified with HTGR development. It is expected that, as more understanding is obtained relative to HTGR safety, tasks assigned to this safety guide will change.

8.1 FISSION PRODUCT TECHNOLOGY (TASK AREA 2)

8.1.1 Definition and objectives of task area

The potential radiological hazard from fission products depends upon the barriers separating them from the environment and on their mobility. Other reasons for studying fission-product distribution relate to their known catalysis of reactions between oxidants and carbon and their potential effects on structural metals. To predict and evaluate the effects related to fission products, it is therefore necessary to be able to describe their distribution and mobility under all circumstances. The goal of the safety work discussed in this section is to provide this capability for conditions that may prevail during postulated accidents. Because the fission-product distribution during normal operation constitutes a description of conditions at the beginning of an accident, its description is also needed, but this is expected to become available from efforts in the safety-related engineering and technology program.

The objectives of task area 2 with regard to safety include the following:

- (1) Definition of explicit needs for fission-product technology in all areas important to safety,
- (2) Determination of the transport characteristics of core materials, including fission products, fuel, and control materials, under conditions that are associated with postulated major accidents, and
- (3) Provision of data to form bases for functional specifications that may be needed for engineered safeguards to mitigate the consequences of major accidents.

8.1.2 Scope

The objectives listed above define the scope of task area 2 in general terms. Because one of the objectives is the assessment of safety technology needs, the scope of that effort is necessarily somewhat broader than that of specific investigations to develop

technical data. However, the current state of HTGR development permits the scope of future work to be limited in terms of both materials and parameter ranges. As knowledge of HTGR accident sequences and their probable frequencies becomes better defined, there will probably be further adjustments in task scope and direction. Currently the following factors serve to limit the scope of this task area with respect to need assessment and technical investigations, as appropriate:

- (1) For various reasons — including short half-life, low energy, deposition rate, lack of tendency to concentrate in man, non-volatility in the presence of graphite, compatibility with structural components, and low fission yield — the list of important nuclides can be reduced to a few isotopes of tellurium, iodine, krypton, xenon, silver, cesium, strontium, barium, and tritium. Zirconium behavior is usually studied as characteristic of a nonvolatile and immobile element.
- (2) Carbide and oxide fuels with carbon and SiC layered coatings need to be included to reflect the choice of reference fuels.
- (3) Fission-product transport studies should be confined to reference materials and the Fort St. Vrain matrix material and should be expanded as further developments are made with regard to fuel-rod matrices.
- (4) Structural graphites used in fission-product transport studies should be those specified or strongly considered for use in HTGRs, except where elucidation of mechanisms requires investigation of a wider range of properties.
- (5) Primary circuit metals used in fission-product deposition studies should initially be those used in HTGRs built to date or specified for the large HTGR for metallurgical reasons (see Sect. 13). Because surface film compositions depend to some extent upon the coolant composition, investigations in which surface films play an important part will need to be wide ranging (see also Sect. 10).
- (6) All investigations involving fuel should initially be carried to at least 1600°C, graphite studies to 1400°C, and metal surface studies to 1000°C unless relevant phenomena cease to occur below these temperatures.
- (7) Accident analyses performed by the vendors and others permit a definition of the conditions likely to be achieved in the current reference set of accidents discussed in Sect. 6.

These analyses should be used to select initial conditions for the investigation of abnormal fission product transport.

Accident sequence analyses to be performed under task group 1.2 (Sect. 7) may create a need for fission-product transport data under as yet unspecified conditions. In this situation, a scoping study would be performed to determine whether the potential risk from the new situation warrants further study.

- (8) The development of improved fuels by addition of materials to control fission-product release, oxygen potential, or carbon transport will not be performed under the safety program. When improved fuels reach reference fuel status, they should be included in the HTGR safety program.
- (9) Information on fission-product transport and fuel behavior will be supplied to task group 1.4 for use in accident analyses, to task group 1.2 for use in accident sequence studies, and to task area 3 for assessment of catalytic and metallurgical effects.

8.1.3 Status and needs

Detailed discussions of the current state of the technology and specific needs are provided for each task group in Sect. 9. The general status of the entire area is presented here only to provide perspective for the more comprehensive treatments.

Owing to the long development history of gas-cooled reactors, both in the U.S. and Europe, large amounts of information have been accumulated with respect to the behavior of fission products. This body of information is generally agreed to be adequate for the design, construction, and operation of large HTGRs if conservative design decisions are made. However, the available information is recognized as incomplete in many areas, so that appropriately conservative assumptions must be applied in fission-product transport analyses, as well as in the design of HTGR. Consequently, the safety margins that appear to exist in all situations involving HTGRs, although conservative, can not yet be unambiguously defined.

Accordingly there are three principal categories of need regarding the behavior of fission products. The first is for a comprehensive assessment and description of the available technology to ensure its wide availability and at the same time to help define areas of more specific needs. The second is the advancement of the technology to enhance confidence in any conclusions about the safety-related behavior of fission products. The third is for improved understanding of the failure mechanisms of coated fuel particles.

8.1.4 Summary of task groups, effort and schedule

Task summary table and task area organization

The tasks in task area 2 have been collected into groups based upon system components.

These groups are entitled:

2.0 Task area need assessment

2.1 Fuel and control material transport

2.2 Fission product behavior in the coolant — engineering-scale dynamics

2.3 Fission product behavior in the reactor containment.

With the exception of task group 2.1, each group is subdivided into source, synthesis, and validation or surveillance tasks.

The source tasks provide for elucidation of mechanisms and determination of parameters. Synthesis tasks involve preparation of mathematical models and ultimately computer codes. Validation tasks test the models and data packages produced in realistic surroundings; they are the largest blocks of work because confirmation of time-dependent phenomena over a four- to six-year time span through post-irradiation fuel-element handling requires an extensive effort. A tabular summary of the tasks selected for inclusion in the safety planning guide is provided in Table 8.1.

Primary emphasis is on assessment of the fission-product technology needs as related to HTGR safety to preserve consistent objectives, standards, and control. This implies that, although priority will be given to safety-related topics, other reasons for studying fission-product distribution will be considered in formulating goals for research and development.

The tasks selected in this safety planning guide are primarily associated with obtaining information on fission-product transfer phenomena during depressurization and cooling interruption accidents.

Costs and manpower

The estimated effort associated with this work is shown in Table 8.1. As shown, the need assessment task continues for the life of

Table 8.1. Summary of identified safety work in task area 2^a

Task description	Task priority	Estimated costs (\$1000) and manpower (man-years in parentheses) by fiscal year										
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	10-year total
Task area 2 - Fission product technology												
Task group 2.0 Need assessment for fission product technology.												
2.0.1 Goals for fission product technology. Establishment of program needs for fission-product retention based on accident considerations. Updating as required. Coordination of reporting of results by element.	A	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	1800 (30)
Task group 2.0 total costs		180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	180 (3)	1800 (30)
Task group 2.1 Fuel and control material migration												
2.1.1 Source tasks. Quantitative description of mechanisms by which control, fissile, and fertile materials might migrate under postulated accident conditions.												
2.1.1.1 Actinide adsorption and diffusion in graphites	C			60 (1)	60 (1)							120 (2)
2.1.1.2 Control material adsorption and diffusion in graphite	A		60 (1)	b								b
2.1.2 Synthesis tasks. Codes for analysis of movement of Th, B, and other control materials during accidents involving very high temperatures or steam and high temperatures.	b		b									b
Task group 2.1 total operating costs			60 (1)	60 (1)	60 (1)							180
Task group 2.2 Fission product behavior in the coolant circuit engineering-scale dynamics												
2.2.1 Source task												
2.2.1.1 Fission product distribution and re-entrainment. Description of aerodynamics of aerosols in HTGR with emphasis on deposition and re-entrainment in realistically complex geometries. Studies of aerosol adhesion to appropriate substrates as functions of material temperature and time.	B	120 (2)	120 (2)	120 (2)	150 (2.5)	120 (2)	60 (1)					690 (11.5)

Table 8.1 (continued)

Task description	Task priority	Estimated costs (\$1000) and manpower (man-years in parentheses) by fiscal year										10-year total
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
2.2.2 Synthesis task												
2.2.2.1 Fission product distribution code for dusty coolant circuit. Modification of PAD code to take account of role of dust in fission product distribution in primary circuit and formulation of dust re-entrainment model for accident analysis.	A		60 (1)	30 (0.5)	30 (0.5)							120 (2)
2.2.3 Validation tasks												
2.2.3.1 Re-entrainment experiments. Experimental determination of the mobility of solids adhering to surfaces of the primary loop Peach Bottom HTGR. Correlation with need and expectation.	A		160 (2.5)	60 (1)								220 (3.5)
2.2.3.2 Fission product-dust distributions in HTGRs. Measurements of distributions of deposited fission products on steam generators, ducting, and circulators in or from Fort St. Vrain or larger HTGRs. Development of appropriate remote measurement techniques.	a											
Task group 2.2 total operating costs		120 (2)	340 (5.5)	210 (3.5)	180 (3)	120 (2)	60 (1)					1030 (17)
Task group 2.3 Fission products in the containment												
2.3.1 Source task												
2.3.1.1 Review of iodine behavior in containment. Review of work done under other programs and determination of need for further refinement of description of iodine behavior in HTGR containment, including concrete.	C	30 (0.5)										30 (0.5)

Table 8.1 (continued)

Task description	Task priority	Estimated costs (\$1000) and manpower (man-years in parentheses) by fiscal year										10-year total
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
2.3.2 Synthesis task												
2.3.2.1 Code for description of iodine behavior in HTGR containment. Recommend or develop code to describe iodine behavior in containment.	D											
Task group 2.3 total operating costs		30 (0.5)										30 (0.5)
Task area 2 total operating costs		330 (5.5)	580 (9.5)	450 (7.5)	420 (7.0)	300 (5)	240 (4)	180 (3)	180 (3)	180 (3)	180 (3)	3040 (50.5)
Equipment costs		50	275	160								425
Task area 2 total costs		380	55	550	420	300	240	180	180	180	180	3465

^aAdditional tasks may be assigned to this task area as information becomes available.

^bExpenditure and/or priority depends on the result of the need assessment study.

the program. The effort currently assigned to task group 2.1 (fuel and control material transport) is only intended to cover a scoping investigation following demonstration of need. Task group 2.3 (fission product behavior in the reactor containment) may not require more than a review and "translation" of data obtained under the water reactor safety program.

A cost has not been assigned to task 2.2.3.2 (fission product and dust distributions in HTGR) because the effort required to obtain information on fission product distribution, as distinct from inventory, has not been shown to be justified.

Schedule

General. Table 8.1 may be read as a time chart. However, the program schedule is influenced by external events and interim results. External events also dictate the scheduling of validation tasks involving experiments in power reactors. A mistimed effort can lose what may be unique information. For example, information from the decommissioning of Peach Bottom is only available between 1974 and 1978 (task 2.2.3.1).

In general, surveillance programs aimed at validating normal fission-product release from the core would be expected to run until the core composition is relatively constant, i.e., for at least six years for Fort St. Vrain and four years for the first large reactor. Surveillance is recommended in subsequent plants if fuel or graphite temperatures increase more than 100°C. Thus, these programs should extend beyond 1984.

Monitoring for longer term effects, such as consequences of changes in plateout factors, should continue for the life of the plant or until there is resolution of significant uncertainties.

Program priorities and milestones. The highest priority tasks within the scope of the safety planning guide are quantitative assessments of the status of information about fission product distribution and of the mobility of core materials at high temperatures. A review of the whole area and the production of a reference data file for use by the HTGR industry are therefore the No. 1 and No. 2 program milestones. Following this general review of task area 2 technology, specific technology needs will be assessed, covering the behavior of each reference nuclide under normal and accident conditions in an HTGR.

Performance of a need analysis is conditional on the formulation of a complete and consistent set of criteria for fission product control. This activity will require industry and AEC participation.

Need analyses would be followed by scoping studies to define budgets and work plans where this had not already been done. The highest priority specific need assessment involves a determination of what work is needed (if any) to further define the behavior of core materials at high temperatures.

Priorities for currently identified work are given in Table 8.1.

Use of existing facilities

A number of existing facilities will be used to implement the proposed program. Their utilization in the HTGR program will improve return on investment in these facilities and substantially reduce the program's capital equipment requirements.

In particular, advantage should be taken of the decommissioning of the Peach Bottom HTGR to examine fuel and components. This exercise will conclude a long period of experimental work at Peach Bottom. The Fort St. Vrain reactor and other large HTGRs should be used in validation studies to the maximum practicable extent.

8.1.5 Need assessment (Task group 2.0)

Justification and scope

This task group objective is to ensure that specific safety needs in the area of fission product technology are identified and that any required technology is developed in a timely manner. The objectives and scope of this task group are those of the whole task area as defined in Sect. 8.1.1 and 8.1.2. In addition, this task group should ensure that the needs of task area 1 are met and that information prescribing fission-product behavior such as temperatures, flows, material descriptions, and accident sequences, provided by other tasks, are factored into continuously reviewed assessments of the need for further work.

Status

The paths that fission products can take to the coolant circuit and the environment are shown in Fig. 8.1. Each component or capacity (for example, the coolant circuit walls that can absorb certain fission products) is identified by a reference number. This permits transfers, designated as τ , between capacities to be identified by two indices. A third index, omitted in Fig. 8.1 for convenience, is needed to define the isotope being transferred.

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Fig. 8.1. HTGR system components controlling fission-product transport to environment.

Thus $\tau(I, J, K)$ describes the rate of transfer of isotope I from J component to K component per unit amount of I in J. The transfer functions depend on temperature, medium, and concentration.

One way to identify fission-product behavior needs is to calculate fission product transport based on conservative estimates of transfer functions and to compare the result thus obtained with established goals. In many cases there is some latitude relative to the permissible values of specific transfer functions to meet an overall objective of fission-product containment. This information will facilitate practical evaluation of fission product needs. Engineering solutions are often adopted in preference to data refinement programs because their implementation is faster. Incentive for data refinement may, however, still exist.

Sets of fuel specifications based upon radiological and economic criteria have been published by the Dragon Project^{1,2} and the UKAEA.³ Uncertainties in data and accident sequences were recognized. Permissible coolant circuit inventories can also be inferred from accident analyses presented in the Summit Station PSAR.⁴ No one set of analyses can, however, be regarded as exhaustive or final. Guides to help establish goals for fission-product releases from fuels are being developed by ORNL.

Identified work

Goals for fission product release from fuels should be estimated for ^{88}Kr , ^{131}I , ^{137}Cs , ^{90}Sr , and other nuclides for normal and accidental release. The Summit site should be used initially as a "reference site." Calculations using appropriate codes should be performed to help define the following:

- (1) goals for circulating and plated-out inventories in the coolant circuit,
- (2) goals for release rates from the core — $\tau(4,5)$,
- (3) goals for release characteristics of the fuel — $\tau(1,2)$,
- (4) goals for fuel failure criteria which determine $\tau(1,3)$.

These goals should be updated as necessary. A state-of-the-art review, and a data file describing fission-product transport mechanisms, should be produced.

8.1.6 Fuel and control material transport (Task group 2.1)

Justification and scope

The justification for the work in this task group is the desire for further knowledge of the behavior of core materials under Class IX accident conditions, in which sustained temperatures above normal are postulated. This task group will develop descriptions of the transport of uranium, thorium, and control materials in a reference core at temperatures in excess of 3000°C. Task area 1 will perform the necessary thermal and nuclear analyses.

Status

An analysis of boron and uranium behavior in graphite at high temperatures was made during the licensing of Fort St. Vrain. The applicability of that analysis to postulated Class IX accidents in other HTGRs has not been established.

Identified work

Actinide and control material transport in graphite (tasks 2.1.1.1 and 2.1.1.2). Pertinent data should be assessed to determine the effects of very high temperatures on the relative distribution of uranium, thorium, boron, and perhaps important heat-producing nuclides such as protactinium; and the need for further information should be evaluated. This evaluation should include a suitable overall analysis (possibly in task area 1) of a Class IX accident in the context of an existing or proposed HTGR. Since further effort depends on the results of the scoping study and need assessment, costs beyond the first year have not been estimated.

8.1.7 Fission products in the coolant circuit - engineering-scale dynamics (Task group 2.2)

Justification and scope

Reference to Fig. 8.1 shows that the coolant circuit occupies a key position in the path of fission products from the fuel to the environment. It is the repository of almost all fission products leaving the core and the most available source of fission products, should the primary helium containment be breached. Description of the radiological consequences of postulated accidents in which primary containment is breached therefore requires either assurance that displacement of 100% of this deposited primary circuit inventory is acceptable or means of calculating actual displacement of iodine, tellurium, cesium, and strontium isotopes.

The conditions of interest in this task group are (1) changes in the direction and/or magnitude of the coolant flow, (2) changes in coolant composition from helium to helium plus steam, and (3) rapid changes in coolant pressure.

Under normal operating conditions, the principal objective is determination of the deposition ratio,* for reasons discussed in Sect. 8.1.8 (task group 2.3). However, in a given accident both the amount of activity displaced and the potential dose to reactor maintenance workers will depend on the normal distribution of gamma-emitters such as ^{137}Cs and ^{131}I as well as on the circuit inventory. It is therefore necessary to investigate cesium and iodine distribution if significant departures from uniform distributions are encountered. The scope of task group 2.2 therefore includes the description of fission-product liftoff or re-entrainment under accident conditions and fission-product distribution under normal conditions. Close integration will have to be maintained between this task group and related studies dealing with the deposition of fission products.

Status

The dynamics of fission-product deposition and re-entrainment on an atomic or molecular level depend strongly upon the molecular weight of the fission-product carrier and whether contact with loop surfaces leads to reversible or irreversible sorption. Mathematical models based upon analogies between heat and mass transfer exist for predicting distributions of molecular species in thermally and geometrically complex systems. Loop experiments performed at Battelle Memorial Institute,⁵ GAC, Harwell, ORNL, and KFA Jülich, and reactor experiments associated with Dragon, AVR, and other gas-cooled reactors have been helpful in validating this approach.

In the situation where a fission product is carried by a relatively heavy dust particle, both deposition and re-entrainment are difficult to describe and validated models are lacking. The need for further data when secondary containment is provided remains to be determined.

* Deposition ratio (I) = $\frac{(\text{Curies of I on surfaces})}{(\text{Curies of I in gas})}$.

Characterization of HTGR dust is in progress, and attempts are being made to modify the fission-product distribution code PAD to represent distributions measured in Peach Bottom where dust plays a significant role. The PAD code,⁶ which has been recently developed at GAC, based on earlier work at ORNL and Battelle Memorial Institute, is used for describing the distribution of fission products in the HTGR primary coolant loop. It takes into account gaseous precursors and does not require the assumption of an infinite sink of fission-product deposition. It does not consider diffusion into circuit walls and may therefore misrepresent behavior at higher temperatures. The code has been used to compare predicted and measured activity distributions in the Dragon coolant circuit and in laboratory experiments. More refined quantitative comparisons need to be made. A code that considers diffusion into the duct wall is being developed at KFA Jülich to assist in gas-turbine accessibility studies.

Identified work

Fission product distribution and re-entrainment (tasks 2.2.1.1, 2.2.2.1, 2.2.3.1, and 2.2.3.2). In experimental studies in this area, difficult problems are associated with guaranteeing the correctness of chemical species, of surface composition, and of geometry. Extrapolation of scale is also a problem for dusty systems where theory is not adequate. Because of this, experimental studies are preferred that involve in-pile loops or experiments like VAMPYRE at KFA Jülich in which an HTGR is used as a source of hot helium-containing dust, fission products, and impurities. Measurements of fission-product distributions in HTGRs should therefore be made when opportunities occur.

Fission-product distributions should be measured in all parts of the Peach Bottom primary coolant circuit during decommissioning, and opportunities should be sought to measure distribution as well as deposition ratios in Fort St. Vrain. Since the Fort St. Vrain steam generators were not designed with this in mind, formidable experimental problems may be encountered.

Surveillance of reactor operation is not likely to yield definitive information about re-entrainment under abnormal conditions. For this reason, task 2.2.1.1 emphasizes dusty systems for source studies, although use of reactors to prepare specimens for re-entrainment experiments would be desirable. The decommissioning of Peach Bottom will yield an ample supply of such specimens. A large-scale loop (not necessarily in-pile) may be necessary if engineered safeguards to prevent higher-than-normal velocities

during a depressurization are not deemed adequate or if low re-entrainment fractions have to be validated. Such a loop would permit simultaneous investigation of both distribution and re-entrainment models.

Integral experiments should be complemented by studies of dust adhesion as a function of material and thermal history (task 2.2.1.1) and by adsorption, solubility, and diffusion studies.

Improvements should be made to the PAD code to allow for the presence of dust and to take into account the diffusion of fission products into metallic walls at higher temperatures. An improved mathematical model for re-entrainment may also be required to reflect differences in dust deposition conditions and thermal history around the primary circuit.

8.1.8 Fission products in the secondary containment (Task group 2.3)

Justification and scope

This task group will provide information relevant to the design of air-cleaning systems for the secondary containment.

The containment resembles the primary coolant circuit in that the gas-borne concentration is determined by transfer between purification systems $\tau(10,11)$ and $\tau(13,16)$, deposition of various species on walls $\tau(10,12)$, and leakage $\tau(10,16)$.

The principal concern for HTGRs in this task group is believed to be the rate at which iodine can react in the containment to produce compounds that are hard to filter or aerosols that are easy to filter. The effective molecular weight and chemical form of iodine discharged determines its deposition velocity in the environment and hence its effect on man.

Status

The behavior of fission products in water-cooled reactor secondary containments is being evaluated by Battelle Pacific Northwest Laboratories (PNL).

The experiments and mathematical models being reviewed cover wet and dry systems containing air and helium. However, the concentrations used are much higher than those to be expected during HTGR accidents. A specific evaluation of how iodine might be expected to behave in an HTGR containment is also being made separately by PNL to determine the need for further work.

Some preliminary work has been completed on krypton, xenon, and iodine delay in packed beds of granulated concrete. Also, studies of iodine-vapor and carbon-dust transport through concrete specimens provided with a crack have been carried out. The preliminary, unpublished data confirmed that xenon and krypton do not sorb and are not measurably delayed on concrete. Iodine and dust transport, however, is attenuated quite markedly by retention in concrete; attenuation of CH_3I has not been demonstrated. Evaluations of the utility of these observations for computing attenuation through all likely PCRV liner and body leak paths are being made.

Identified work

Evaluation of need for further work (tasks 2.3.1.1 and 2.3.2.1).

The review performed by PNL should be used to determine whether useful safety-related gains are to be made from further study. The attenuation of CH_3I by reactions with water in concrete should be determined.

8.2 PRIMARY COOLANT TECHNOLOGY (TASK AREA 3)

This task area is concerned with the effects of chemical reactions that can occur between impurities in the primary coolant and the coolant-circuit materials, including the core graphite, fuel, and structural materials. Particular emphasis is given to considerations of steam ingress into the primary coolant system, because this condition results in the greatest impurity concentrations.

Both short- and long-term effects are involved. The former are associated with sudden, relatively large inleakage events, whereas long-term effects derive from continuous, ostensibly minor leakages that occur over long periods of time.

8.2.1 Area objectives

One of the objectives of this task area is to identify those aspects involving primary coolant system chemical technology which have the potential for initiating or contributing to the sequences of events that ultimately lead to the dispersal of radiological species into the biosphere. However, the primary objective is to assess the adequacy of extant technology both to describe the phenomena involved and to provide safeguards as might be required to avoid such biological insults.

8.2.2 Scope

The scope of task area 3 is generally limited to considerations of the effects of chemical attack on fuel and the graphite and metallic components of the reactor as a result of the introduction of impurities into the primary coolant circuit. Although the causes for the initial introduction of contaminants are normally beyond the scope of this task area, propagation of sources for further contaminant ingress is, of course, not excluded as a possible course of the sequence of events.

In like manner, consideration is given to fission product transport in this task area only to the extent that such transport accelerates or intensifies the effects of impurity introduction (e.g., catalysis of the steam-graphite reaction by fission-product species). However, consideration is obviously given to the effects of impurities in the coolant system as they determine source-term data or contribute to the violation of fission product barriers of cleanup systems.

8.2.3 Status and needs

The impurities of major concern in this task area are, or result from, the introduction of oxidizing species into the coolant circuit, and there are five modes for the introduction of these oxidizing species. In four of these, the oxidant is steam or water converted into steam: (1) leakage from a main steam generator tube, (2) failure of a main helium circulator water bearing and seals, (3) rupture in an auxiliary heat exchanger (or leakage at low pressure conditions), and (4) leakage from a liner cooling pipe. In the fifth mode the oxidant is air; however, this mode requires depressurization of the primary circuit.

Another mode for the accidental oxidation of a graphite fuel element, but one that does not involve the HTGR core, is the unintentional exposure of a fuel element to air while the element is located within a spent fuel vault.

The oxidation of graphite by air is exothermic and is much more rapid than corrosion by steam under similar conditions. As a result, in those cases of interest, the rate of oxidation of the fuel elements (and hence the rate of heat release) will most likely be determined by the rate at which air is brought into contact with the elements rather than by the chemical kinetics.

Corrosion by steam, on the other hand, is endothermic; so the reaction cannot be self-sustaining. But the potential for steam ingress is much greater than for contact of the fuel elements by air. In addition, the effects of steam ingress into the coolant circuit are more numerous and require more complex analysis. This is due in part to the flammability of the major reaction products, hydrogen and carbon monoxide, and the potential for carbon transport by a cyclic mechanism involving the CO-CO_2 equilibrium. This last circumstance may produce undesirable effects on structural metals.

Two types of steam leakage are possible. One of these — continuous leakage at operating conditions — can only result from defects in the steam generators. The effects of such leakage, if sufficiently small (and carefully monitored), are of concern because of economic factors, such as the frequency with which components of the core or cleanup system must be replaced or regenerated. However, leakage of this nature could lead to carbon transport and carburization of the steam generator tubes and ultimately result in failures that could affect reactor safety.

The second type of steam leakage is of a pulsed nature, in which a large quantity of steam is introduced into the core of the HTGR over a brief period of time (the sources of ingress of this type have already been identified). The effects of large steam leakages include:

- (1) saturation or blockage of the helium cleanup system,
- (2) removal of carbon from graphite structural members,
- (3) removal of carbon from graphite matrix,
- (4) generation of hydrogen and carbon monoxide, which may lead to fire hazards if these gases are allowed to mix with air,
- (5) alteration of properties of coolant-circuit surfaces,
- (6) hydrolysis of carbide fuel, with concomitant increases in fission product release,
- (7) redistribution of fission products by formation of more volatile species (e.g., oxyhydroxides and carbonyl compounds).

At least six computer codes have been developed to account for corrosion of graphite by steam, but three of these have been developed outside this country and information regarding these codes is limited. Of the three U.S. codes, two have already been employed for analyses

of steam inleakage into a large HTGR, whereas the third code, STEAMCAR, would require extensive modification if adopted for this purpose. None of the codes consider all of the known phenomena associated with graphite oxidation in a radiation field, but it is not yet clear that all of these phenomena need to be considered in the first place. Furthermore, although an adequate experimental program which is designed specifically to test the computer codes has yet to be developed, steam injection experiments have been performed in the Dragon Project reactor, and a loop experiment is under way in France with which code validation tests can be conducted.

Input data regarding the kinetics of the steam-graphite reaction, including effects of internal geometry, need to be subjected to sensitivity analyses using an adequate, if simplified, computer code; and additional studies need to be made to refine these data as required. Similar extensions of previous investigations are needed on hydrolysis of carbide fuel particles, though these studies are likewise subject to an ordering of importance as a result of modeling studies. Consideration should also be given to the likelihood of formation of volatile forms of fission products, such as oxyhydroxides or carbonyl compounds.

8.2.4 Task area effort and schedule

The cost and manpower requirements of task area 3 are presented in Table 8.2. The only critical path involved concerns the development and certification of a computer code with which some of the technology assessments are to be made. Delay in this development, which is not included in this safety guide, would seriously hamper progress in this task area, especially if an operational code were not available before fiscal year 1976.

8.2.5 Assessment of primary coolant technology (Task group 3.0)

The purposes of this task group are (1) to provide a continuous technical review of the existing information on the consequences of steam ingress into HTGR primary coolant circuits in order to delineate safety aspects of such occurrences, (2) to describe in detail the information required to establish the behavior of the system insofar as identified safety concerns are involved, and (3) to assess the adequacy of the technology to provide the needed information.

Table 8.2. Summary of identified safety work in task area 3^a

Task description	Task priority	Costs (\$1000) and manpower (man-years in parentheses) by fiscal year										10-year total
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
Task group 3.0 - Assessment of the adequacy of primary coolant technology as regards safety factor	A	55 (1)	60 (1)	60 (1)	50 (1)	30 (0.5)	30 (0.5)	30 (0.5)	30 (0.5)	30 (0.5)	30 (0.5)	415 (7)

^aAdditional tasks may be assigned to this task area as information becomes available.

8.3 SEISMIC AND VIBRATION TECHNOLOGY (TASK AREA 4)

8.3.1 Task definition and objectives

The tasks in this area have the common goal of providing analyses of safety-related behavior involving modes of failure and responses. The factors considered in planning these tasks are mechanical vibration and dynamic loadings induced by normal operation and earthquakes. The dominant discipline in this group is structural mechanics.

8.3.2 Scope

The task area is divided into the following five task groups:

- 4.0 Need assessment for seismic and vibration technology
- 4.1 Core seismic response
- 4.2 Component seismic response
- 4.3 Soil-structure dynamic interaction
- 4.4 PCRV support

The components concerned in the last four of these task groups are all important to the safe operation of a reactor. The first task group is generic in nature. It will evaluate needs for all aspects of concrete structure and component behavior under seismic and vibration loadings.

The last four task groups involve research aimed at providing information required to identify and when necessary reduce the response of HTGRs to seismic loadings. Included are studies of the core structure and the various operation and support structure components. Although the potential failure of every element of a reactor system must be considered, only those whose failure could lead to serious consequences should be selected for inclusion in the safety program.

8.3.3 Status and needs

The status and needs of each task area are discussed in general terms in this section. A more detailed discussion of many of these task groups is included in subsequent sections dealing explicitly with the task groups.

Current HTGR designs call for the core to consist of vertical columns of hexagonal elements supported on a graphite floor and keyed together horizontally at the top layer only. The columns

have small clearance gaps between each other. Each hexagonal element is doweled to the element above and below it in the column, and the lower hexagonal element in each column is doweled to the core support floor. Elements can rock on each other to the small extent permitted by limited clearances within the core. The circumferential boundary of the core is restrained by flexible lateral supports mounted on the inner surface of the PCRV.

Thus, the response of the core to the horizontal seismic input becomes a problem of many individual elements that can move in horizontal planes within the limits of the movement of the plenum layer and core support floor, the flexibility of the columns, the restraint of the dowels, and the confining proximity of the adjacent columns.

In addition to the core seismic response, the maximum impact threshold for chipping or cracking of the fuel element should be known, since fuel elements in the HTGR core would experience numerous collisions with elements in adjacent columns during earthquakes. If the collisions are sufficiently numerous and energetic, the graphite components might be weakened or might crack. There is a need for further understanding of this behavior as well as for analytical methods to determine the time-history dynamic response of core structures.

Under earthquake conditions, main steam generators must be shut down and core cooling maintained. This requires satisfactory operation of many components such as the rods, actuators, and auxiliary cooling-loop backflow prevention devices. Moreover, the integrity of other devices is necessary to prevent compounding of the plant contingency condition. Analytical methods are needed for describing the seismic responses of these vital components.

The effects of soil-structure dynamic interaction on the seismic response of nuclear power stations have long been recognized as being significant. The response of the overall plant structure in turn governs the input into all the dynamic subsystems such as the reactor core, steam generators, etc. The transmission of energy from the base slab into the soil normally results in lower response levels than those predicted when the interaction effects are neglected. However, for certain ranges of parameters, increased levels can result, particularly when the effects of adjacent structures are considered.

Although theoretical formulations for nonlinear soil properties are in the development stage, they have not been completed. Some experimental studies have also been performed, but these tests have been confined to forced vibrations of rigid footings on soil

or very low-level forced vibrations of an actual nuclear plant having a particular site condition. While these tests are informative, they do not provide design data for a range of possible ground conditions and acceleration levels. Consequently there is a need to conduct scale-model tests to provide sufficient data for general analysis verification.

The PCRV support method being applied to HTGR plant designs provides essentially a rigid connection between ground and structure. An alternative method would be to decouple the PCRV from ground seismic forces by combinations of flexible supports, dampers, and slip members, thereby reducing seismic loading on the structure and plant equipment. Any such reduction in seismic loading would enhance plant safety under earthquake conditions, particularly where the seismic intensities are high. There is need for a study of new types of support systems and the adaptation or development of analyses to be used in this work.

In addition to seismic loadings reactor internal structures such as the side-restraint structure, core-support structure, thermal-barrier cover plates, steam generator, and internal seals are subjected to vibrations created by circulation machinery and/or flow-induced turbulences. There is a need to determine specifically how the requirements of the AEC Safety Guide Standards (particularly those in Safety Guide 20) are to be met by a study of methods of predicting vibration effects and measuring vibrations on-site.

8.3.4 Task area effort and schedule

The associated tasks and information flow of the task areas are shown in Fig. 8.2. Table 8.3 gives the cost projection of each task. The following sections give a general description of the proposed studies as organized under the appropriate task group.

8.3.5 Need assessment for seismic and vibration technology (Task group 4.0)

This task will provide a general safety assessment of seismic and vibration technology for HTGRs with particular emphasis on PCRV structure, core, and components.

Scope and justification

There are a number of efforts under way to acquire experimental data and improve methods of analysis for determining the effects

Fig. 8.2. Task groups 4.0 - 4.4. Seismic effects investigations.

Table 8.3. Summary of identified safety work in task area 4^a

Task description	Task priority	Costs (\$1000) and manpower (man-years in parentheses) by fiscal year										10-year total
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
Task group 4.0 Need assessment												
4.0.1.1 Need assessment and technology review	A	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	25 (0.5)	25 (0.5)	25 (0.5)	25 (0.5)	400 (8.0)
Task group 4.0 total		50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	25 (0.5)	25 (0.5)	25 (0.5)	25 (0.5)	400 (8.0)
Task group 4.1 Core seismic response												
4.1.1 Source tasks												
4.1.1.1 Seismic motions and effects	C		80 (1.0)	150 (2.0)	150 (2.0)	150 (2.0)	150 (2.0)	100 (1.3)	50 (0.7)			830 (11)
4.1.1.2 Core scale-model seismic tests	A	600 (8)	900 (12)	1200 (16)	1500 (20)	1000 (13)	700 (9)	300 (4)				6,200 (82)
4.1.1.3 Fuel-element impact tests	A	60 (1)	150 (2)	150 (2)	100 (1.5)							460 (6.5)
4.1.2 Synthesis tasks												
4.1.2.1 Analytical model of core seismic response	C	75 (1)	300 (4)	300 (4)	300 (4)	300 (4)	200 (2.5)	100 (1.5)				1,575 (21)
Task group 4.1 total		735 (10)	1430 (19)	1800 (24)	2050 (27.5)	1450 (19)	1050 (13.5)	500 (6.8)	50 (0.7)			9,065 (120.5)
Task group 4.2 Component seismic response		50 (0.7)	250 (3.3)	500 (6.7)	800 (10.7)	600 (8)	300 (4)					2,500 (33.4)
4.2.1 Source tasks												
4.2.1.1 Seismic-induced component motions and their effects	B											
4.2.1.2 Component seismic response tests	B											
4.2.2 Synthesis task												
4.2.2.1 Analytical models of component seismic response	B											
Task group 4.2 total		50 (0.7)	250 (3.3)	500 (6.7)	800 (10.7)	600 (8)	300 (4)					2,500 (33.4)
Task group 4.3 Soil-structure dynamic interaction		70 (1)	150 (2)	250 (3.3)	250 (3.3)	100 (1.4)						820 (11)

Table 8.3 (continued)

Task description	Task priority	Costs (\$1000) and manpower (man-years in parentheses) by fiscal year										10-year total
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
4.3.1 Source tasks												
4.3.1.1 Soil-structure interaction	C											
4.3.1.2 Soil-structure seismic model tests	C											
4.3.2 Synthesis task												
4.3.2.1 Analytical models of soil-structure seismic response	C											
Task group 4.3 total		70 (1)	150 (2)	250 (3.3)	250 ^a (3.3)	100 (1.4)						820 (11)
Task group 4.4 PCRV support			180 (2.4)	400 (5.3)	500 (6.7)	400 (5.3)	400 (5.3)	300 (4)	200 (2.7)	100 (1.3)		2,480 (33)
4.4.1 Source tasks												
4.4.1.1 PCRV support-system concepts	C											
4.4.1.2 PCRV support model tests	C											
4.4.2 Synthesis task												
4.4.2.1 Analytical models of PCRV seismic response	B											
Task group 4.4 total			180 (2.4)	400 (5.3)	500 (6.7)	400 (5.3)	400 (5.3)	300 (4)	200 (2.7)	100 (1.3)		2,480 (33)
Equipment costs		810	1705	2125	1000	1000	700	400				7,740
Task area 4 total		1715 (12.7)	3765 (27.7)	5125 (40.3)	4650 (49.2)	3600 (34.7)	2500 (23.8)	1225 (11.3)	275 (3.9)	125 (1.8)	25 (0.5)	23,005 (205.9)

^aAdditional tasks may be assigned to this task area as information becomes available.

of seismic loadings and operational vibrations on PCRVs. However, there is need for an independent review of the HTGR system with respect to its unique structural components and an evaluation of the probable effects of vibrations occurring during both normal operation and seismic disturbances. An assessment could then be made of the adequacy of present methods of design and analysis employed for the various components. This assessment would provide the information needed to develop recommendations for further work leading to more clearly defined margins of safety.

Status

Considerable work has been undertaken to determine seismic and vibration response of the PCRV structure, core, and components as discussed in the following sections of this task group. Present procedures consist of using existing general purpose and special types of analysis in conjunction with model studies. There is a need to review the capabilities of the existing methods of analysis as well as the type and scale of model studies being employed.

Identified work

An overall independent assessment (task 4.0.1.1) should be made of the status of HTGR seismic and vibration technology to determine areas in which additional safety margins, features, and/or study may be needed to provide the required degree of safety. Areas of particular interest are core and component seismic analysis and design procedures. A critical review should be made of ongoing seismic work. This work should be coordinated with the task area 6 assessment study of HTGR core support structures.

8.3.6 Core seismic response (Task group 4.1)

The purpose of this task group is to improve the capability to predict the response of the HTGR core to a seismic disturbance.

Scope and justification

The proposed core studies will be conducted to develop capabilities for testing reduced-scale core mockups to determine the seismic response of full-size HTGR core structures. The approach will be to test core mockups of reduced size and to develop the capability of scaling the results to actual size. Core integrity during a seismic disturbance is vital to the reactor shutdown and core-cooling capabilities.

Status

A seismic research and development program is under way at GAC to ensure the structural integrity of the HTGR core under safe-shutdown earthquake conditions. The objective is to verify the design adequacy of the core and the core supports for at least 0.5-g peak ground acceleration in accordance with the AEC regulatory requirements. Additionally, the program will study the effect on core response of such parameters as core size, support configuration, core gap, and variation in earthquake characteristics and the effects due to the operating environment.

The seismic program is carried out in two parallel phases, an analytical phase and an experimental phase. The experimental phase consists of tests on scale models of the core designed to satisfy the requirements for dynamic similarity. By verification of the model scaling laws, HTGR design loads are obtained directly. In the analytical phase, mathematical models of the core will be developed and correlated with the results from the tests. Once accurate correlation has been achieved, HTGR design loads are determined analytically. The full-scale analytical model input parameters are obtained using the verified scaling laws or from full-scale component tests (i.e., full-scale collision test).

This work is confined to the components in the PCRV core cavity: the fuel elements, reflector blocks, core support blocks, and core support structures. Thus, the core structures have been isolated from the PCRV structure, an assumption that is justified by the fact that the PCRV is infinitely stiffer than the core structures, in comparison, and the weight of the core is only about 2.5% of the PCRV weight.

For test and analysis, the time-histories of synthesized earthquakes and actual earthquakes, such as the El Centro and Taft events, are used. The earthquakes used to obtain core and core-support design loads satisfy the latest AEC design spectra. These earthquakes are the PCRV response at the core-support level obtained from seismic analysis with a linear model of the containment building and the PCRV. This model is coupled to an elastic half-space that represents the appropriate soil conditions for the plant.

The experimental portion of the program is divided into three groups of tests:

- Group 1. Functional tests on fuel elements and other components of the core and its associated structures. (The appropriate tests will provide acceptance criteria for these

components. (Forcing functions for these tests are derived from the full array tests of group 3.)

Group 2. Dynamic response tests of limited assemblies of components. (These tests are part of the preparation work for the tests in group 3. Also, the results provide basic input to the analytical core models.)

Group 3. Tests on scaled models of the whole or portions of the core. (The models are subjected to harmonic vibrations and/or random excitations corresponding to real and artificial earthquakes. The results from corresponding analytical models are correlated with these test results. Also, the results from the full array models subjected to forcing functions that satisfy the AEC criteria are employed as core design loads.)

In group 1, the strength of the standard fuel element under both static and dynamic loads has been determined. The dynamic tests were high-velocity impact of fuel elements with a near rigid mass. Based on subsequent two-body impact tests between fuel elements and control rod elements employing the time-history response of impact from typical earthquakes as derived from the full-array tests of group 3, fatigue curves are being established showing the number of cycles to failure as a function of relative impact velocity and impact force.

The following tests in group 2 are completed except for test 4.

- (1) Two-body collision tests between fuel blocks of 1/5, 1/2, and 1/1 scale were performed to study the collision dynamics of graphite, that is, to obtain the force/time relationship of impact, coefficient of restitution, etc. Multi body collision tests between blocks (five blocks on rails) generated information about fuel-block impact and separation behavior. Such information provided basic input to the analytical core models and aided in the verification of the assumed model scaling factors.
- (2) Vibration tests were conducted in which sinusoidal excitations were imposed on the ends of single fuel columns scaled to 1/4 and 1/8. A 1/8-scale graphite column, a 1/4-scale graphite column, and 1/4-scale acrylic column were tested. An axial static force simulating the pressure force was

applied to the top of the column. The input frequency ranged from 0 to 18 Hz, and the input displacement from 1/16 to 1/4 in. The data obtained were used to study the "column effects" on a planar array as part of the effort to assess how the three-dimensional core can be described by a two-dimensional model.

- (3) Hydromechanical shaker tests on 91 1/5-scale fuel blocks were conducted in a geometry representing a two-dimensional planar section of the core. This test was considered a forerunner to the 1/5-scale full-array tests of the HTGR core and core-support structures and provided a check-out for the hydro-mechanical shaker control system, instrumentation, and computerized data-acquisition system. These tests concluded an extensive instrumentation development program designed to provide displacements, velocities, and forces on elements in the core and displacements and forces measured at the core support structure.
- (4) Tests are being performed on two modified two-dimensional planar section models of the core with 1/5-scale and 1/2-scale fuel blocks. The test objectives are: (1) obtain the effect of change in core gap on core resonance, fuel block impact response, and boundary response; (2) correlate results from two-dimensional analytical models with test results; and (3) obtain further data in aid of verifying the model scaling laws.

In group 3, two tests have been undertaken:

- (1) Tests of a 1/4-scale model of a partial core are planned in cooperation with the French Atomic Energy Commission (Commissariat à l'Energie Atomique) at their research facilities in Saclay near Paris. The model is built of graphite and consists of 10 levels of 91 fuel columns in full (scaled) height.
- (2) Tests of a 1/5-scale model of the full core built in the United States have been carried out. The model is built of graphite and is designed to be a faithfully scaled reproduction of the real HTGR core and core support structure, incorporating all design details that may affect the dynamic response. It represents the core in its cold and fully irradiated state. Briefly the test objectives were to determine the core motion characteristics (force and deflection of lateral supports) and the core deflection and forces (number of repetitious loadings of fuel blocks). Core motion characteristics include core lumping, natural frequencies and mode shapes, and system damping; core deflections and

forces include fuel column and reflector column deflections, fuel block impact forces, and fuel block dowel pin forces.

The earthquakes have been applied at equivalent ground intensities ranging from 0.075 g to 0.75 g. The core was tested uniaxially in two orthogonal directions, across-the-flats of a fuel element and across-the-corners. Two different side-support designs were studied, a soft spring support (20,000 lb/in. equivalent full-scale stiffness) and a combination soft spring and hard stop support, which represents the HTGR core-support system.

To comply with the new AEC licensing requirements, further tests with the 1/5-scale model (modified partial or full array) will be undertaken with simultaneous multi-axis input. The multi-axis input will include one horizontal axis combined with one vertical axis excitation.

Computer code development and core modeling have been carried out in parallel with the test work. Although large finite-element structural codes are available, it has been recognized that the complex modeling of the nonlinear collision dynamics of stacked hexagonal graphite blocks is not readily achieved by the techniques available in such codes. Hence, emphasis was put on developing specialized codes for this purpose.

Two codes, COSAM⁷ and CRUNCH,⁸ have been developed for modeling the core structures. These codes use different principles in mathematical formulation. COSAM is based on the second-order differential equations treating a spring mass system, and CRUNCH employs impulse-momentum equations to describe the core-block collisions. So far these codes have been developed to represent a planar section of the core in one or two dimensions. Continuous development is in progress, however, to expand this capability to include the treatment of a vertical core section in two dimensions and possibly in full three dimensions.

Two single-array core models have been constructed for COSAM and CRUNCH. These represent the masses of two reflector blocks on each side of the core, each supported by one lateral spring pack. The array across the core consists of six fuel blocks in sections, 2, deep (one reflector block covers a sectional area equal to about six fuel blocks). In representing the core as an array, the cross coupling of core motion is assumed to be negligible. This in fact is a good assumption according to 1/5-scale test results. The coupled motion of the reflector block is accounted for by a spring between the blocks which represents the circumferential stiffness characteristics observed during 1/5-scale tests. Further, the effect of column stiffness on the individual fuel block motions are considered by designing an appropriate spring from the element

to the PCRV boundary. All elements are viscously damped to simulate system energy losses due to friction.

Good correlations with the 1/5-scale full-array test results have been obtained with these simplified COSAM and CRUNCH core models. Meanwhile, two-dimensional models are being constructed to handle multi-axis excitation. These models will be correlated with existing test data and subsequently with test data from the multi-axis tests.

Identified work

The seismic input data for the tests of the 1/4-scale and 1/5-scale models will consist of both sinusoidal and actual and synthesized time-history excitation. The results will be correlated with analytical models such as COSAM and CRUNCH. Required data for the core components include collision forces and frequencies, shear forces on the dowels, deflections or disarrays that could affect alignment, and extent of element lumping. For the lateral restraint structure, needed measurements include loads at each support point, deflection of the flexible support members, and number of load repetitions.

Fuel-element impact tests are also currently being performed to establish the resilience of the HTGR design with the reference material. The work identified in this task group consists of four main tasks - three source tasks and a synthesis task.

Source task 4.1.1.1 covers adaption to an HTGR design of information obtained from other studies aimed at describing the responses of the core to a given earth movement.

Source task 4.1.1.2 consists of bi-axial shaking tests on a 1/5- and possibly a 1/2-scale model. The 1/2-scale test would not be performed in the same place as the 1/5-scale test, but instrumentation would be the same. This task also provides validation of model predictions.

Source task 4.1.1.3 covers further work on the impact tests on fuel elements with particular emphasis on the effects of temperature and irradiation. Impact tests should be performed on blocks of the large HTGR core design. These blocks have a smaller number of large holes than the previously tested Fort St. Vrain blocks. The number of collisions of various energies required to initiate graphite web failure would be determined for both standard fuel elements and control rod elements. Compression strength changes due to repeated impacts should be determined. In the design of

the new experiments, techniques for simulating the actual stress conditions within the fuel elements at reactor temperatures after some irradiation should be explored.

Synthesis task 4.1.2.1 covers development of analytical models.

8.3.7 Component seismic response (Task group 4.2)

The purpose of this task group is to adapt and validate analytical methods for describing the seismic responses of vital components with emphasis on the nonlinear aspects of the system.

Scope and justification

Under severe earthquake conditions, the reactor and main stream generators must be shut down and core cooling maintained. This requires satisfactory operation of many components such as the rods, actuators, and auxiliary cooling-loop backflow prevention devices. Moreover, the integrity of other devices is necessary to prevent compounding of the plant contingency condition. Consequently valid analytical methods must be developed or adapted to describe the seismic responses of vital components, with emphasis on the nonlinear aspects of the system.

Status

A number of large, general-purpose computer programs are available for analyzing the dynamic response of complex systems with many degrees of freedom. Programs such as NASTRAN,⁹ SHAKE,¹⁰ DSAP,¹¹ and others are widely used but have little or no capability for handling nonlinear systems. Experimental investigations have clearly demonstrated the existence of strong nonlinearities in steam generators and other systems.

Identified work

This task group should investigate initially the feasibility of and then formulate a general-purpose solution technique for the time-history dynamic response of some selected nonlinear systems that exist in a large HTGR. The solution technique should be capable of treating both inherent nonlinearities, such as material yielding or opening and closing of clearances within the system, and discrete, and possibly large, sources of damping as opposed to model damping assumptions. This requires the numerical solution

of the nonlinear differential equations of motion on a large digital computer. Primary effort should be devoted to developing a method for handling the nonlinear stiffness matrix and to selecting or developing the best numerical integration algorithm that is mathematically stable for the widest possible range of practical problems without sacrificing either numerical accuracy or efficiency.

Concurrently the supporting program should be planned. Later phases of the task group should include development of analytical models for specific component classes and the extensive test program. The work has been divided into two source tasks and one synthesis task.

Source task 4.2.1.1 performs the initial investigations that establish the theoretical basis for the response models. In addition, this task develops plans for the large-scale experimental program that will be required to generate empirical data for the analytical models and to validate the model results.

Source task 4.2.1.2 includes the experimental program itself, involving excitation of components by shake table or other means and observation of the responses. This task fulfills the aims of a validation task since many tests will be done on real components.

Synthesis task 4.2.2.1 provides analytical models.

8.3.8 Soil-structure dynamic interaction (Task group 4.3)

The purposes of this task group are to complete the development of analysis methods for nonlinear soil properties and to provide reliable test data for verification purposes.

Scope and justification

Both finite-element and elastic half-space methods of analysis should be studied. Experimental studies should consist of testing models under a wide variety of simulated foundation conditions. Such a program would provide experimental verification of the analytical model over a wide range of parameters including effects of soil properties, acceleration levels, layered media, and embedment.

These efforts toward establishing advanced models with additional capability concerning various significant ground effects should provide basic data for all large reactors, including fast reactors,

and would therefore constitute scientific work of a general nature that would benefit all reactor development and safety.

Status

Recent investigations generally fall into either finite-element methods or those methods treating the soil medium as an elastic half-space, and both approaches have advantages as well as limitations. The finite-element method permits an easier treatment of nonuniform soil properties but is costly in terms of computer time and is restricted to very simple representation of the structure. Extension of an elastic half-space to account for nonlinear soil properties is not as straightforward as for the finite-element method but allows essentially an unlimited number of degrees of freedom for the structure modeling in addition to economical computer-run times. This approach has recently been extended to account for the effects of through-soil coupling on the dynamic response of adjacent structures founded on separate base slabs. In this model, the ground medium is represented by a homogenous, isotropic, elastic half-space that is coupled with up to three flexible, damped, three-dimensional structures on separate base slabs. This effort resulted in the successful development of a computer code SOSDIT¹² that is capable of treating both steady-state and transient response problems.

Some experimental studies have also been performed, but these tests have been confined to the forced vibrations of rigid footings on soil or very low-level forced vibrations of an actual nuclear plant with particular site conditions. While these tests were informative, they could not provide design data concerning the structural response for a range of possible ground conditions and acceleration levels under earthquake excitation.

Identified work

The theoretical formulation for nonlinear soil properties should be completed and a general purpose program developed which would be capable of treating nonlinear effects on the seismic response of reactor structures. This program should then be extended to account for additional considerations such as layered soils.

An experimental program is suggested to verify the analytical portion of the investigation. Because testing of full-scale reactors is impractical for other than extremely low acceleration levels and, even if accomplished, would yield only a single data point corresponding to a given site condition, a scale-model test program

is identified which would utilize foam beds to simulate the soil media. The work is divided into two source tasks and one synthesis task.

Source task 4.3.1.1 extends current investigations of nonlinear soil properties. Source task 4.3.1.2 covers the model tests described above. Synthesis task 4.3.2.1 covers the production of analytical models.

2.3.9 PCR support (Task group 4.4)

The purpose of this task group is to investigate alternative PCR support system concepts.

Scope and justification

The proposed work involves studying PCR support-system concepts for various degrees of flexibility and different seismic intensities, with the emphasis on the high seismic-intensity range. Other research and development work on components may be needed, such as supplementary dampers required for very flexible support. Development of analytical techniques that may be required in the study will be included. The development of a damping system or flexible coupling would provide additional resistance to seismic loadings.

Status

The current PCR support structure, which consists of six radial walls, supports the PCR above the containment mat. The PCR is cast integral with the support structure and containment mat.

Identified work

A test program including both static and dynamic model tests should be carried out to confirm the analysis and design. This work has been divided into two source tasks and one synthesis task.

Source task 4.4.1.1 investigates alternative PCR support-system concepts, considering effects of various degrees of flexibility, damping, and slip.

Source task 4.4.1.2 covers any model tests needed to confirm analytical predictions or to investigate new PCR support arrangements.

Synthesis task 4.4.2.1 includes development of analytical models for parametric studies.

8.4 CONFINEMENT COMPONENTS (CONCRETE REACTOR VESSEL AND CONTAINMENT STRUCTURES) (TASK AREA 5)

8.4.1 Task definition and objectives

The tasks selected for inclusion in the confinement components portion of the safety program plan will provide for an independent review of PCRV and reactor containment-structure technology. The objective is to provide a continuing assessment of existing capabilities to identify and maintain well-defined margins of safety during both construction and plant operation. In addition, available methods of analysis will be used to study PCRV and reactor containment behavior to provide information needed for evaluation of present concrete containment-structure designs and to prepare recommendations for further research and/or development.

8.4.2 Scope

The task area is divided into two task groups. Task group 5.0 provides for a continuing review and assessment of confinement component technology; task group 5.1 considers both the development of improved methods of failure analysis and the use of accepted methods of analysis in studying possible regions of distress in PCRVs and reactor containments.

These studies will cover every aspect of concrete containment-structure technology including design, construction, and operation. Emphasis will be on recent developments, and recommendations will be made for further safety research and development to satisfy identified needs; the necessary follow-on studies will be planned.

8.4.3 Status and needs

Present and planned HTGR systems utilize a PCRV that houses the entire gas-pressure circuit in addition to containing the reactor. The more recent large-scale HTGR plants also call for concrete secondary or reactor containment structures. In the following sections, PCRVs and reactor containment structures when discussed collectively will be referred to as concrete containment structures.

The PCRV is a complex concrete structure that depends primarily on steel prestressing and indirectly on a thermal barrier for structural integrity and on a steel liner and penetrations for leak tightness. Two basic types of PCRVs are employed - a single-cavity vessel as used in the Fort St. Vrain Power Station and a multi-cavity vessel as proposed for the Summit Power Station. Reactor containment structures are constructed of conventional reinforced, prestressed, or partially prestressed concrete. HTGR power plants have been designed both with and without reactor containment structures. A more detailed description of concrete containment structures is included in Sect. 12 (Part III) of this report.

Although concrete containment structures have been in use for a number of years, much of the technology is still under development owing to the complexity of the structure and the increasing demands for a thorough and accurate understanding of design safety margins. A proposed "Standard Code for Reactor Vessels and Containments" (proposed Section III, Division 2, ASME Boiler and Pressure Vessel Code) has been developed and is in the trial-use period prior to final adoption. As in the case of any new code document, it will be subjected to continual critical review and assessment.

The two task groups included in the proposed safety program are a need assessment study of concrete containment structures and an analytical study of PCRV failure modes. The proposed efforts are described in the following subsections.

8.4.4 Task effort and schedule

Table 8.4 gives the cost projection of each task. General descriptions of the proposed studies as organized under the two task groups are given below.

8.4.5 Need assessment (Task group 5.0)

This task group will provide a continuing independent evaluation of the safety of PCRVs and reactor containment structures in light of existing and new technological developments as well as data obtained from operating HTGRs.

Scope and justification

This task group is divided into a critical review and a safety assessment of existing technology to identify possible areas needing

Table 8.4. Summary of identified safety work in task area 5^a

Task description	Task priority	Costs (\$1000) and manpower (man-years in parentheses) by fiscal year										10-year total
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
Task area 5 -- Confinement components												
Task group 5.0 Need assessment for confinement components												
5.0.1.1 Initial technology assessment	A	50 (1.0)										50 (1.0)
5.0.1.2 Continuing need assessment study	A		50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	25 (0.5)	25 (0.5)	25 (0.5)	25 (0.5)	25 (0.5)	325 (6.5)
Task group 5.0 total		50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	25 (0.5)	25 (0.5)	25 (0.5)	25 (0.5)	25 (0.5)	375 (7.5)
Task group 5.1 Failure analysis												
5.1.2.1 Analytical PCRV failure studies	B	25 (0.5)	25 (0.5)	25 (0.5)	25 (0.5)	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	25 (0.5)	375 (7.5)
Task group 5.1 total		25 (0.5)	25 (0.5)	25 (0.5)	25 (0.5)	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	50 (1.0)	25 (0.5)	375 (7.5)
Task area 5 total		75 (1.5)	75 (1.5)	75 (1.5)	75 (1.5)	100 (2.0)	75 (1.5)	75 (1.5)	75 (1.5)	75 (1.5)	50 (1.0)	750 (15.0)

^aAdditional tasks may be assigned to this task area as information becomes available.

further research and/or development. This initial study will be followed by a continuing review and assessment of HTGR concrete reactor containment technology in light of new information and developments. The primary purpose of this task is to obtain the information needed to determine the actual margins of safety of these complex concrete containment structures.

Status

Three survey reports, a bibliography¹³ and critical review¹⁴ of PCRV literature and a review of reactor containment design and construction practices,¹⁵ were prepared under the ORNL concrete program. Because the most recent of these documents was published more than four years ago, it does not present information on innovations such as multi-cavity PCRVs and reactor containments for HTGRs.

Identified work (Tasks 5.0.1.1 and 5.0.1.2)

Information published in the literature and obtained directly from manufacturers, designers, vendors, and constructors should be collected and subjected to critical review, and an evaluation made of the safety aspects of concrete containment structures. The survey should concentrate primarily on information developed since completion of the earlier critical reviews.^{14,15} The evaluation of this information should both identify those areas of the technology having possible safety uncertainties and outline the investigations required to eliminate these uncertainties. The design, construction, and operational aspects of the concrete, prestressing and conventional reinforcement, liners and penetrations, insulation and cooling systems as well as the instrumentation required to measure stress, strain, time, and temperature behavior should be evaluated.

Task 5.0.1.2 should follow the initial task to provide an independent review of efforts to eliminate any identified deficiencies and of the influence of new technological developments on reactor safety. This follow-on task is intended to be a continuing evaluation effort to identify problem areas and to specify the additional work required to assure the safety of HTGR concrete containment structures.

8.4.6 Failure analysis (Task group 5.1)

The goals of this task group are to verify and apply the capabilities for predicting the mode of failure of PCRVs and for accurately predicting stress distributions and dimensional changes of a PCRv over its service life.

Scope and justification

PCRVs are designed using elastic analyses, creep analyses, and semiempirical failure or ultimate load analyses. Elastic analyses are used to predict stresses in the unpressurized vessel at the beginning of design life and in the pressurized vessel at the end of its design life, while the long-term behavior of PCRVs is predicted using creep analysis. Of special interest is the prediction of prestress losses and the additional strains that concrete creep imposes on the steel liners. The overall stress distributions however will not be greatly affected by creep. The ideal failure analysis is one that accurately predicts the initiation and propagation of internal cracks within the head region as well as the side walls of a concrete vessel.

The task group is closely coupled to the heat-failure studies of task group 5.2* as well as to the model studies and surveillance work in task group 5.8 which provide the data against which the analytical methods are compared. There is also substantial dependence on the development of physical properties of concrete and interaction of various effects (task groups 5.2 and 5.3). Once the analysis techniques have been adequately verified, they could be used to conduct studies of representative concrete containment structures and to identify potential problem areas. This would provide information to be used in code rule development (task group 5.9).

Status

There are a number of sophisticated methods for analyzing both short-term elastic and long-term time-dependent loadings of concrete containment structures. Computer program forms of finite-element methods have been developed which can account for three-dimensional

* All task groups numbered 5.2 and higher in task area 5 are described in Part III of this document.

elastic, short-term deformation and axisymmetric creep and cracking effects. Short-term behavior can be analyzed with good accuracy by considering primarily elastic deformations and, where elastic limits are exceeded, some degree of plastic deformation. The three-dimensional methods of stress analysis cannot yet fully account for long-term time-dependent behavior, particularly of complex geometries. The necessary constitutive equations could be developed to describe these long-term effects, provided the basic materials behavior can be characterized. All of the resulting analyses must be thoroughly verified by appropriate model tests. Present strategy is to overdesign complex portions of the structure so as to ensure that any failure occurs in the more benign barrel region, thereby precluding catastrophic failure in other, less well-understood regions of the vessel. Development of a satisfactory failure analysis has been hampered by the unavailability of reliable information on load-deformation properties under complex states of stress (see task group 5.2) and the lack of a general failure criterion. This particular weakness in the analysis needs correcting before the goal of developing reliable numerical models of PCRV head regions that can accommodate material behavior realistically can be realized.

Identified work (Task 5.1.2.1)

The currently accepted methods and eventually the improved analysis methods developed within the engineering and technology program should be used to conduct studies of the modes of failure of current and planned vessel configurations. The most satisfactory methods of analysis should be used to conduct evaluation studies of the PCRV liner, concrete prestressing, and thermal barrier for the purpose of identifying possible areas of distress.

8.5 PRIMARY SYSTEM MATERIALS TECHNOLOGY (TASK AREA 6)

8.5.1 Task area objectives

This task area is concerned with safety aspects of the materials used in the HTGR primary system and is directed to (1) assessment of areas in which additional safety margins, features, and/or work should be performed and (2) conduct of such work.

8.5.2 Scope

Task area 6 is divided into an assessment task group and a task group on control components. The assessment task group concerns the behavior of metals and graphite, as well as control components of HTGR.

Need assessment for primary system materials technology (Task group 6.0)

This task group provides continuing assessment of primary system materials technology with the aim of identifying areas in which additional safety margins, features, and/or work could add substantial improvements in HTGR safety.

Control components (Task group 6.1)

Research in this task group will include study of the reaction of HTGR control materials with coolant impurities and the boron transport process. Structural stability of control materials will be evaluated, and the integrity and compatibility of control rod cladding materials will be determined. Models for predicting the behavior of control materials will be developed and validated by tests on control materials removed from the Fort St. Vrain reactor.

8.5.3 Status and needs

Need assessment for primary system materials technology (Task group 6.0)

Several factors contribute to a need for continuing assessment of the safety aspects of HTGR primary system materials. First, there is little long-time in-reactor experience with the materials of current HTGR system design. If some unexpected materials behavior is observed during reactor life, it must be assessed and related to safety. If design or operating changes are made, the effects of these on materials behavior and safety must be determined. Any new materials used in the future will also require safety assessment.

Control components (Task group 6.1)

Volatile boron oxides are known to be formed by reaction of B_4C with air and/or water vapor. The extent of such reactions on all candidate materials in all possible HTGR environments has not been studied. Further data are needed on the swelling of boronated graphites of interest for HTGR control materials and the effects of thermal transients. Finally, the compatibility of Incoloy 800 with boronated graphite in the reactor environment needs additional demonstration.

8.5.4 Task area effort and schedule

The work identified for materials technology is summarized in Table 8.5, along with estimated costs and manpower needs. A significant activity is the ongoing study of technology status and identification of needs as they apply to safety. This activity is expected to continue for the duration of the safety program and may result in the definition of additional safety work.

The only task group in which additional activity is currently identified relates to reactor control materials (task group 6.1). The two source tasks listed in Table 8.5 call for expenditures continuing over a four-year period. Task 6.1.1.1 calls for intensive effort over this period. Additional details regarding the identified safety work in this task area are presented in the following sections.

8.5.5 Need assessment for primary system materials technology (Task group 6.0)

Scope and justification

As changes in design and materials are made in HTGR systems, the performance and behavior of primary system structural methods must be assessed relative to safety and safety margins. Further, knowledge obtained in base program studies and through reactor operating experience also forms a basis for needed safety assessment.

Status

The design of HTGR systems has necessarily included assessments of the type being discussed. Additionally, similar assessment was performed in the course of developing this document.

Identified work

Need assessment for control materials (Task 6.0.1.1). The need for improved control materials as further testing of present materials may become apparent as a result of the efforts associated with source tasks 6.1.1.1 and 6.1.1.2, below, and/or related activities in the engineering and technology program. Proposed changes in control materials will need to be evaluated to determine whether further safety or safety-related testing or analyses need to be performed. The performance of boronated graphite control materials is strongly related to the graphite component. For this reason,

Table 8.5. Summary of identified safety work in task area 6^a

Task description	Task priority	Costs (\$1000) and manpower (man-years in parentheses) by fiscal year										10-year total
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	
Task group 6.0 Need assessment for primary system materials technology		50 (1)	50 (1)	50 (1)	60 (1)	60 (1)	60 (1)	70 (1)	70 (1)	70 (1)	70 (1)	610 (10)
6.0.1.1 Need assessment for control materials. Safety-related assessments of control rod, reserve shutdown system, and burnable poison materials.	A											
6.0.1.2 Need assessment for graphite materials. As above for core support, reflector, and moderator graphites.	A											
6.0.1.3 Need assessment for metallic materials. As above for metallic materials of steam generator, thermal barriers, and circulator.	A											
Task group 6.0 total		50 (1)	50 (1)	50 (1)	60 (1)	60 (1)	60 (1)	70 (1)	70 (1)	70 (1)	70 (1)	610 (10)
Task group 6.1 Control materials												
6.1.1 Source tasks												
6.1.1.1 Reactions of coolant impurities with HTGR control materials. Assessment of potential for boron movement due to water vapor, etc. Effect of temperature, impurity concentrations, B ₄ C content and distribution, and irradiation.	A	160 (4.0)	180 (4.5)	180 (4.5)	20 (0.5)							540 (13.5)
6.1.1.2 Structural stability of HTGR control materials. Irradiation stability of full-size compacts and effects of thermal transients.	B	80 (2.0)	80 (2.0)	80 (2.0)	40 (1.0)							280 (7.0)
Task group 6.1 total		240 (6.0)	260 (6.5)	260 (6.5)	60 (1.5)							820 (20.5)
Task area 6 total		290 (7.0)	310 (7.5)	310 (7.5)	120 (2.5)	60 (1)	60 (1)	70 (1)	70 (1)	70 (1)	70 (1)	1430 (30.5)

^aAdditional tasks may be assigned to this task area as information becomes available.

a continuing assessment of information generated on graphite components (task group 6.2) is appropriate for control materials as well.

Need assessment for graphite materials (Task 6.0.1.2). The properties and behavior of graphite employed as core supports, reflectors, and moderator in the HTGR primary system will be assessed to identify any safety-related areas that should be examined further.

Need assessment for metallic materials (Task 6.0.1.3). The metallic materials used in the steam generator, thermal barriers, and circulators have important safety significance. Any safety-related problems involving the behavior of these materials will be identified in this task through continuing overview of materials technology and operating experience.

6.5.6 Control materials (Task group 6.1)

Scope and justification

The reference neutron-absorber material (boronated graphite) employed in HTGR control rods, reserve shutdown system, hexagonal reflector blocks, and burnable poison consists of discrete particles of B_4C in a graphite matrix. The oxidation behavior of the boronated graphites is significant for all applications because oxidation rates, if high, could lead to the redistribution of unprotected boron in the core through transport of volatile B_2O_3 . The extent of this oxidation and the potential for boron transport need to be determined for long-term normal operating conditions with low concentrations of oxidizing impurities (including oxygen and water vapor) in the coolant and for accident conditions in which concentrations of such impurities could increase significantly above normal levels.

The integrity of the control-rod containment and shock absorber during reactor operation must be maintained. The radiation-induced dimensional changes in control-rod compacts and burnable-poison compacts must be well known to design components without mechanical interaction between the control material and its containment. The effects of thermal transients on the control materials should be determined. Thermal aging, irradiation damage, and long-term exposure to boronated graphite and low-level coolant impurities on the mechanical properties of containment and shock absorber materials should be evaluated.

The surveillance of neutron-absorber-containing components is required to provide a continuing assessment of the adequacy of designs.

Status

The formation of a volatile boron oxide caused by exposure of B_4C to air and/or water vapor has been observed.¹⁶ The oxidation behavior of Fort St. Vrain burnable-poison compacts was examined at several temperatures and oxidant concentrations representative of the most severe HTGR conditions.¹⁷ The investigation concluded that the oxidation rate would be under Fort St. Vrain operating conditions including postulated accidents. However, similar tests on encapsulated, but vented, control-rod materials have not been made, and no information exists on whether oxidation is aggravated by neutron irradiation.

Radiation-induced swelling of boronated graphite containing 5 to 7 wt % boron has been observed,¹⁸ and dimensional changes have been determined for Fort St. Vrain control-rod materials to conditions beyond lifetime service requirements. However, no data have been reported for Fort St. Vrain burnable poison materials or full-sized control-rod compacts. Similarly, no data have been reported on the effects of thermal transients on such irradiated materials.

Information on the compatibility of Incoloy 800 with 40 wt % boron-containing graphite is not documented. Of particular interest would be the effect of low boron oxide concentration. Similarly, the impact capability of the thin-gage Incoloy 800 shock absorber after long-term exposure to the reactor coolant has not been reported.

Identified work

Reactions of coolant impurities with HTGR control materials (Task 6.1.1.1). The potential for movement of boron from boronated graphite under possible accident conditions, with high concentrations of water vapor and other oxidizing impurities in the helium coolant, should be studied. The potential for boron movement under long-time exposure to low concentrations of these impurities should also be studied.

Boronated graphite should be tested under various conditions of temperature and impurities to determine reaction rates and the nature of the reaction products. Reaction rates should also be determined for boronated graphites with various B_4C content and B_4C particle sizes. The behavior of irradiated material should be determined to see if the potential for boron movement is accelerated by irradiation-induced swelling and associated porosity or fragmentation. The effect of ionizing radiation on rates should also be determined.

Structural stability of HTGR control materials (Task 6.1.1.2).

Irradiation-induced swelling of boronated graphites should be determined through a program of irradiation testing and examination. Of particular interest is the stability of irradiated material to thermal transients representing conditions appropriate to interruption-of-cooling accidents. Thermal transient testing should be conducted on materials with various fluence and irradiation temperature histories.

The products resulting from these tests would provide samples for the oxidation stability tests in task 6.1.1.1.

8.6 SAFETY INSTRUMENTATION (TASK AREA 7)

8.6.1 Task area objectives

The subject of this task area is the assessment of the need for additional instrument protection features to help assure or enhance the safety of HTGR plants. The assessment includes identifying or establishing plant requirements for surveillance, monitoring, and trouble diagnosis which might aid in the safe operation of the plant. The assessment should take into account the three levels of safety defined in ref. 19, which require design for a high level of safety in normal operation as well as protection against unavoidable upset conditions. This definition of safety demands consideration of the safety implications of measurement, display, and control features so that the plant can operate with a low challenge rate to the plant protection system. Major emphasis should be devoted to plant protection system requirements where accurate and timely response is necessary for the measurement of significant plant parameters when plant upsets occur. Further, protective systems requirements should be consistent with providing appropriate action to restore the plant to a safe condition under all contingencies and to actuate engineered safeguards for the prevention of subsequent damage and release of radioactivity. Also to be included is a study to determine what measurements might be required for the plant to remain operable during and after accidents in which some parts of the plant protection system are presumed to have failed and to indicate development requirements for instrumentation to accomplish such measurements.

8.6.2 Scope

The scope and specific objectives of this task area include not only assessment of system safety requirements that are instrument related and unique to HTGR plants, but also the embodiment or

special arrangements of instrumentation developed for use on other reactor types. In some cases development needs may be cited which are also identifiable with other reactor types, but which are nonetheless important for HTGRs. Such items are included in the plan with the recognition that parallel or complementary efforts may be planned or under way as part of other reactor development programs. Instrumentation for surveillance of PCRVs is not included in this task area but is included in Sect. 12, task group 5.7.

8.6.3 Status and needs

Brief summaries of the current status and anticipated technology assessment needs are included as part of the specific tasks that have been identified as related to specific instrumentation areas. In addition, a task is identified which includes a state-of-the-art study to examine in detail the present status and the potential for improvement of safety instrumentation not yet defined, and this might lead to specification of additional development tasks.

8.6.4 Task area effort and schedule

Although the only activities in task area 7 that are identified with the safety guide address the subject of need assessment, it is recognized that other activities may, in the future, receive support under HTGR safety. Because of the variety of subjects treated in task group 7.0, the work has been divided into several separate tasks dealing with need assessments and determination of development requirements. The tasks represent different disciplines that would normally be undertaken by different people. The projected efforts and the schedules associated with these tasks are summarized below and in Table 8.6.

Task 7.0.1.1 is a study to determine the temperature measurements that are needed for safety and the consequences and probability of thermocouple failure that would adversely affect the safety status of an HTGR. Behavior of core exit gas thermocouples under normal operating conditions should be assessed, and, in addition, the behavior of temperature sensors in other locations (core inlet, in-core, heat exchangers, etc.) and under start-up, shutdown, or abnormal conditions should be evaluated for their effect on plant safety. This study should also identify safety needs that might be determined from on-line thermocouple surveillance. The effort associated with this task is expected to be two man-years for each of the first two years of the program.

Table 8.6. Summary of identified safety work in task area 7^a

Task description	Task priority	Costs (\$1000) and manpower (man-years in parentheses) by fiscal year										
		1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	10-year total
Task group 7.0 - Safety instrumentation assessment												
7.0.1.1 HTGR thermometry. A study to determine thermometry needs for safety and the consequences and probabilities of thermocouple failure adversely affecting the safety status of an HTGR. Includes identifying needs for on-line surveillance.	A	120 (2)	120 (2)									240 (4)
7.0.1.2 In-core flux monitoring. Assess HTGR safety needs for high-temperature in-core neutron detectors; this also involves assessing the state-of-the-art of existing detectors.	A	30 (0.5)										30 (0.5)
7.0.1.3 Surveillance and diagnostics by noise analysis and dynamic testing. Identify incipient failures that require early detection and process variables that might be monitored or dynamically perturbed for analysis related to adequate safety assessment.	C	60 (1)										60 (1)
7.0.1.4 Reactor coolant moisture monitoring. A detailed study of safety requirements.	A	60 (1)	60 (1)									120 (2)
7.0.1.5 Study to identify additional safety instrumentation needs. A comprehensive assessment of status and requirements of all instrumentation related to the safety of the plant.	A	180 (3)	120 (2)									300 (5)
Equipment costs												
Task area 7 total		450 (7.5)	300 (5)									750 (12.5)

^aAdditional tasks may be assigned to this task area as information becomes available.

Task 7.0.1.2 is a study to assess the need for in-core neutron detectors and measurements for HTGRs and to determine the state-of-the-art of in-core neutron detectors that might be applicable. This effort is expected to require about 0.5 man-years.

Task 7.0.1.3 is a study to define the requirements and propose a suitable program for surveillance and diagnosis by noise analysis and dynamic testing. The study is expected to determine suitable requirements for both program and equipment for typical HTGRs and is expected to require an effort of one man-year.

Task 7.0.1.4 is an assessment of the requirements for moisture detection in HTGRs and a comprehensive assessment of the current capabilities of available moisture monitors. The task is expected to require an effort of two man-years over a two-year period.

Task 7.0.1.5 is a comprehensive study to identify any additional safety instrumentation needs of HTGRs. Due to the lack of broad U.S. involvement and experience with gas-cooled reactors, some unrecognized problems may exist that will require development or correction. Also, additional development work on currently available equipment may be indicated. A four to five man-year effort in the next two years is proposed, so that additional development may proceed on a timely basis when the needs are identified. Because this study is expected to be comprehensive, it may overlap with some of the specific assessment tasks indicated above.

8.6.5 HTGR thermometry (Task 7.0.1.1)

Scope and justification

The safety status of an operating HTGR may be jeopardized or incorrectly assessed if temperature sensors fail or function inadequately; that is, thermocouples that may be adequate for normal operation may be too slow or may have too limited a range for reactor control during startup or shutdown conditions. In this task, the thermometry needs of the HTGR will be reviewed, the capabilities of existing thermocouples will be assessed, and the ability of plant safety to be maintained using such thermocouples will be evaluated. This study should also evaluate the degree to which HTGR plant safety can be improved by the use of surveillance methods.

Status

The capability of Chromel/Alumel thermocouples to meet HTGR core-exit-gas temperature-measuring requirements has been surveyed by Perez and Sandefur.²⁰ The use of core-exit-gas temperatures for determining core temperature and gas flow conditions has been recognized in HTGR plant design.^{21,22} Additional evaluations are needed to provide a quantitative basis for determining whether Chromel/Alumel, other thermocouple materials, or other temperature sensors can measure HTGR core-exit temperatures with sufficient accuracy and reliability to maintain adequate safety margins. Such data have been obtained for sheathed Chromel/Alumel thermocouples under LMFBR conditions but have not been obtained for higher temperatures and adverse chemical environment in HTGRs. Other temperature sensors, such as resistance, ultrasonic, fluidic, and noise thermometers, have been developed for other reactor programs but have not been quantitatively evaluated for use in HTGRs. Some thermocouple surveillance techniques have been studied, but need further development.

Identified work

The temperatures in all locations within the containment vessel for large HTGR plants should be estimated from prior experience with Peach Bottom and from calculations on models of the reactor system. The effects on safety of incorrect measurement of those temperatures should be assessed for normal, startup, shutdown, and other transient conditions. The ability of existing and alternative temperature sensors to provide accurate measurements for prolonged periods — 10,000 to 100,000 hr — should be surveyed. Where information is not available and is needed for safety assessment, a plan for its acquisition should be formulated. On-line surveillance methods for determining thermocouple performance should be evaluated relative to the influence that thermocouple surveillance can have on plant safety.

8.6.6 In-core flux monitoring (Task 7.0.1.2)

Scope and justification

Because proposed HTGR cores are very large, the out-of-core flux measurements do not accurately monitor local flux oscillations or peaking. In-core flux measurement capability at power, supplementing thermometry, would reduce the opportunity for undetected local overheating and fuel damage. Furthermore, the out-of-core measurements are not sufficiently sensitive for monitoring reactor status during refueling, maintenance, and startup. High-temperature in-core

neutron detectors for flux measurements in the source range could afford additional safety by allowing the reactor status during startup to be monitored. At present, shutdown is monitored by inserting a chamber after the core has cooled. Accordingly, this task will be aimed at assessing the safety needs for (1) self-powered neutron detectors to measure in-core power-range neutron flux and (2) high-temperature fission chambers for low power (startup) flux measurements.

Status

Self-powered (rhodium) neutron detectors are available commercially for moderate temperature applications.²³ High-temperature fission chambers are being developed for LMFBRs at ORNL and GE; they are planned for service at 1050°F and 10^6 to 10^7 rcentgens/hr.^{24,25}

Identified work

HTGR requirements for in-core measurements should be assessed; this includes determination of the state-of-the-art of self-powered neutron detectors and high-sensitivity fission chambers for high-temperature applications typical of HTGR needs. The study should include the characterization of the HTGR environments and operating modes in which the detectors must operate and should define the safety performance criteria of both types of detectors. Also included should be evaluation of necessary positioning devices. This study would be expected to define any safety development requirements for in-core neutron detectors.

8.6.7. Surveillance and diagnostics by noise analysis and dynamic testing (Task 7.0.1.3)

Scope and justification

Advanced surveillance and diagnostic systems could improve safety, reliability, operability, and efficiency. Noise analysis and dynamic testing methods are emerging technologies that offer the opportunity of advancing considerably the reactor surveillance and diagnostics state of the art. These techniques are used to determine the system operational characteristics, to detect incipient failures, and to characterize anomalous behavior and failures of plant components while using only regularly available plant hardware.

Characteristic noise patterns ("signatures") are associated with the operation of a reactor and its heat removal equipment. Changes in these patterns could signify developing wear of rotating machinery, weakening of vibrating parts, or changes in operating characteristics. Analysis of noise can and is being done on a variety of reactor plant process signals including temperature, flow, pressure, acoustics, and neutron flux. Results obtained indicate that useful safety information might be obtained by "noise" and dynamic analyses, and work covered in this task concerns assessment of the need for such analyses.

Status

Considerable use of noise analysis methods has been made in non-nuclear systems, particularly in the aircraft industry. Nuclear applications have been under study for the past decade, and many different applications have been reported.^{26,27} However, a comprehensive system approach has not been taken in any of the investigations to date, and the proper implementation of the technique will not be effected until this approach is taken.

Small-signal dynamic testing has been applied to several research reactors²⁸ and to several operating pressurized-water²⁹ and heavy-water³⁰ reactors recently. As in the case of noise analysis, the full benefit of this technique has yet to be realized because of a lack of systematic approach as part of the reactor operations.

Identified work

A study of typical HTGR plants should be made to determine what types of incipient failures are most important in terms of early detection and which process variables might be monitored along with their frequency ranges of interest for noise analysis. Further, an assessment should be made as to which subsystems can be monitored by dynamic testing to give useful safety information and how the tests can be implemented.

8.6.8 Reactor coolant moisture monitoring (Task 7.0.1.4)

Scope and justification

The identified potential source of moisture input to the helium coolant of the HTGR is primarily leakage from steam generator tubes. Such leakage is likely to be one of two categories: (1) pinhole leakage or (2) tube rupture or breakage. The consequences of a

pinhole leak are long term in nature; so an adequate defense is the off-line helium-impurities monitoring system. However, a tube break must be detected and identified in about 10 sec so that the defective steam generator can be isolated and depressurized to minimize the quantity of steam-ingress to the coolant. To fail to do this correctly could lead to extensive oxidation of the graphite and to an increase in primary system pressure, with possible actuation of the PCRV relief valves and consequent release of fission products to the containment. Thus, this task should assess the safety requirements associated with monitoring the moisture level in the primary coolant.

Status

Some long-term test data have been obtained at the Peach Bottom reactor. The instruments in this test are standard moisture monitors with the exception of the thermal conductivity system which is a prototype system assembled at GAC. None of these instruments could operate in the large HTGR system without modifications such as heavy-wall construction for pressure capability. The Peach Bottom operating experience has yielded valuable information on the advantages and disadvantages of the various operating principles used in the design of the instruments. The systems producing acceptable data at Peach Bottom are: (1) dew point hygrometer, (2) infrared analyzer, and (3) thermal conductivity trip unit. Most of the systems tested are somewhat complicated (i.e., employ a significant number of component parts) and are therefore subject to periodic maintenance.

Identified work

Assessments should be made to determine the detailed moisture monitoring requirements of HTGR plants from the viewpoint of safety; this includes assessing the present state of the art with respect to moisture monitoring capability. These studies should lead to a report which (1) identifies moisture monitoring requirements, (2) compiles the Peach Bottom experience³¹ with moisture monitors, both operational and developmental, (3) identifies commercial systems and discusses their applicability to HTGRs, (4) describes foreign work such as that of Bennet and Conibear,³² (5) discusses other U.S. research and development activity such as the work of Billeter,³³ and (6) identifies and discusses other concepts that have potential applicability but have not been investigated.

8.6.9 Study to identify additional safety instrumentation needs (Task 7.0.1.5)

Scope and justification

There are expected to be some areas of instrumentation where available equipment and techniques may be judged to be adequate but where additional development may add substantially to safety, reliability, serviceability, or economy. Closer scrutiny may even reveal marginal acceptability or inadequacy as detailed design and analysis progress. Because of the lack of widespread U.S. involvement in the specific safety instrumentation problems of HTGRs, an independent assessment of the current status of such instrumentation is needed along with a review of the anticipated vital safety needs. Such an assessment would be expected to identify additional areas where development is needed. Some of the subjects suggested for inclusion in the study are not now identified as relating to safety yet they have obvious relationships to overall plant safety and should properly be a part of the study. Another concern of the proposed study should be the adequacy of instruments and techniques that have been used on other reactor types and that are expected to perform adequately for HTGRs without modification, but which have not been proven in actual HTGR service.

Identified work

The principal effort of this task would be a comprehensive review of the status and requirements of all instrumentation related to the safety of the plant so that additional safety development needs can be identified. This may include further investigation of other specifically proposed tasks as well as a search to reveal unidentified problems.

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PART III

SAFETY-RELATED PLANNING GUIDE

9. FISSION PRODUCT TECHNOLOGY (TASK AREA 2)

9.1 DEFINITION AND OBJECTIVES OF TASK AREA

The potential for radiological hazard from fission products depends on the barriers separating them from the environment and on their mobility. Other reasons for studying fission-product distribution relate to their known catalysis of reactions between oxidants and carbon and to their potential effects on structural metals. To predict and evaluate the effects related to fission products, a description of their distribution and mobility under all circumstances is necessary. This description is the safety-related goal of task area 2. Since the fission-product distribution during normal operation constitutes a description of conditions at the beginning of an accident, its description has necessarily been included in this task area.

Because the failure of fuel particle coatings can have a very significant effect on fission-product mobility, the study of fuel-failure mechanisms has been included in this task area as task group 2.7.

The objectives of task area 2 include the following:

- (1) Determining of fission product, fuel, and control-material behavior relevant to HTGR performance and safety assessment,
- (2) Providing validated mathematical models, and computer codes where appropriate, for the calculation of normal fission-product distributions as functions of fuel-quality specifications, material properties, and plant operational parameters,
- (3) Providing information for the calculation of fuel-failure criteria for use in accident analyses and the assessment of operational safety margins,
- (4) Developing methods to predict changes in fission-product distributions within the plant and behavior of fission-product barriers following postulated perturbations such as those discussed in Sect. 6, i.e., steam ingress, depressurization, and core thermal transients,
- (5) Providing data for functional specifications that are needed for engineered safeguards.
- (6) Providing data applicable to the estimation of radiation exposures associated with operation of an HTGR system. (Doses considered may be received during normal operation, maintenance and repair work, accidents, and fuel reprocessing.)

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The organization of task area 2 to meet these objectives is outlined in Fig. 9.1.

9.2 SCOPE

The objectives listed above define the scope of task area 2 in general terms. However, development to date permits the scope of future work to be limited. As knowledge of HTGR accident sequences and their probable frequencies becomes better defined, there will probably be further adjustments in task scope and direction. Currently the following factors serve to limit the scope of this task area:

- (1) For various reasons -- including short half-life, low energy, deposition rate, lack of tendency to concentrate in man, involatility in the presence of graphite, compatibility with structural components, and low fission yield -- the list of important nuclides can be reduced to a few isotopes of tellurium, iodine, krypton, xenon, silver, cesium, strontium, barium, and tritium. Zirconium behavior is usually studied as characteristic of a nonvolatile and immobile element.
- (2) Carbide and oxide fuels with carbon and SiC layered coatings need to be included to reflect the choice of reference fuels.
- (3) Fission-product release studies should be confined to reference materials and the Fort St. Vrain matrix material but should be expanded as further developments are made with regard to fuel rod matrices.
- (4) Structural graphites used in fission-product transport studies should be those specified or strongly considered for use in HTGRs, except where elucidation of mechanisms requires investigation of a wider range of properties.
- (5) Primary circuit metals used in fission-product deposition studies should initially be those used in HTGRs built to date or specified for the large HTGR for metallurgical reasons (see Sect. 13). Because surface film compositions depend to some extent upon the coolant composition, investigations in which surface films play an important part will need to be wide ranging (see also Sect. 10).
- (6) All investigations involving fuel should initially be carried to at least 1600°C, graphite studies to 1400°C, and metal surface studies to 1000°C unless relevant phenomena cease to occur below these temperatures.

Fig. 9.1. Scope and objectives of task area 2 (fission-product technology).

- (7) Accident analyses performed by the vendors and others permit a definition of the conditions likely to be achieved in the current reference set of accidents discussed in Sect. 6. These analyses have been used to select initial conditions for the investigation of abnormal fission product transport.

Accident sequence analyses to be performed under task group 1.2 (Sect. 7) may create a need for fission-product transport data under as yet unspecified conditions. In this situation, a scoping study would be performed to determine whether the potential risk from the new situation warrants further study.

- (8) The development of improved fuels by addition of materials to control fission-product release, oxygen potential, or carbon transport will not be performed under the safety program. When improved fuels reach reference fuel status, they should be included in the HTGR safety program.
- (9) An important part of task area 2 are surveillance programs; these are designed to determine the applicability of laboratory-based technology relating to normal fission-product distributions on a realistic scale. This should be done, where possible, by performing validation experiments in operating power reactors or on fuels and other materials removed from reactors. Successful completion of a series of such validation tests would show that that part of the task had been completed.
- (10) Information on fission-product transport and fuel behavior will be supplied to task group 1.4 for use in accident analyses, to task group 1.2 for use in accident sequence studies and to task area 3 for assessment of catalytic and metallurgical effects. Information defining system conditions during specific accidents will be required from task area 1 along with information regarding materials properties from task areas 3 and 6.

9.3 STATUS AND NEEDS

Detailed discussions of the current state of the technology and specific needs are provided for each task group in Sect. 9. The general status of the entire area is presented here only to provide perspective for the more comprehensive treatments.

Owing to the long development history of gas-cooled reactors, both in the U.S. and Europe, large amounts of information have been accumulated with respect to the behavior of fission products. This body of information is generally agreed to be adequate for the design, construction, and operation of large HTGRs if conservative

design decisions are made. However, the available information is recognized as incomplete in many areas, so that appropriately conservative assumptions must be applied in fission-product transport analyses, as well as in the design of HTGR. Consequently, the safety margins that appear to exist in all situations involving HTGRs, although conservative, can not yet be unambiguously defined.

Accordingly there are three principal categories of need regarding the behavior of fission products. The first is for a comprehensive assessment and description of the available technology to ensure its wide availability and at the same time to help define areas of more specific needs. The second is the advancement of the technology to enhance confidence in any conclusions about the safety-related behavior of fission products. The third is for improved understanding of the failure mechanisms of coated fuel particles.

9.4 SUMMARY OF TASK GROUPS, EFFORT AND SCHEDULE

9.4.1 Task summary table and task area organization

The tasks in task area 2 have been collected into groups based upon system components.

These groups are entitled:

- 2.0 Task area need assessment
- 2.1 Fuel and control material transport
- 2.2 Fission product behavior in the coolant circuit (engineering-scale dynamics)
- 2.3 Fission product behavior in the reactor containment
- 2.4 Fission product release from fuels
- 2.5 Fission product behavior in graphites
- 2.6 Fission product behavior in the coolant circuit (surface chemistry)
- 2.7 Tritium behavior in HTGR
- 2.8 Fuel particle failure mechanisms

With the exception of task group 2.0, each group is subdivided into source, synthesis, and validation or surveillance tasks.

The source tasks provide for elucidation of mechanisms and determination of parameters. Synthesis tasks involve preparation of mathematical models and ultimately computer codes. Validation tasks test the models and data packages produced in realistic surroundings; they comprise the largest blocks of work because confirmation of time-dependent phenomena over a four- to six-year time span through post-irradiation fuel-element analysis requires an extensive effort. A tabular summary of the tasks is provided in Table 9.1 covering both safety and safety-related work.

Table 9.1. Summary of identified safety and safety-related work in task area 2
Task groups or tasks marked with an asterisk (*) are included in the safety planning guide
(Part II of this document)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task area 2 Fission product technology										
Task group 2.0 Need assessment for fission product technology										
2.1.1 Goals for fission product technology. Establishment of program goals for fission-product retention based on accident considerations. Critical review of technology. Updating as required. Coordination of reporting of results by element.	3	3	3	3	3	3	3	3	3	3
Task group 2.0 totals	3	3	3	3	3	3	3	3	3	3
Task group 2.1 Fuel and control material transport*										
2.1.1 Source tasks. Quantitative description of mechanisms by which control, fissile, and fertile materials might separate under accident conditions.										
2.1.1.1 Actinide adsorption and diffusion in graphites			1	1						
2.1.1.2 Control material adsorption and diffusion in graphite		1								
2.1.2 Synthesis tasks. Codes for analysis of movement of Th, B, and other control materials during accidents involving (a) very high temperatures and (b) steam and high temperatures.										
Task group 2.1 totals		1	1	1						
Task group 2.2 Fission product behavior in the coolant circuit Engineering-scale dynamics*										
2.2.1 Source task.										
2.2.1.1 Fission product distribution and re-entrainment. Description of aerodynamics of aerosols in HTGR with emphasis on deposition and re-entrainment in realistically complex geometries. Studies of aerosol adhesion to appropriate substrates as functions of material temperature and time.	2	2	2	2.5	2	1				

Table 9.1 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
2.2.2 Synthesis task.										
2.2.2.1 Fission product distribution code for coolant circuit. Modification of PAD code to take account of role of dust in fission product distribution in primary circuit and formulation of dust re-entrainment model for accident analysis. See also 2.5.2.1.		1	0.5	0.5						
2.2.3 Validation tasks.										
2.2.3.1 Re-entrainment experiments. Experimental determination of the mobility of solids adhering to surfaces of the primary loop Peach Bottom HTGR. Correlation with need and expectation.		2.5	1							
2.2.3.2 Fission product and dust distributions in HTGRs. Measurements of distributions of deposited fission products on steam generators, ducting, and circulators in or from Fort St. Vrain or larger HTGRs. Development of appropriate remote measurement techniques (task 2.6.3.2 will supply this information for Peach Bottom).										
Task group 2.2 totals	2	5.5	3.5	3	2	1				
Task group 2.3 Fission products in the containment*										
2.3.1 Source task.										
2.3.1.1 Review of iodine behavior in containment. Review of work done under other programs and determine need for further refinement of description of iodine behavior in HTGR containment, including concrete. Consider other fission product behavior.	0.5									
2.3.2 Synthesis task.										
2.3.2.1 Code for description of iodine behavior in HTGR containment. Recommend or develop code to describe iodine behavior in containment.										
Task group 2.3 totals	0.5									

Table 9.i (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 2.4: Fission product release from fuels										
2.4.1 Source tasks. Description of mechanisms of fission product release, with intent of providing source terms for release codes. Determination of parameters for reference fuel.						2 ^a	2 ^a	2 ^a	2 ^a	2 ^a
2.4.1.1 Release of krypton and xenon	1	1								
2.4.1.2 Release of iodine, tellurium, and selenium		0.5	0.5							
2.4.1.3 Release of metallic fission products	7	7	6	5	4					
2.4.1.4 Release of actinides			1	1	1					
2.4.2 Synthesis tasks. Source terms for fission-product transport codes. Compare FIPER, FIPDIG, and other similar source terms. Formulate new models to match experimental data as necessary. Incorporate feedback from task 2.4.3.										
2.4.2.1 Kr, Xe release codes										
2.4.2.2 I, Te, Se release codes										
2.4.2.3 Metallic fission-product and actinide release codes	0.5	0.5								
2.4.3 Validation tasks. Fission-product distribution in HTGR fuel bodies. Fuel failure, fission-product release from kernels, coated particles, and matrices of Fort St. Vrain and large HTGR fuel; comparison of results with code predictions. Minimal work on Peach Bottom fuel. Fission-product distributions in relevant experimental fuel bodies as available from other programs (work on fuel from large HTGR not costed).										
2.4.3.1 Kr, Xe release vs burnup in failed fuel		1	1	1	1	1	1	1	1	1
2.4.3.2 I, Te, Se release vs burnup in failed fuel		1	1	1	1	1	1	1	1	1
2.4.3.3 Metallic fission-product release from high burnup fuels	2	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5
2.4.3.4 Actinide retention	(Effort included in task 2.4.3.3)									
Task group 2.4 totals	10.5	13.5	12	10.5	9.5	6.5	6.5	6.5	5.5	5.5

Table 9.1 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 2.5 Fission-product behavior in graphites										
2.5.1 Source tasks. Fission-product behavior in core graphites. Elucidation and description of mechanisms of transport of Cs, Ag, Sr, Ba, and Ce through structural graphites in systems with and without forced He flow. Behavior of fission products at very high temperatures. Vapor pressures of Cs, Ag, Sr, Ba, and Ce over pyrocarbon, matrix carbon, and structural graphites as functions of temperature, surface metal concentration and composition, and gas composition.						2 ^a	2 ^a	2 ^a	2 ^a	2 ^a
2.5.1.1 Iodine adsorption and diffusion in graphites	1	1								
2.5.1.1 Metallic fission-product adsorption and diffusion	5	5	4	4	2					
2.5.2 Synthesis tasks. Codes describing fission-product releases from HTGR elements. Continued development of codes of FIPER type in light of feedback from experimental work in task 2.5.3.										
2.5.2.1 Iodine transport codes	1.5	1	1							
2.5.2.2 Metallic fission-product transport codes										
2.5.3 Validation tasks. Postirradiation examination of fuel element and reflector graphite. Measurement of radial and axial distributions of I, Cs, Ag, Sr, Ce, Zr, and Ru isotopes in fuel elements and reflector graphite from Peach Bottom (driver), Fort St. Vrain and large HTGR. Comparisons of predicted behavior with observation (large HTGR not costed, Fort St. Vrain surveillance expected to cease in 1981. Contingency provided for continuation).										
2.5.3.1 Rare-gas transport in HTGR elements	0.5	0.5								
2.5.3.2 Iodine and tellurium transport in HTGR elements	0.5	0.5								
2.5.3.3 Metallic fission-product transport in Peach Bottom graphite	4	3	1	1						
2.5.3.4 Metallic fission-product transport in Fort St. Vrain graphite		2	3	13	12	12	16	14	8	2
2.5.3.5 Metallic fission-product transport in large HTGR graphite										
Task group 2.5 totals	12.5	13	9	18	14	14	18	16	10	4

Table 9.1 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 2.6. Fission products in the coolant circuit—surface chemistry										
2.6.1 Source tasks. Fission-product behavior on coolant-circuit surfaces. Surface equilibria involving steam generator metals, their oxides, and fission products in appropriate chemical forms. Effect of temperature, steam, and air. Partition of fission products between different "surface phases." Diffusion of fission products in metals					1 ^a	1 ^a	1 ^a	1 ^a	1 ^a	1 ^a
2.6.1.1 Iodine and tellurium adsorption on metallic surfaces	2	1								
2.6.1.2 Metallic fission-product adsorption on metallic surfaces		1	1	2						
2.6.1.3 Fission-product diffusion in metals		1	1	1						
2.6.1.4 Aerosol studies		0.5	0.5							
2.6.2 Synthesis task. Descriptions of fission-product adsorption and evaporation in coolant circuit. Incorporate adsorption/desorption kinetics into PAD code where necessary. Adjust in light of observation.										
2.6.2.1 Fission-product distribution codes	0.5	0.5								
2.6.3 Validation tasks.										
2.6.3.1 Fission-product deposition in Peach Bottom coolant circuit during operation. Measurement of release from core into coolant, average deposition rate between gas samples, distribution of deposited activities. Characterization of dust; determination of fission products in dust; distribution of fission products between dust on surfaces, surface films, and metal substrates. Measurement of krypton and xenon contributions to daughter activities in circuit. Comparison of observation with model predictions and model adjustment as needed.	1.5									

Table 9.1 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
2.6.3.2 Fission-product distributions in Peach Bottom coolant circuit following operation. Determination of amounts and distributions of all γ emitting nuclides in primary circuit plus ^{129}I , ^{89}Sr , and ^{90}Sr as feasible. Characterization of dust and deposits in steam generator, circulator, ducting, and hot valve. Determination of distributions of activity between deposited dust, surface films, and base metal.	0.5	5	1.5	1						
2.6.3.3 Fission-product deposition in Fort St. Vrain coolant circuit. Measurement of releases from core into coolant, average deposition rate onto steam generator, and distribution of deposited activities as feasible. Characterization of collected dust. Measurement of krypton and xenon contributions to daughter activities in circuit. Comparison of observation with PAD code prediction.	2	1.5	1.5	1	1	1	1.5	0.5	0.5	0.5
2.6.3.4 Fission-product deposition in large HTGRs. Design and install coolant samplers up and downstream of more than one steam generator. Determination of amounts and distributions of all γ emitting nuclides in primary circuit plus ^{129}I , ^{89}Sr , and ^{90}Sr as feasible. Characterization of dust and deposits in steam generator, circulator, ducting, and hot valve. Determination of distributions of activity between deposited dust, surface films, and base metal.	0.5	1						1	1	1
Task group 2.6 totals	7	11.5	5.5	5	2	2	2.5	2.5	2.5	2.5
Task group 2.7 Tritium behavior in HTGRs										
2.7.1 Source tasks.										
2.7.1.1 Tritium adsorption on graphite. Measurement of ^3H adsorption and/or exchange with ^3H on graphite.			1	1						
2.7.1.2 Tritium permeation through metals. Measurement of ^3H permeation through steam-generator materials	0.5	0.5								
2.7.1.3 Tritium release from fuels, recoil sites in graphite and boron carbide. Determination of source functions describing release of ^3H from bare kernels, coated particles, and recoil sites in graphite and boron carbide for use in TRITGO code.	1	1								

Table 9.1 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
2.7.2 Synthesis task.										
2.7.2.1 Codes describing tritium behavior in HTGR. Continued development of code describing ^3H formation, distribution, and release in HTGR.	0.5									
2.7.3 Validation tasks.										
2.7.3.1 Tritium in Peach Bottom coolants. Determination of steady state concentrations of ^3H and $^3\text{H}/^1\text{H}$ ratio in primary and secondary coolants. Determination of sizes of hydrogen sinks and effects of $^3\text{H}/^1\text{H}$ ratio on steam generator permeability. Measurement of Peach Bottom steam generator average permeability. Measurement of ^3H concentrations in waste streams, purge stream, and inlet to purification system.	1.5									
2.7.3.2 Tritium in Peach Bottom fuel, graphite, and control rods. Measurements of ^3H concentrations in fuel, fuel element, and moveable reflector graphite and control rod materials.	0.5	0.5								
2.7.3.3 Tritium in Fort St. Vrain coolant. Determination of ^3H concentrations in primary and secondary coolants, in all waste streams, and in entry and exit of purification system.	1	0.5	0.5							
2.7.3.4 Tritium in Fort St. Vrain fuel, graphite, and control materials. Measurement of ^3H concentration relative to initial ^{235}U , ^6Li , and ^{10}B concentrations in fuel, fuel element, and moveable reflector graphite and control rod materials.			1		1		1			
2.7.3.5 Tritium in large HTGR coolant. Determination of ^3H concentrations in primary and secondary coolants, in all waste streams, and in entry and exit of purification system.								1	0.5	0.5
2.7.3.6 Tritium in large HTGR fuel, graphite, and control materials. Measurement of ^3H concentration relative to initial ^{235}U , ^6Li , and ^{10}B concentrations in fuel, fuel element, and moveable reflector graphite and control rod materials.										0.5

Table 9.1 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 2.7 totals	5	2.5	2.5	1	1		1	1	0.5	1
Task group 2.8 Fuel particle failure mechanisms						2.5 ^a	2.5 ^a	2.5 ^a	2.5 ^a	2.5 ^a
2.8.1 Source tasks.										
2.8.1.1 Mass transport in temperature gradients. Identification and description of carbon, silicon, and carbide coated particles. Determination of P_{CO} and $P_{(Kr+Xe)}$. Determination of parameters needed to calculate core thermal limits.	2	2	2	1	1					
2.8.1.2 High temperature isothermal failure mechanisms. Determination of conditions required to fail various types of coated fuel particles by high-temperature annealing following irradiation.	2	2	2	1	1					
2.8.1.3 Chemical compatibility. Determination of factors governing compatibility of HTGR fuels with coating materials after extensive burnup. Derivation of limiting conditions.	3	4	2	2	1					
2.8.2 Synthesis task.										
2.8.2.1 Formulation of fuel-failure criteria. Formulation of mathematical descriptions of chemical failure processes in oxide and carbide fuels for calculation of core working limits. Incorporate CO production terms into STRESS II and other similar codes. Update as needed.	0.5	0.5	0.5	0.5						
2.8.3 Validation tasks.										
2.8.3.1 Postirradiation examination of fuels from capsules. Measurements of kernel migration rates, internal gas pressures, and compositions and observation of kernel-coating compatibility in fuels from capsules.	0.5	0.5	0.5	0.5	0.5					
2.8.3.2 Special irradiation capsule for mass transport studies. Design, construction, operation, and examination of capsule for "amoeba effect" mechanism studies.	0.5	0.5	2	2	2					

Table 9.1 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
2.8.3.3 Postirradiation examination of Fort St. Vrain fuel. Determination of kernel migration coefficients, fraction of failed SiC layers, and composition and pressure of gas in intact particles. Observation of kernel-fuel compatibility. Scale of examination to be re-assessed in light of realistic objectives. Provision of data for revision of core operating limits and fuel specifications.		1	2	2.5	2.5	1	1	1	1	
2.8.3.4 Postirradiation examination of large HTGR fuel. Determination of kernel migration coefficients, fraction of exposed fuel, fraction of failed SiC layers, and composition and pressure of gas in intact particles. Observation of kernel-fuel compatibility. Scale of examination to be re-assessed in light of realistic objective. Provision of data for revision of core operating limits and fuel specifications.									1	1
Task group 2.8 totals	8.5	10.5	11	9.5	8	3.5	3.5	3.5	4.5	3.5
Task area 2 totals	49	60.5	47.5	51	39.5	30	34.5	32.5	26	19.5

^aMinimal effort to maintain nominal capability.

One reason for the large number of tasks is rapid progress of the technology which has presented designers of HTGR cores with a number of apparently equally attractive options, e.g., three significantly different types of coating (LTI, HTI, SiC) and a large number of fuel and graphite formulations.

9.4.2 Schedule

Table 9.1 may be read as a time chart. However, some discussion of scheduling is necessary. First, the program schedule is strongly influenced by external events and interim results. External events also dictate the scheduling of validation tasks involving experiments in power reactors. A mistimed effort can lose what may be unique information. For example, information from the decommissioning of Peach Bottom is only available between 1974 and 1978 (task 2.2.3.1).

Figures 9.2 through 9.6 show logical progressions for work in all tasks on important groups of fission products (see Sect. 9.5 for justification of grouping).

In these diagrams, a horizontal line represents effort, a circle a decision, and a vertical dashed line intersecting horizontal effort lines an interaction. Only four basic decisions are considered: (1) START; (2) REVIEW (and adjustment), implying a review of progress relative to goals that can change; (3) REVIEW AND REDUCE R&D, implying maintenance of validation and consulting capability only after source and synthesis tasks have apparently achieved their goals; and (4) STOP, which signals writing the final report and the cessation of surveillance. There is no limit to the number of reviews so that program emphasis can change to meet new externally generated goals (e.g., from tasks in hexagonal boxes or from licensing processes) or internally generated goals (task 2.0). No specific times are shown. Relative time is implied by the length and position of horizontal lines.

In general, surveillance programs aimed at validating normal fission-product release from the core would be expected to run until the core composition is relatively constant, i.e., for at least six years for Fort St. Vrain and four years for the first large reactor. Surveillance is recommended in subsequent plants if fuel or graphite temperatures increase more than 100°C. Thus, these programs should extend beyond 1983.

Monitoring for longer term effects, such as consequences of changes in plateout factors, should continue for the life of the plant or until there is resolution of significant uncertainties.

Fig. 9.2. Effort for krypton and xenon studies.

Fig. 9.3. Effort for iodine studies.

Fig. 9.4. Effort for studies of metallic fission products.

Fig. 9.5. Effort for tritium studies.

Fig. 9.6. Effort for fuel-particle failure studies.

Specific schedule needs

Noble gases. Figure 9.3, the logic diagram for rare gas studies, shows no conditional tasks. All identified tasks could proceed in parallel. No special equipment that does not exist is needed for surveillance in Fort St. Vrain, so there is negligible lead time.

Iodine, tellurium, selenium. For ^{131}I (Fig. 9.4), goals can be defined for normal operation. Studies of the probable frequency of accidents in which adsorbed iodine and steam are mixed if the primary circuit is breached (task group 1.2) could change the permissible core releases of ^{131}I and hence tellurium. This would lead to changes in scheduling of iodine distribution studies.

Further work in source and synthesis tasks aimed at improved definition of core release parameters and iodine distribution (source task 2.2.1.1) should be deferred until the value of greater precision has been demonstrated.

Tasks 2.6.3.1 and 2.6.3.2 involve lead times of one year associated with planning the decommissioning of Peach Bottom.

Surveillance in a large HTGR has a very long lead time since instrumentation and access points need to be provided. Post-irradiation examination of Fort St. Vrain blocks needs at least a year for equipment development.

Surveillance in Peach Bottom (tasks 2.6.3.1 and 2.6.3.2) and adsorption studies aimed at defining circuit capacity for iodine are judged to have highest priorities.

Nonradiological restrictions on tellurium release need to be considered.

Metallic fission products. Information on fission-product release from BISO fuels with LTI coatings is needed early and will be obtained by study of release processes (tasks 2.4.1.3, 2.5.1.2, and 2.5.2.2) (Fig. 9.4). The priority of effort on cesium behavior cannot be assessed since goals relating to control of cesium primary circuit inventory are undefined; the priority is thought to be higher than those for other metals.

The need and priority of circuit distribution studies (task 2.6.1.2) will depend on the outcome of such studies.

Long lead times are associated with decommissioning Peach Bottom, surveillance in large HTGRs, and any work found to be needed on dust dynamics (tasks 2.6.1.4 and 2.6.1.5).

Strontium release from the reactor fuel can be large for a reactor with secondary containment. Actions that would reduce cesium release could also reduce strontium release.

Tritium. Tritium release studies will benefit increasingly from AEC-funded fusion research. Studies of tritium adsorption in graphite should be started only after a demonstration of need by surveillance in the Fort St. Vrain reactor. Resolution of problems addressed by tasks 2.7.1.2 and 2.7.1.3 are more important because they have greater uncertainties. Tritium surveillance activities are not expected to have long lead times (Fig. 9.5).

Fuel. Though the duration of this task may be affected by changing fuel-quality requirements, all tasks may proceed in parallel (Fig. 9.6).

Long lead times will be involved in any special irradiation experiments. Improved thermometry (task area 7) is an essential prerequisite to more precise in-pile experiments.

9.4.3 Use of existing facilities

A number of existing facilities will be used to implement the proposed program. Their utilization in the HTGR program will improve return on investment in these facilities and substantially reduce the program's capital equipment requirements.

Fission-product and fuel-failure studies could utilize the HFIR and ORR reactors at Oak Ridge for fuel irradiations under accurately defined conditions. GAC expects to make substantial use of its TRIGA reactors to study fission-product release. Hot cells in both laboratories could be used for fuel and fuel-element examinations, reactor component examinations, and studies of fission-product and plutonium transport. Their adequacy will depend on program goals.

Advantage should be taken of the decommissioning of the Peach Bottom HTGR to examine fuel and components. This exercise would conclude a long period of experimental work at Peach Bottom. The Fort St. Vrain reactor and other larger PTGRs should be used in validation studies to the maximum practicable extent.

9.5 NEED ASSESSMENT (TASK GROUP 2.0)

9.5.1 Justification and scope

This task group objective is to ensure that specific safety needs in the area of fission-product technology are identified and that any required technology is developed in a timely manner. The objectives and scope of this task group are those of the whole task area as defined in Sects. 9.1 and 9.2. In addition, this task group should ensure that the needs of task area 1 are met and that information describing fission-product behavior such as temperatures, flows, material descriptions, and accident sequences, provided by other tasks, are factored into continuously reviewed assessments of the need for further work.

9.5.2 Status

Figure 9.1 illustrates the technical objectives of task area 2 and the way information is expected to flow. The quantitative specification of these objectives is based upon estimates of radiological effects associated with fission-product release and known deleterious effects of fission products on materials.

The paths that fission products can take to the coolant circuit and the environment are shown in Fig. 9.7 and are designated by τ . Each component or capacity (for example, the coolant circuit walls that can adsorb certain fission products) is identified by a reference number. This permits transfers, designated as τ , between capacities to be identified by two indices. A third index, omitted in Fig. 9.7 for convenience, is needed to define the isotope being transferred. Thus $\tau(I, J, K)$ describes the rate of transfer of isotope I from J component to K component per unit amount of I in J. The transfer functions depend on temperature, medium, and concentration.

One way to identify fission-product behavior needs is to calculate fission-product transport based on conservative estimates of transfer functions and to compare the result, thus obtained, with established goals. In many cases there is some latitude relative to the permissible values of specific transfer functions to meet an overall objective of fission-product containment. This information will facilitate practical evaluation of fission product needs. Engineering solutions are often adopted in preference to data refinement programs because their implementation is faster. Incentive for data refinement may, however, still exist.

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Fig. 9.7. HTGR system components controlling fission-product transport to environment.

Sets of fuel specifications based upon radiological and economic criteria have been published by the Dragon Project^{1,2} and the UKAEA.³ Uncertainties in data and accident sequence were recognized. Permissible coolant circuit inventories can also be deduced from accident analyses presented in the Summit Station PSAR.⁴ No one set of analyses can, however, be regarded as exhaustive or final. Fuel-release specification guides are being developed by ORNL.

9.5.3 Identified work

Fuel release limits should be estimated for ^{88}Kr , ^{131}I , ^{137}Cs , ^{90}Sr , and other nuclides as necessary, for normal and accidental release. The Summit site should be used as a "reference site." Calculations using appropriate codes should be performed to help define the following:

- (1) goals for circulating and plated-out inventories in the coolant circuit,
- (2) goals for release rates from the core - $\tau(4,5)$,
- (3) goals for release characteristics of the fuel - $\tau(1,2)$,
- (4) goals for fuel failure criteria.

These goals should be updated as necessary. A state-of-the-art review and a data file describing fission-product transport mechanisms should be produced.

9.6 FUEL AND CONTROL MATERIAL TRANSPORT (TASK GROUP 2.1)

9.6.1 Justification and scope

The justification for the work in this task group is the desire for further knowledge of the behavior of core materials under Class IX accident conditions, in which sustained temperatures above normal are postulated. This task group will develop descriptions of the transport of uranium, thorium, and control materials in a reference core at temperatures in excess of 3000°C. Task area 1 will perform the necessary thermal and nuclear analyses.

9.6.2 Status

An analysis of boron and uranium behavior in graphite at high temperatures was made during the licensing of Fort St. Vrain. The applicability of that analysis to postulated Class IX accidents in other HTGRs has not been established.

9.6.3 Identified work

Actinide and control material transport in graphite (Tasks 2.1.1.1 and 2.1.1.2)

Fertinent data should be assessed to determine the effects of very high temperatures on the relative distribution of uranium, thorium, boron, and perhaps important heat-producing nuclides such as protactinium; and the need for further information evaluated. This evaluation should include a suitable overall analysis (possibly in Task area 1) of a Class IX accident in the context of an existing or proposed HTGR. Since further effort depends on the results of the scoping study and need assessment, costs beyond the first year have not been estimated.

9.7 FISSION PRODUCTS IN THE COOLANT CIRCUIT — ENGINEERING SCALE DYNAMICS (TASK GROUP 2.2)

9.7.1 Justification and scope

Reference to Fig. 9.1 shows that the coolant circuit occupies a key position in the path of fission products from the fuel to the environment. It is the repository of almost all fission products leaving the core and the most available source of fission products, should the primary helium containment be breached. Description of the radiological consequences of postulated accidents in which primary containment is breached therefore requires either assurance that displacement of 100% of this deposited primary circuit inventory is acceptable or means of calculating actual displacement of iodine, tellurium, cesium, and strontium isotopes.

The conditions of interest in this task group are (1) changes in the direction and/or magnitude of the coolant flow, (2) changes in coolant composition from helium to helium plus steam, and (3) rapid changes in coolant pressure.

Under normal operating conditions, the principal objective is calculation of the deposition ratio,* for reasons discussed in

* Deposition ratio (I) = $\frac{(\text{Curies of I on surfaces})}{(\text{Curies of I in gas})}$

Sect. 9.1.8 (Task group 2.3). However, in a given accident both the amount of activity displaced and the potential dose to reactor maintenance workers will depend on the normal distribution of gamma emitters such as ^{137}Cs and ^{131}I as well as on the circuit inventory. Investigation of cesium and iodine distribution is therefore necessary if significant departures from uniform distributions are encountered. The scope of task 2.2 therefore includes the description of fission-product re-entrainment under accident conditions and fission-product distribution under normal conditions. Close integration will have to be maintained between this task group and related studies dealing with the deposition of fission products.

9.7.2 Status

The dynamics of fission-product deposition and re-entrainment depend strongly upon the molecular weight of the fission-product carrier and whether contact with loop surfaces leads to reversible or irreversible sorption. Mathematical models based upon analogies between heat and mass transfer exist for predicting distributions of molecular species in thermally and geometrically complex systems. Loop experiments performed at Battelle Memorial Institute,⁵ GAC, Harwell, ORNL, and KFA Jülich, and reactor experiments associated with Dragon, AVR, and other gas-cooled reactors have been helpful in validating this approach.

In the situation where a fission product is carried by a relatively heavy dust particle, both deposition and re-entrainment are difficult to describe and validated models are lacking. The need for further data when secondary containment is provided remains to be determined.

Characterization of HTGR dust is in progress, and attempts are being made to modify the fission-product distribution code PAD to represent distributions measured in Peach Bottom where dust plays a significant role. The PAD code,⁶ which has been recently developed at GAC, based on earlier work at ORNL and Battelle Memorial Institute, is used for describing the distribution of fission products in the HTGR primary coolant loop. It takes into account gaseous precursors and does not require the assumption of an infinite sink of fission-product deposition. It does not consider diffusion into circuit walls and may therefore misrepresent behavior at higher temperatures. The code has been used to compare predicted and measured activity distributions in the Dragon coolant circuit and in laboratory experiments. More refined quantitative comparisons need to be made. A code that considers diffusion into the duct wall is being developed at KFA Jülich to assist in gas-turbine accessibility studies.

9.7.3 Identified work

Fission-product distribution and re-entrainment (Tasks 2.2.1.1, 2.2.2.1, 2.2.3.1, and 2.2.3.2)

In experimental studies in this area, difficult problems are associated with guaranteeing the correctness of chemical species, of surface composition, and of geometry. Extrapolation of scale is also a problem for dusty systems where theory is not adequate. Because of this, experimental studies are preferred that involve in-pile loops or experiments like VAMPYRE at KFA Jülich in which an HTGR is used as a source of hot helium containing dust, fission products, and impurities. Fission-product distributions in HTGRs should therefore be measured when opportunities occur.

Fission-product distributions should be measured in all parts of the Peach Bottom primary coolant circuit during decommissioning, and opportunities should be sought to measure distribution as well as deposition ratios in Fort St. Vrain. Since the Fort St. Vrain steam generators were not designed with this in mind, formidable experimental problems may be encountered.

Surveillance of reactor operation is not likely to yield definitive information about re-entrainment under abnormal conditions. For this reason task 2.2.1.1 emphasizes dusty systems for source studies, though use of reactors to prepare specimens for re-entrainment experiments would be desirable. The decommissioning of Peach Bottom will yield an ample supply of such specimens. A large-scale loop (not necessarily in-pile) may be necessary if engineered safeguards to prevent higher-than-normal velocities during a depressurization are not deemed adequate or if low re-entrainment fractions have to be validated. Such a loop would permit simultaneous investigation of both distribution and re-entrainment models.

Integral experiments should be complemented by studies of dust adhesion as a function of material and thermal history (task 2.2.1.1) and by adsorption, solubility, and diffusion studies.

Improvements should be made to the PAD code to allow for the presence of dust and to take into account the diffusion of fission products into metallic walls at higher temperatures. An improved mathematical model for re-entrainment may also be required to reflect differences in dust deposition conditions and thermal history around the primary circuit.

9.8 FISSION PRODUCTS IN THE SECONDARY CONTAINMENT (TASK GROUP 2.3)

9.8.1 Justification and scope

This task group will provide information relevant to the design of air-cleaning systems for the secondary containment.

The containment resembles the primary coolant circuit in that the gas-borne concentration is determined by transfer between purification systems $\tau(10,11)$ and $\tau(13,16)$, deposition of various species on walls $\tau(10,12)$, and leakage $\tau(10,16)$.

The principal concern for HTGRs in this task group is believed to be the rate at which iodine can react in the containment to produce compounds that are hard to filter or aerosols that are easy to filter. The effective molecular weight and chemical form of iodine discharged determine its deposition velocity in the environment and hence its effect on man.

9.8.2 Status

The behavior of fission products in water-cooled-reactor secondary containments is being evaluated by Battelle Pacific Northwest Laboratories (PNL).

The experiments and mathematical models being reviewed cover wet and dry systems containing air and helium. However, the concentrations used are much higher than those to be expected during HTGR accidents. A specific evaluation of how iodine might be expected to behave in an HTGR containment is also being made separately by PNL to determine the need for further work.

Fission product retention in concrete

Some preliminary work has been completed on krypton, xenon, and iodine delay in packed beds of granulated concrete. Also, studies of iodine-vapor and carbon-dust transport through concrete specimens provided with a crack have been carried out. The preliminary, unpublished data confirmed that xenon and krypton do not sorb and are not measurably delayed on concrete. Iodine and dust transport, however, is attenuated quite markedly by retention in concrete; attenuation of CH_3I has not been demonstrated. Evaluations of the utility of these observations for computing attenuation through all likely PCRV liner and body leak paths are being made.

9.8.3 Identified work

Evaluation of need for further work (tasks 2.3.1.1 and 2.3.2.1)

The review performed by PNL should be used to determine whether useful safety-related gains are to be made from further study. The attenuation of CH_3I by reactions with water in the concrete should be determined.

9.9 FUEL AND FISSION PRODUCT RELEASE FROM FUELS (TASK GROUP 2.4)

9.9.1 Justification and scope

This task group determines transfer functions and capacities within the fuel rod or compact and in any gap between it and fuel-free structural graphite.

Fission-product nuclides selected for study should be primarily $^{129\text{m}}\text{Te}$, ^{131}I , ^{85}Kr , ^{137}Cs , ^{90}Sr , ^{140}Ba , and ^{95}Zr , although occasionally other isotopes of these elements should be chosen for convenience. These nuclides have been selected after considering the half-life, radiological toxicity, fission yield, and known behavior of all fission products in the HTGR.

To a first approximation, they are representative of the behavior of groups of chemical elements with different affinities for carbon. For example, krypton and xenon are characterized by their low boiling points and by their interaction with carbon being limited to Van der Waals' forces. Iodine, tellurium, and other precursors of the noble gases share with krypton and xenon an inability to form carbides, but they are significantly adsorbed by carbon and are capable of reaction with some of its impurities and with many metallic fission products.

At the opposite end of the volatility scale lie those metals which are essentially involatile and immobile in carbon at normal HTGR temperatures. Zirconium is the most common fission-product isotope in this class, which includes uranium and thorium. Other metals including cesium, strontium, barium, and europium readily form carbides which have significant vapor pressures at 1000°C . Probably for this reason, they are mobile in graphite.

Additional special nuclides or elements, e.g., $^{110\text{m}}\text{Ag}$ or tellurium, should be examined as required.

Principal justifications for this work are:

- (1) A better knowledge of $\tau(2,3)$ and $\tau(1,3)$ (Fig. 9.7) is needed to perform normal release calculations and to analyze the consequences of accidents in which fuel temperatures rise or fuel coatings fail. Definition of the value of $\tau(1,2)$ for strontium and $\tau(2,3)$ for cesium and strontium would also aid in the choice between Biso and Triso coatings of the ThO_2 particles.
- (2) The release of fission products during accidents depends directly on the primary circuit inventory. For cesium and iodine, $\tau(1,2)$ and $\tau(1,3)$ are important in determining release from the core and hence the primary circuit inventory.
- (3) The limits of useful performance of SiC are being approached at the maximum temperatures expected during normal operation and will be exceeded during accidents involving a rise in fuel temperature. This could affect $\tau(2,3)$, possibly with permanent results.
- (4) Introduction of new fissile or fertile compounds will necessitate a reexamination of actinide transport rates in pyrocarbon.

9.9.2 Status

Gas release

Gases, with the possible exception of hydrogen (and therefore tritium), do not permeate pyrolytic carbon under normal reactor conditions. Gas release is therefore a sign of coating failure or contamination of outer coatings by uranium or thorium.

Fuel contamination and its relationship to fission-gas release have been studied. Levels of fuel contamination on the surface and in the outer pyrolytic carbon layers of Triso-coated particles can be directly related to the fractional releases of fission-product nuclides.

Gas release from fuel kernels determines release from the fuel body when particles with failed coatings are present and the level of fuel contamination is relatively low; it also establishes the pressure in an intact coated fuel particle.

Considerable information exists on the release of fission gases from fuel materials; however, there are still gaps in understanding some specific phenomena. Fission-gas release data measured at temperatures below about 800°C generally exhibit a lack of dependence on temperature and on half-life, indicating a process

of recoil release in which some of the fission-gas atoms that are recoiled out of the fuel material essentially stop in the gas phase (or in a position of low sorptive energy) and migration of the atoms to the primary coolant is not appreciably delayed. On the other hand, data measured at temperatures above about 800°C generally exhibit both temperature and half-life dependence, indicating that the release is controlled by a diffusion process. However, it is not clear why, under certain conditions, krypton and xenon isotopes are released at the same rate in the range 800 to 1200°C.

Studies have shown that the fractional releases of radioactive fission gases from failed fuel particles are substantially less than one. For example, the fractional release of ^{85}mKr from failed coated carbide fuel particles at 1100°C was shown⁷ to be $\sim 5 \times 10^{-3}$. Stable rare gases are completely released by kernels in fuel cycle times. Unadsorbed gases are not significantly delayed by fuel matrices.

The primary coolant system activities of noble gases, halogens, other volatile elements, and nonvolatile elements that are assigned a conservative release are calculated using the PAD code, which was developed at GAC. The code accounts for the fission yield, release, decay, and clean-up of each member of the appropriate fission-product chain.

Fission-product metal release

Existing information on the release of metallic fission products from coated fuel particles in HTGR systems shows that (1) the release of metallic fission products from coated fuel particles is diffusion-controlled and is burnup- and strongly temperature-dependent, (2) silicon carbide coatings are much more effective than pyrolytic carbon coatings in retaining certain metallic fission products at normal reactor times and temperatures, (3) pyrolytic carbon is more effective in retaining cesium than in retaining strontium or barium, (4) HTI pyrocarbon retains cesium better than LTI pyrocarbon, and (5) the refractory fission-product metals such as zirconium are found outside coated particles only as a result of fuel contamination.⁸⁻¹⁰

Better understanding of the release of fission-product strontium, barium, and cesium from fuel particles is vital in defining the release of these metals to the primary coolant of HTGRs because release to the coolant can increase exponentially with release from the fuel particles. Current FIPER source terms neglect all retention of cesium by the kernel and strontium by the coating as a result of lack of information.

Work of a preliminary nature has been performed in the U.S. and in Germany to investigate the use of chemical getters (e.g., magnesium fluoride, silica and aluminum oxide) for reducing the release of the fission product metals such as cesium and strontium.¹¹ Alternatives to silicon carbide need to be re-evaluated as metallic fission-product barriers.

The retention of fission products by the fuel rod matrix or binder carbon is greatest for the less volatile fission products. Experiments covering retention of fission products by fuel rod matrices are in progress.

An important use of fission-product metal release data is to define source terms for the FIPER and SORS codes, which are used in safety analyses to calculate the release of fission-product metals into the helium coolant of HTGR systems. These source terms are in need of improvement and validation, particularly for fuels with all-carbon coatings.

The transport of actinides in pyrocarbons has been studied and reviewed to 2200°C by Evans.^{12,13} Information on behavior at higher temperatures is available from the nuclear rocket program.

9.9.3 Identified work

This work relates to determination and/or confirmation of transfer coefficients and capacities in the fuel body (see Fig. 9.7). Within a task, items are arranged in order of decreasing importance. This work is closely related to the base program studies of methods for fuel inspection, since the aim is to couple estimates of the fraction of failed fuel or the fraction of exposed fuel with release rate to obtain $r(1,3)$, the release from a failed fuel particle as distinct from an uncoated particle.

Based upon the justifications advanced above and within the scope defined in Sects. 9.1 and 9.9.1, the work described below is identified.

Release of noble gases from fuel (tasks 2.4.1.1, 2.4.2.1, and 2.4.3.1)

- (1) Qualification of reference experimental methods for determining the fraction of broken coatings and the fraction of exposed fuel in a fuel stick, irrespective of burnup, is needed.

- (2) Completion of experimental work needed to define $\tau(\text{Xe}, 1, 3)$ and $\tau(\text{Kr}, 1, 3)$ as functions of burnup, temperature, and fuel type is recommended. Fuel type in this context may include grain size and/or other fabrication variables.
- (3) When plutonium is used in coated fuel particles, experiments to define the conditions under which plutonium diffuses slowly in the coating will be necessary (task 2.4.1.4).

No further work is anticipated on codes describing rare-gas release from intact or failed fuels. Deficiencies are currently believed to be deficiencies in parameter values, not in models.

Some work on modeling the release of gases into the buffer volume of an intact particle $\tau(1, 2)$ may however be required in conjunction with tasks 2.8.2.1 and 2.8.1.2.

Validation (task 2.4.3.1) of models and data will be sought experimentally using fuel rods from the Fort St. Vrain reactor and irradiation capsules. Measurements made under steady-state conditions in a reactor will be used where possible, but transient hot-cell annealing experiments, which can provide data on releases of other fission products and on release during accident transients, will also be carried out.

Release of iodine, tellurium, and selenium from fuels (tasks 2.4.1.2, 2.4.2.2, and 2.4.3.2)

Work should only be re-opened if the currently accepted conservative assumptions — that $\tau(x, 1, 2)$ and $\tau(x, 3, 4)$ are equal to corresponding transfer functions for rare gases — are found to be unacceptable. Validation experiments using high burnup fuels and measurements in the Fort St. Vrain reactor will be used to test this assumption. The same hot-cell annealing experiments will yield data for many fission products (see task 2.4.3.1).

Release of metallic fission products from fuels (task 2.4.1.3)

Experiments should concentrate on cesium and strontium behavior in BISO coatings, because at normal temperatures metallic fission-product release from fuels with silicon carbide barrier layers is almost entirely a function of the mechanical integrity of the barrier.

On these bases the following work is identified:

- (1) $\tau(\text{metal}, 2,3)$, $\tau(\text{metal}, 1,2)$, and $\tau(\text{metal}, 1,3)$ in Biso $\text{ThO}_2/^{233}\text{UO}_2$ fuel in which the initial concentration of ^{233}U is zero should be experimentally determined using reference fuels.
- (2) $\tau(\text{metal}, 1,2)$ for $(\text{U,Th})\text{O}_2$ fuels should be experimentally determined by analysis of the metal in the inner coatings of Triso fuels using reference coatings.
- (3) All data relating to cesium and strontium release from particles should be reviewed.
- (4) The useful performance limits of SiC coatings and the rate at which they become permeable at high temperature should be determined in conjunction with task 2.8.3.3).
- (5) Factors affecting retention of cesium and strontium in fuel rod matrices should be evaluated, to be followed by experiments if necessary. Currently only validation experiments under task 2.8.3.3 are visualized.

Actinide release studies (task 2.4.1.4)

Re-evaluation of actinide migration rates in pyrocarbon and silicon carbide will be needed if normal operating temperatures exceed 1600°C for significant times, if plutonium compounds are used as fuels, or if accidents in which temperatures exceed 2000°C are investigated.

Metallic-fission-product release codes (task 2.4.2.3)

Improved phenomenological and mechanistic mathematical representations of metallic-fission-product release should continue to be sought. While initial emphasis should be on phenomenological source terms for the FIPER code, mechanistic studies should be pursued as a necessary adjunct to development of safer fuels.

Codes that represent the behavior of fission products in pyrocarbon are expected to describe actinide diffusion also, although parameter values will be very different.

Validation experiments (task 2.4.3.3)

There are widely different temperatures in a fuel rod. Establishment of $\tau(3,4)$ and $\tau(2,3)$ for a fuel rod therefore requires confirmation that the releases of various fuel particles as a function of temperature can be correctly summed or averaged. If phenomenological release constants are used, differences in fuel rating must be shown to be discounted correctly. Failure to do this will cause difficulties in relating the results of laboratory annealing experiments and fuel surveillance.

An integral value of $\tau(3,4)$ averaged over a fuel rod should be determined by measuring releases from a fuel rod into adjacent graphite and apportioning any release from the graphite into other components of the experiment. For early HTGRs, $\tau(4,5)$ is expected to be small compared with $\tau(3,4)$ for strontium and perhaps cesium. When $\tau(3,4)$ is $>20\%$, it may be reliably determined by measuring loss from the fuel rod or even the individual particle.

Experimental capsules in which all fuel is the same, and all components can be analyzed, provide the only reliable way of measuring small fission-product releases ($<20\%$) when the released material is not retained by the graphite surrounding the fuel rod, either because it is too hot or because the fission product does not have adequate affinity for carbon.

Annealing of irradiated fuels at reactor temperatures yields differential values of $\tau(3,4)$ for a given burnup, provided the effect of the change in rating from that in the reactor to zero can be adequately (or conservatively) discounted.

Matrix capacity should be measured by deconsolidation of fuel rods and analysis of fission products in the matrix debris. The absence of failed fuel must be demonstrated by ^{235}U analysis of matrix liquor.

Behavior of fuels at temperatures above 1500°C should be investigated entirely by annealing of fuel rods or single particles at various constant temperatures and in ways that simulate stereotyped accident histories. In-pile experiments in confirmation of earlier ORNL work and current French (GAC-inspired) work^{14,15} are not proposed under this plan. This latter group of experiments would be closely related to those performed under task group 2.8 to determine fuel coating failure criteria.

9.10 FISSION PRODUCT BEHAVIOR IN GRAPHITES (TASK GROUP 2.5)

9.10.1 Justification and scope

The scope of this task includes all fission products (except tritium) and actinides; it also concerns all structural graphites, but not pyrocarbons.

The important fission products are iodine, tellurium, cesium, silver, strontium, barium, and cerium. The mobilities of metallic fission products in this group are significantly uncertain. Important actinides are uranium, thorium, and plutonium. The only control material considered so far is boron.

The range of conditions to be covered will include radiation and neutron doses and fluxes equivalent to those experienced in service. Experiments with fuel-element graphite will be carried out from 800°C to temperatures at which the relevant transport process is sufficiently fast for limiting assumptions, such as 100% release or attainment of homogeneous distribution in the appropriate time scale, to be valid without economic penalty.

Graphite fuel-free zones between fuel and coolant constitute a secondary barrier for HTGR fuels. Like the primary pyrocarbon barrier, it is not uniformly effective for all fission products, but the pyrocarbon and graphite have complementary properties. Under normal conditions and during accidents in which power and cooling are maintained, it is cooler than the fuel coating. This and its greater thickness delay metallic fission products usefully. The fuel-element graphite therefore plays two important safety-related roles. First, depending upon the fuel quality, it can determine the size of the relevant fission-product inventory in the coolant circuit. Second, its own capacity for fission products means that it can be a fission-product source during a thermal transient or during an accident in which graphite is oxidized.

Reflector graphite may function as a low-temperature sink for fission products (e.g., iodine) transported to it by the coolant.

Description of these roles requires determination of transport and adsorption controlling parameters over a wide range of conditions. These parameters are combined in $\tau(4,5)$ and $\tau(5,4)$.

9.10.2 Status

Transport studies

The fuel-element graphite, as a barrier to fission-product metals, is more effective than the matrix and, in some cases, the particle coating, because to enter the coolant every metal atom must travel a distance greater than 6 mm through the graphite web.

Short-lived gases (with half-lives less than about 1 min) are also delayed enough to cause a significant decay in 6 to 10 mm of graphite in the absence of forced convection.

Iodine is partially retained by graphite below 900°C. The reasons for this retention are not clear. Published adsorption coefficients for iodine in graphite are too small to account fully for the observed retention. However, retention is known to be enhanced by the presence of metallic elements in the graphite. At present, iodine, bromine, selenium, and tellurium are generally treated like a xenon isotope of similar half-life. Further studies may permit taking credit for the retention of these elements in graphite.

Barium-140, whose daughter ^{140}La has a high gamma energy, can be shown not to reach the coolant under normal conditions because it decays in the graphite. Concern over the effects of high concentrations of stable barium on the diffusior of ^{140}Ba have lessened with the increased precision of measurements of diffusion coefficients for barium in graphite as a function of barium concentration. Stable barium catalyzes the steam-graphite reaction.

Silver, cesium, and strontium diffuse at significant rates through graphite — $\text{Ag} > \text{Cs} > \text{Sr}$. Other fission-product metals are generally not of concern (except perhaps europium) because (1) they have low yields or short half-lives or (2) they are essentially immobile in graphite at normal reactor temperatures.

The diffusion of cesium and strontium in graphite has been studied over many years. Silver transport has only been studied since fuels in which plutonium fissions occur have become commercially interesting. Work to date has provided diffusion coefficient data and has progressed to the point where the different modes of transport in the graphite can be recognized. These include in-pore diffusion, grain boundary diffusion, surface diffusion, and bulk diffusion.

In early studies, evaluations of transport had been made assuming a strictly bulk model, and this model may be used to describe steady-state transport of stable or long-lived metal isotopes,

but it will not handle both transient and steady-state diffusion in all graphites nor will it handle the migration of short-lived metal isotopes. A more sophisticated transport description that includes any effects of helium flow through the graphite is required.

Analyses of concentration profiles for fission-product metals in graphite from fuel elements irradiated in loops, Peach Bottom, and Dragon have yielded effective diffusion coefficient values that are generally lower than laboratory-determined values. The reason for this is not clear. Irradiation and "back-sweeping" by helium entering the purge channel through the fuel-element sieves may have had an effect in some instances.

Little information exists on the effect of interaction between different metals diffusing simultaneously in graphite. More work needs to be done in this area if Biso coatings are to be used extensively.

Vaporization studies

Adsorption isotherms (vapor pressure vs concentration curves) exist for cesium, strontium, and barium on some porous graphites. Data for silver are very sparse, and all adsorption phenomena are complex. The isotherms generally display Freundlich behavior at high concentrations and obey Henry's Law at lower concentrations. Only preliminary measurements have been made involving co-sorbed species, and a mathematical model to handle that situation requires confirmation and probably refinement. Changes in sorptivity due to irradiation of graphite have been observed. Changes due to thermal history, irradiation, and differences between graphite samples need to be better defined. They are generally greater for the less strongly adsorbed metals, i.e., greater for $Ag > Cs > Sr$.

6

Codes for description of fission-product release from HTGR elements

The FIPER code developed at GAC is used for estimating the release of metallic fission products from HTGR fuel elements. The code takes into account the combined effects of the retention of metallic fission products by fuel particles, the decay of these fission products during transport, and their rather strong sorption by the fuel-rod matrix material and graphite of the fuel elements. The loss of fission-product nuclides such as strontium, cesium, and barium from the exterior surface of the element is determined by their evaporation from the graphite surfaces through a "boundary

layer." Changes of fission rate (fuel-element power), fuel temperature, and graphite temperature during an element life may be incorporated into the calculation. Temperature and fission-product concentration are most important in determining release because, in general, release from fuel particles, diffusion in graphite, and loss by evaporation vary exponentially with the reciprocal of the absolute temperature and in a more complex manner with metal concentration. The UKAEA, the Dragon Project, and KFA Jülich utilize similar codes called FIPDIG and SLIPPER.

Mathematical representation of fission-product transport in graphite is commonly done in terms of bulk diffusion, assuming the applicability of Fick's law with a single diffusion coefficient in each medium, an assumption that disregards much of the nature and complexity of the process but which is justified by the relative simplicity that results and by the agreement obtained between many observed and calculated macroscopic distributions. Methods for taking coupled convective and diffusive flow in the fuel sleeve into account are currently being developed for use when necessary.

While FIPER and similar codes can be used to describe the attenuation of gases and iodine by structural graphites, the use of such complex mathematical models is not necessary to describe the steady-state diffusion of unadsorbed species; analytical solutions of the diffusion equation are simple and adequate under these conditions.

Sources of input data include phenomenological release constants derived from laboratory measurements of fission-product release vs time and diffusion coefficients derived from postirradiation sectioning of graphite from various experimental fuel elements (including some from Peach Bottom) and laboratory measurements.

Validation of predictions of fission-product release - $\tau(4,5)$

A satisfactory comparison between predicted and observed releases of metallic fission products from an HTGR element has yet to be published and sustained. Measurements intended to make such a comparison possible are being made at Peach Bottom, and some appropriate data are available from loop experiments. However these comparisons do not cover full irradiation times or current materials. This lack of confirmation, coupled with uncertainties relating to fuel and graphite temperatures, is causing designers to have to use very conservative (high) estimates for $\tau(4,5)$. Possible consequences of this are minimization of the fission-product inventory in the structural graphite, exaggeration of the hazards associated with a breach of primary containment or access to circuit components, and/or imposition of excessively stringent fuel-quality requirements.

with consequent economic loss. For cesium and silver, the uncertainty in $\tau(4,5)$ is greater than that in any other transfer function.

Scale-model experiments in which simultaneous transport of fission products and boron in an overheated core were measured have been performed up to 2500°C. Parts of a large HTGR core could reach higher temperatures during a postulated loss of cooling accident. Further modeling or experimental studies will therefore be needed to analyze the consequences of prolonged loss of cooling in a large HTGR.

Remaining uncertainties relating to krypton, xenon, iodine, and tellurium transport through fuel-element graphite arise from uncertainties in the helium flow induced between fuel and coolant holes by the pressure drop across the core. Experiments in the Fort St. Vrain reactor should provide pertinent information.

9.10.3 Identified work

Fission-product gas transport in graphite (task 2.5.3.1)

No further laboratory work is proposed on the transport of krypton and xenon through moderator graphite. Characterization of the moderator graphite adequate for calculation of transport characteristics and parameters using existing theory is assumed to have been carried out under task group 3.1 of this program.

Likewise, no new modeling of noble gas transport in graphite is proposed.

Sampling of the primary coolant in the Fort St. Vrain reactor for short-lived noble gases is proposed. The specific objective is to determine the effect of induced helium flows between coolant and fuel channels on the fluxes of ^{90}Kr , ^{137}Xe , and ^{140}Xe leaving the core. These nuclides decay to nuclides whose presence in the circuit constitutes a hazard.

Iodine, tellurium, and selenium transport and adsorption (tasks 2.5.1.1, 2.5.2.1, and 2.5.3.2)

Measurements of the capacity of reflector graphite for iodine are recommended, so that if the capacity of metallic surfaces in the coolant circuit is inadequate to retain a 30-year iodine inventory, a second sink can be evaluated. The range of temperature to be covered is 300 to 700°C, or whatever higher temperature is required to desorb iodine. The partial pressure range is from as-low-as-possible to 10^{-9} atm.

Tellurium behavior should be reviewed and the maximum tolerable value of $\tau(\text{Te}, 4,5)$ relative to $\tau(\text{Te}, 3,4)$ defined before further work is planned.

Studies of iodine and tellurium release from Peach Bottom and Fort St. Vrain cores and irradiation capsules should define better whether $\tau(\text{Te}, 4,5)$ is significantly less than $\tau(\text{I}, 4,5)$ and/or $\tau(\text{Xe}, 4,5)$. Information from Dragon suggests that $\tau(\text{Te}, 4,5)$ may be $\geq 0.1\tau(\text{Xe}, 4,5)$ under normal conditions.

Metallic fission-product diffusion and adsorption in graphite
(tasks 2.5.1.2, 2.5.2.2, and 2.5.3.3 through 2.5.3.5)

Adsorption and diffusion in graphite (task 2.5.1.2). The adsorption and diffusion of the metals cesium and strontium, and silver and cerium, should be studied as functions of concentration, temperature, and graphite type. In-pile experiments should be performed to study diffusion under reactor conditions. Comparison of in-pile and out-of-pile data should yield information on the combined effects of an irradiation field and helium pressure on diffusion rates. The effect of co-sorbed metals on the diffusion rate of a given metal should be investigated if fuel-quality specifications make it necessary. Relative contributions of gas phase, surface, and bulk diffusion processes should be elucidated as they affect metal transport. Emphasis should be on reference structural graphites, but others should be included to prove mechanisms. Effects of forced (transverse) helium flow on transport of fission products should be investigated. Work should be performed to determine the vapor pressures of the fission products sorbed on graphite. Strontium and cesium are of major concern, but other metallic fission products should be studied also. One aim is to obtain information on the sorption processes of co-sorbed species. Studies on various forms of carbons and graphite, including representative fuel-rod matrix materials and pyrocarbons, should be completed. Irradiation effects on sorption properties should be investigated further. Vapor-pressure measurements at low concentration should be emphasized to provide needed data and to characterize the sorption behavior at low concentration. Experiments should be performed to determine distribution coefficients for fission-product metals between graphite and matrix material.

The most important parameters to be derived from this work are the times required, first, for a fission product to break through the graphite and, second, for steady state to be obtained. Designers would prefer these times to be long compared to the fuel-cycle irradiation time. When they are not, the fuel must be relied on to limit fission product release [i.e., $\tau(2,3) \approx \tau(4,5)$] and hence the primary circuit inventory.

Fission-product transport code development (task 2.5.2.2). FIPER Q should be refined as follows: (1) Changes should be made to permit faster operation and easier modification. (2) Diffusion coefficients should be allowed to vary with concentration. (3) Allowance should be made for the effects of forced helium convection on metal transport.

Concurrently, methods for performing whole reactor calculations that give proper weight to parameter uncertainties should be defined. As a check, a number of test calculations of varying complexity should be performed with FIPER and FIPDIG or other nominally equivalent codes obtained under foreign exchange agreements.

Validation of models for metallic fission-product behavior in graphite (tasks 2.5.3.3 through 2.5.3.5). A valid model of fission-product transport in graphite should permit calculation of the following quantities to the desired degree of accuracy (to within a factor of 3):

- (1) the ratio $\tau(I, 4,5)/\tau(I, 3,4)$ as a function of time for each important nuclide. τ would be integrated over a single element or a whole core, depending upon the scale of comparison possible.
- (2) the integral of $\tau(I, 4,5)/\tau(I, 3,4)$ over the operating time for a given element or core.
- (3) the amount and distribution of each important nuclide within the core graphite at various times.

The experimental measurements required to obtain these quantities are complex and expensive because they involve thorough postirradiation examinations of many fuel elements as well as systematic measurements of the very low concentrations of nuclides leaving and entering the core. However, the same examination yields the integral fraction of each nuclide released from the fuel and the amount deposited in the primary circuit. It is therefore convenient to combine a number of validation tasks into a single fission-product surveillance program covering tasks 2.4.3, 2.5.3, and 2.6.3 for a given reactor or loop. For the purpose of this program, costs and work have been apportioned between fuel, graphite, and coolant tasks on the basis that graphite analysis should bear all fuel-element handling and shipping charges because it is the most complex.

Techniques for making the necessary measurements exist in principle. There are, however, incentives for improvement. Specific difficulties associated with the examination of block elements exist, which would require about one year and \$100,000 worth of equipment

for their solution. A dedicated hot cell is not considered necessary at this time, but it would be needed if contamination from other jobs could not be controlled. A dry atmosphere would be required if fuel-rod swelling caused by hydrolysis of exposed fuel in air could lead to additional damage to the graphite. Samples of graphite would be made available to task group 6.2 for determinations of mechanical properties and structural integrity.

Places for making the measurements of fission-products leaving and returning to the core need to be provided in first-of-a-kind large HTGRs and in any fuel test loops that may be built. A procedure for sampling gas leaving a given element in an HTGR would improve test precision substantially by eliminating uncertainties relating to element-to-element temperature and material variations.

For gaseous fission products (task 2.5.3.1), steady state is reached quickly, and concentration measurements in the coolant are easy.

For iodine and tellurium, steady state is also expected to be reached rapidly. Gas sampling and fuel analysis therefore yields the average delay, and graphite analysis is simplified.

For metals, the most important practical questions are how long does a nuclide take to break through the graphite web and how much longer is required to achieve steady state? All measurements may therefore be needed. Inventory determinations require extensive and careful sampling if steep concentration gradients are to be characterized precisely. Valuable information about transport mechanisms and the availability of fission products for release to the coolant during accidents involving abnormal temperatures can also be derived from concentration gradients.

The number of elements to be selected for examination must be sufficient to instill adequate confidence in codes, models, and input data and to take into account the degree of material and performance similarity expected from elements of the same specification. Some reflector blocks should be examined to establish what fraction of fission products deposit there.

The suggested minimum schedule for removal of fuel elements and associated reflector blocks from the Peach Bottom and Fort St. Vrain reactors is as follows:

FY 1974 1 Peach Bottom element

FY 1975 3 Peach Bottom elements, 3 side reflectors

FY 1976 1 Fort St. Vrain element (block)

FY 1977	1 Fort St. Vrain element
	4 Fort St. Vrain elements, preferably from one column, and 1 top and 1 bottom reflector block
FY 1979	4 Fort St. Vrain elements
FY 1980	4 Fort St. Vrain elements
FY 1981	6 Fort St. Vrain elements from one column, and top and bottom reflector blocks

This schedule should be revised in the light of performance and any changes in goals for fission product control. A requirement for a cleaner primary circuit should lead to an increase in core surveillance.

Postirradiation examination will sometimes occur in the fiscal year following the one in the schedule. Provision has been made in task 2.4.3.1 for metallographic examination of fuel adjacent to areas of graphite sampled.

Given satisfactory fuel performance and no large changes in fuel specification, the program would fall to a low level in FY 1982 while findings were reported and decisions formulated about surveillance needs for the 1100-MWe HTGR.

Actinide and control material diffusion in graphite tests (tasks 2.5.1.3, 2.5.1.4, and 2.5.2.3). The availability of data needed to determine the consequences of very high core temperatures on the distributions of boron, uranium, thorium, protactinium, and important heat-producing fission products should be determined and the need for further data evaluated.

9.11 FISSION PRODUCTS IN THE COOLANT CIRCUIT — SURFACE CHEMISTRY (TASK GROUP 2.6)

9.11.1 Justification and scope

Reference to Fig. 9.7 will show that the coolant circuit occupies a key position in the path of fission products from the fuel to the environment. It is the repository of almost all fission products leaving the core and the most available source of fission products, should the primary helium containment — the PCRV liner — be breached.

A fission-product atom entering the coolant circuit from the core may either decay, enter the purification plant [$\tau(5,7)$], be deposited upon a relatively cool metal surface [$\tau(5,6)$], leak into the reheater section of the steam generator [$\tau(5,8)$], or leak to the secondary containment [$\tau(5,9)$ and $\tau(5,10)$]. The purpose of this task is to provide mathematical models and parameters to determine the relative and absolute values of these various transfer functions during normal operation and accidents.

During normal operation, the ratios $\tau(I, 5,9)/\tau(I, 5,6)$ and $\tau(I, 5,10)/\tau(I, 5,6)$ — where I represents iodine, not an index number — are of greatest significance in determining permissible PCRV leak rates and permissible fuel quality.

During an accident, the reverse processes represented by $\tau(6,5)$ and $\tau(7,5)$ and the possibility of large changes in $\tau(5,9)$ and $\tau(5,10)$ become of predominant safety significance.

The total coolant-circuit inventory of gamma-emitters such as cesium and iodine may be as important as their distributions. Inventories are determined by fuel quality and $\tau(4,5)$ rather than by a property of the coolant circuit.

The fission products of significance in the coolant circuit are the same as those in the core, but the order of importance is now $I > Cs > Sr$. The importance of tellurium in HTGR materials technology remains to be assessed.

During normal operation, these fission products are exposed to helium containing specified impurities at concentrations much greater than those of the fission products but still only about 10^{-5} to 10^{-6} times the helium concentration. Some large fraction of these fission products will have been absorbed, condensed, or reacted with graphite or steel surfaces.

The perturbations that can most affect the distribution of these fission products are depressurization, water ingress, and a temperature rise. Depressurization can in principle cause evaporation into the leaking helium and mechanical displacement of loosely adhering material. Water ingress can also cause mechanical and chemical displacement from metal and core surfaces. A surface temperature rise will change the proportion of a fission product (like iodine) in the vapor and adsorbed states.

This task group is concerned with the characterization of adsorption and desorption processes. Experiments in reactors must play a large part in this task because of their realism and the difficulty of specifying and reproducing all relevant conditions in the laboratory.

9.11.2 Status

The activity level of noble gases (krypton and xenon) in the primary coolant circuit is generally established by the fuel particle release rate, the purification system removal rate, and the decay rate. Some of the noble gas nuclides decay to metallic nuclides, which deposit on steel surfaces or dust particles. The activities arising by this route are easily and accurately assessed.

Iodine and tellurium adsorb on graphite and metallic surfaces to an important extent, which is being defined experimentally. Large values of the ratio (iodine on the circuit wall)/(iodine in the coolant) have been observed in operating HTGRs to date, but always in association with small iodine inventories.

Cesium, strontium, and other metals adsorb even more efficiently on a variety of metallic surfaces, so much so that every atom of these metals that hits the tubes of the steam generator can probably be assumed to stick.

What is not resolved are the absolute capacities of the various surfaces for fission products relative to a 30-year inventory, the role of aerosols (dust) in fission product retention by surfaces, and the possible effects of large changes in the concentration of water in the circuit.

The first step in resolution of these uncertainties must be characterization of the species involved. Relevant studies of adsorption and desorption equilibria and dynamics may then be carried out with confidence in their ultimate practical utility.

Some progress in the characterization of molecular species, aerosols, and surfaces has been made with the help of diffusion probe experiments conducted in the General Atomic in-pile loop (GAIL), the PLUTO loop (UKAEA), and the Peach Bottom and Dragon HTGRs.¹⁶⁻¹⁹ For example, it was shown that iodine nuclides are transported in the helium coolant predominantly in the form of atoms rather than in the form of molecules or aerosol. For cesium, molecular weight depended much more strongly on the relative concentrations of dust and cesium. Generalizations about molecular form must therefore be made with caution.

For iodine, the possibility of forming weakly adsorbing organic iodine compounds must also be considered. At the core exit temperature, the CH_3I level in the main coolant of an HTGR is expected to be negligible because I_2 and CH_4 , which could lead to CH_3I formation, should be present only at very low concentrations (<10 ppm) and because CH_3I decomposes readily thermally and

radiolytically. Experiments in Dragon and Peach Bottom substantiate this view.^{18,19} Small amounts of organic iodine have been reported downstream of the steam generator.¹⁸

Adsorption of iodine and important iodine compounds on iron, Fe_3O_4 , and certain alloys is being studied under conditions relevant to normal operation and accident analysis. Kinetic investigations are included.²⁰ Information on iodine adsorption on graphite is inadequate.

Cesium and strontium adsorption on steel surfaces has also been studied.²¹ While it is evident that adsorption is strong, the thickness of the surface oxide film, and hence the coolant composition, is an important factor in determining adsorptive capacity. Some kinetic data are available, but more work is needed. The relative affinities and capacities of the various potential components of the surface of a steam generator tube (i.e., dust, oxide film, and base metal) for cesium and strontium need to be investigated further so that the role of dust in fission transport during a depressurization can be defined better. Experiments to date indicate that surface (film + base metal) has a greater affinity for cesium and strontium than carbon dust, but rates of uptake and capacities are uncertain. Characterization of the dust in Peach Bottom and Dragon is continuing.

The PAD code,⁶ which has been recently developed at GAC based on earlier work at ORNL and Battelle Memorial Institute, is used for describing the distribution of fission products in the HTGR primary coolant loop under normal conditions. It takes into account gaseous precursors and does not require the assumption of an infinite sink for fission-products. It does not consider diffusion into circuit walls and may therefore misrepresent behavior at higher temperatures. The code has been used to compare predicted and measured activity distributions in the Dragon coolant circuit and in laboratory experiments. More refined quantitative comparisons need to be made. A code that considers diffusion into the duct wall is being developed at KFA Jülich to assist in gas-turbine accessibility studies.

9.11.3 Identified work

Noble gases

Measurement of noble gas concentrations in coolant circuits should only be made as an adjunct to studies of metallic fission-product distributions and for the reasons given above.

Iodine and tellurium adsorption studies (task 2.6.1.1 and 2.6.2.1)

Values of $\tau(I, 5, 6)$ and $\tau(I, 6, 5)$ and the capacity of the coolant circuit for iodine are needed for safety analysis, specifically analysis of permissible PCRV leak rates and fuel quality, which are strongly related to the 40-year capacity of the coolant circuit for iodine. Analyses of depressurization and water-ingress accidents require measurements of desorption kinetics.

Measurements of adsorption isotherms in the systems $I-Fe_3O_4$, $I-Cr_2O_3$, and $I-Fe$ should be completed and compared with results obtained on samples of various alloys and HTGR aerosol components. The measurements should be carried to the lowest accessible iodine pressures and cover the temperature range 200 to 550°C. Measurements of the capacity of graphite for iodine should be done under task 2.5.1.1. Studies of desorption kinetics in helium and helium-steam mixtures should be completed and the results incorporated in the PAD code.

Information on the adsorption of tellurium on metals that occur in coolant circuits and on graphite should be reviewed to determine the need for further work.

Concurrently thermodynamic analysis of the possible reactions between metal iodides and tellurides and coolant impurities and water should be undertaken in support of evaluations of depressurization and steam-ingress accidents.

No laboratory work on the rate of formation of organic iodides is proposed because more realistic experiments could be performed readily in the Fort. St. Vrain reactor.

Iodine surveillance studies (tasks 2.6.2.1 and 2.6.3)

Iodine and tellurium concentrations on surfaces and in the gas phase should be measured in operating and shutdown HTGRs at all possible opportunities. Measurements need to be continued over the life of the Fort St. Vrain reactor to gain assurance that changes in the processes controlling the iodine partial pressure in the coolant can be noticed. An example of such a change is a change from control of $P_{(I+I_2)}$ by deposition to control by FeI_2 condensation or by the helium purification rate.

The need for reactor sampling depends upon the primary circuit inventory, i.e., upon the total amount of iodine to be released from the fuel. It, therefore, becomes less necessary as the fuel quality improves. The priority assigned to iodine surveillance is

based upon a "design" fuel-failure fraction of 10%. Given this fuel quality performance, coolant samplers such as those used to monitor iodine in Peach Bottom, Dragon, and Fort St. Vrain may be needed in the first large HTGR.

Results should be evaluated using the PAD and other similar codes. Code improvements should be made as needed.

Metallic fission products in the coolant circuit

Concentrations of gas-borne metallic fission products are determined mainly by deposition, i.e., by $\tau(5,6)$ and the capacity of the coolant circuit surfaces, and by the corresponding functions for circulating dust.

The hazard associated with fission products in the coolant can be defined, given the normal operating inventory, if $\tau(6,5)$ is known for dust and fission products in helium and in steam. Definition of the gamma-radiation dose to maintenance crews requires that a nonuniformity of surface concentration be recognized. It is more important initially to establish adequate capacity than to refine knowledge of distribution.

Metallic fission-product adsorption and diffusion studies (tasks 2.6.1.2, 2.6.1.3, 2.6.2.1, and 2.6.3.2)

Reasons will be sought for the very high capacity of metallic surfaces for cesium and strontium, with the aim of determining the stability of the chemical systems involved and investigating the reversibility of their adsorption processes. This goal will necessitate adsorption, desorption, and other surface-oriented chemical investigations. New techniques will be needed since strontium is an extremely good oxygen getter. The relative affinities of dust, metal, and metal oxides for fission metals must be re-evaluated.

Measurements of the diffusion of silver, cesium, and strontium into the metals in the primary circuit may also need to be made. Here again substantial benefits from the ORNL-KFA exchange can be expected. Specimens cut from the Peach Bottom HTGR during decommissioning should be used in these studies (task 2.6.3.2). Results of these investigations should be used to verify and/or modify the PAD code.

Absorption of fission products on aerosols (tasks 2.6.1.4 and 2.6.3.2)

The first requirements of these tasks are to determine that aerosols transport fission products in significant quantities and that the aerosol retains fission products when it deposits on a wall. Without this assurance, studies of aerosol dynamics will reveal nothing about fission product mobility and vice versa. Pertinent information will come from studies of aerosols and fission-product distribution in Peach Bottom and Fort. St. Vrain and analysis of the partition of fission products between dust, oxide films, and base metal on the Peach Bottom steam generator tubes (tasks 2.6.3.2 and 2.6.1.2).

Laboratory work would then be initiated (if necessary) on the adhesion of particulate matter to metallic surfaces as a function of thermal history and system composition (see task group 2.2).

9.12 TRITIUM BEHAVIOR IN HTGR (TASK GROUP 2.7)

9.12.1 Justification and scope

The scope of this task is definition of the paths by which tritium can reach the environment.

Tritium has been assigned a task group of its own partly by historical accident and partly because it has unique sources and distribution mechanisms.

Permissible discharge levels for tritium during an accident are large compared with the amount of tritium easily available in an HTGR. Normal discharges, however, require more careful evaluation.

9.12.2 Status

The principal sources of tritium in HTGR systems are ternary fission and neutron activation of (1) ^3He present in the helium coolant, (2) ^6Li and ^7Li present as impurities in the core and reflector materials, and (3) ^{10}B in the control rods and reflector materials. Other tritium-producing reactions are relatively unimportant. The main concern with tritium is its rate of release to the environment by diffusion through the steam generator tube walls into the secondary coolant (see Fig. 9.1).²²

Some work has been done by GAC and German investigators on the retention of tritium by fuel kernels and coated particles. The results of this work generally indicate that tritium is retained in fuel materials at normal temperatures and that the retention decreases rapidly with increasing temperature. Silicon carbide coatings appear to be more retentive of tritium than are pyro-carbon coatings.

A few measurements, both at GAC and ORNL, have been made on the retention of tritium in irradiated graphite. Recent results suggest that tritium produced from ^3He can be trapped by core materials (either by exchange or sorption) and held there, but the capacity of the graphite is unknown.

A small amount of work has been done on the retention of tritium born in control materials. That work shows appreciable tritium retention by control materials. Confirmation is required.

Several tritium surveys have been performed at Peach Bottom. The work, all of which was performed during Core II operation, has yielded valuable data which are being used as a basis for estimating tritium behavior and release in Fort St. Vrain and large plants. These data indicate that tritium born in Peach Bottom core materials is largely held there, but further substantiation and explanation of observations is needed.

An improved device for continuous monitoring of the tritium content of the helium coolant has been developed at GAC.

A code describing tritium distribution in an HTGR and release to atmospheric and liquid waste streams under steady state conditions has been prepared by ORNL.²²

9.12.3 Identified work

The relative importance of the different tasks relating to tritium distribution are discussed under Sect. 9.4.2.

A large part of this task group involves the determination of ^3H in reactor components or coolants both during and after irradiation.

Release and transport parameters (tasks 2.7.1.1, 2.7.1.2, and 2.7.1.3)

The important existing work on release of tritium from fuel - $\tau(T, 1,2)$, $\tau(T, 1,3)$ and $\tau(T, 2,3)$ - and other sources should be completed (task 2.7.1.3). Work on tritium permeation through oxidized steam generator tube materials [$\tau(T, 5,8)$] should be brought to a point where realistic permeation constants can be recommended. Reactor experiments carried out under tasks 2.6.3.1 and 2.6.3.3 will be important in establishing the credibility of recommended values because of the difficulty of defining the states of surface films.

The capacity of graphite for tritium (task 2.7.1.1) should initially be determined by injection of deuterium into Peach Bottom (task 2.7.3.1) and post-irradiation examinations of fuel elements (task 2.7.3.2). The need for laboratory studies should then be evaluated.

Tritium release code - TRITGO (task 2.7.2.1)

The ORNL code TRITGO should be refined. The improvements needed relate principally to the representation of the variation of tritium release from fuel and recoil sites in other source materials with temperature. Further changes can be expected as a result of experiments in reactors.

Tritium distribution model validation (tasks 2.7.3.1 through 2.7.3.6)

Work on these tasks consists primarily of tests to be performed in the Peach Bottom and Fort St. Vrain reactors and is aimed at deriving information on tritium generation, transport, and release in operating HTGR systems. The expected outcome of this work is validation of source data and calculational methods used in estimating tritium transport and release in large HTGR systems.

Peach Bottom is very similar to the large HTGRs with regard to important parameters that control the generation and transport of tritium and, therefore, constitutes a representative operating model from which to derive reliable estimates of tritium transport and release from large HTGR plants. Accordingly, tritium survey work should continue at Peach Bottom, and, in addition, tests should be performed in the reactor before the final shutdown. These tests include:

- (1) variation of the partial pressure of hydrogen (P_{H_2}) in the helium coolant to investigate exchange between adsorbed tritium and hydrogen and mass action effects of hydrogen on the transport of tritium into the secondary coolant. Deuterium may be substituted for hydrogen in this test as analysis shows more information can be obtained.
- (2) variation of hydrazine concentrations in the secondary coolant water to determine whether hydrazine is a source of hydrogen in the primary circuit.

Test to be performed after the final shutdown at Peach Bottom include:

- (1) determination of tritium content in selected helium and liquid systems, in particular those systems that are not readily accessible during reactor operation.
- (2) determination of tritium retention in irradiated core materials, including fuel, graphite, and control-rod materials.

Additional detail on tritium-related work in Peach Bottom is presented in Appendix B.

Tritium surveys also will be performed in the Fort St. Vrain reactor with emphasis on obtaining a mass balance for tritium. The tritium content of samples of irradiated Fort St. Vrain core materials should be determined as samples become available. The primary aim of the tritium work in Fort St. Vrain and the large HTGR is to establish the validity of extrapolations from Peach Bottom experience.

9.13 FUEL PARTICLE FAILURE MECHANISMS (TASK GROUP 2.8)

9.13.1 Justification and scope

This task group is concerned with defining conditions for the safe operation of HTGR fuels and mechanisms of failure.

Fuel-coating failure can change fission-product release rates by several orders of magnitude, depending upon the temperature at which failure occurs, the type of coating being used, and the fission product of interest. Furthermore, coating degradation experienced during a thermal transient can influence fission-product release rates in subsequent normal operation.

Two sorts of "failure" are of practical interest: irreversible fracture of one or more of the load-bearing coatings and reversible changes in fission-product retaining properties associated with large temperature rises, that reduce the value of the coating in fission-product control. This task group is concerned only with irreversible phenomena occurring under abnormal conditions. The reversible variation of $\tau(2,3)$ with temperature is dealt with in task group 2.4. The HTGR base program is currently responsible for providing parameters and models for the mechanical design of fuel particles, samples of irradiated fuels, and validated failure criteria for service under normal conditions.

All candidate HTGR fuels are assumed to lie within the scope of the program, though experimental emphasis will be upon the more stable U.S. reference materials.

The thermal conditions to be expected during accidents will be defined by analyses carried out under task group 1.3. In general, experimental and theoretical studies will be carried to temperatures at which failure is rapid in an appropriate context. The range of specific power to be considered is dictated by the need to use accelerated irradiations in fuel testing.

9.13.2 Status

Coated fuel particles can fail during accidents as a result of high internal pressure, changes in coating properties occurring during the accident, changes that occurred during previous irradiation, and chemical reaction.

The internal loads on a coating following a given irradiation are described by codes such as STRESS II, which takes into account rare-gas release, swelling pressure, thermal stresses, and stresses caused by radiation damage.

The current status of mechanical design codes has been reviewed by Homan²³ of ORNL and by Dragon staff. In general, modeling capability is far in advance of parameter availability, particularly if parameters are required to be individually determined rather than deduced from the behavior of integral experiments. Analysis of failure probability vs temperature is hampered by lack of data at appropriate temperatures for the strengths of currently used coating materials, uncertainty in the radiation-induced creep constants that strongly determine the stress state of a coating at the onset of a thermal transient, and uncertainties relating to the variations in those parameters that may be expected to result from intentional or random changes in coating conditions within a given fuel specification. Base programs are expected to supply this information.

Kernel swelling is usually assumed to be relatively unimportant as a source of internal pressure in a properly designed particle. 100% release of stable krypton and xenon isotopes is currently assumed for design purposes. This assumption may be pessimistic for short experiments or for fuels with low FIMA (<5%).²⁴ Confirmation should be sought for thorium cycle fuels.

A formula for calculating the pressure of carbon monoxide in oxide particles as a function of temperature has been derived by Lindemer.²⁵ Estimates of P_{CO} as a function of burnup need to be more precise, particularly for very-high-burnup fuels.

External loads arising from matrix-particle and matrix-block interaction have not been considered in assessing coating-failure probability vs temperature, because most experimental work is being done with loose or unrepresentatively bonded fuels.

Experiment indicates that Triso-coated carbide and oxide fuels can be expected to survive to >1700°C for a few hours without irreversible damage. Substantial further experimentation and analysis is needed.

Internal chemical reactions degrade coatings in two ways: (1) some kernels and fission products can react with or dissolve in coatings and (2) in a temperature gradient, carbon and kernel materials can be transported across particles, thus weakening one side of the coating. Both these sets of processes are more sensitive to temperature than to the mechanical properties of graphite. They will therefore become relatively more important at higher temperatures.

Carbon and fuel-kernel transport rates and mechanisms have been determined for ThC_2 , $(U,Th)C_2$, and UC_2 in coated particles placed in temperature gradients in out-of-reactor experiments. Initial in-reactor temperature-gradient experiments with the carbides are currently being evaluated to quantify the in-reactor transport rates.²⁶ A kinetically limited process can be expected to be sensitive to fission in the medium and changes in the structure of the medium produced by fission.

Carbon and fuel-kernel transport mechanisms in ThO_2 , $(Th,U)O_2$, and UO_2 fuels are less well understood. Carbon transport involving the diffusion and interconversion of CO and CO_2 may be involved in the thinning and eventual failure of the pyrocarbon coatings. At the same time, the migration of the oxide kernel by solid-state diffusion of oxygen or carbon through the kernel could produce a similar net transport. A theoretical analysis of these mechanisms

has been made and has been applied to the analysis of observed in-pile migration of Biso ThO_2 , $(\text{Th,U})\text{O}_2$, and UO_2 particles and Triso UO_2 particles.²⁷ Preliminary values of the in-pile kernel migration coefficients have been calculated from in-pile experimental data from 900 to 1900°C

In the absence of experiments in which substantial amounts of reference fissile fuel have been irradiated for the full time to full burnup (75% FIMA), a core-operating limit calculation based upon currently available phenomenological kernel-migration coefficients cannot be guaranteed.

Experiments currently planned and in progress extend time and burn up beyond current limits, but they still do not cover the full 1200-day irradiation period of a thorium-cycle HTGR. The only way to avoid such long experiments without excessive risk would be to obtain a better appreciation of rate-controlling mechanisms.

Stabilization of kernels by dilution with carbon and additions of UC_2 , CeO_2 , and SiC to control oxygen potentials is being tried, based upon the working hypothesis that all adverse processes will proceed more slowly at lower G/U ($\approx \text{P}_{\text{CO}}$) ratios. Some favorable results have been reported with SiC .

Low enrichment cycle fuel has been irradiated to full time, burnup, and fluence simultaneously.²⁸

After very high burnup, the particle contains more fission products than fuel, and the compatibility problem also becomes more complex. Two practically important questions relate to (1) the carbon monoxide pressure to be expected in a very highly irradiated oxide particle and (2) the compatibility of fission products with the SiC coating. Another important question relates to the compatibility of fission products and coating materials in high-burnup carbide particles.

Plutonium-containing systems have not been examined extensively in the U.S. for HTGRs. Work on $(\text{Pu,U})\text{O}_2$ fuels has been reported by the Dragon Project, and a plutonium test element is currently undergoing irradiation in the Peach Bottom HTGR.

2.13.3 Identified work

Fuel-coating failure mechanisms - Mass transport in thermal gradients and compatibility problems (tasks 2.8.1.1 and 2.8.1.3)

Mechanisms for the observed transport of carbon and silicon across coated oxide particles and the subsequent coating failure should be analyzed further. The analyses and models should be continuously updated in the light of experimental results from all available sources. The relative thermodynamic stability of SiC, fuel, and fission products will also be determined in an effort to understand the attack of SiC often observed in irradiation experiments.

In-pile and out-of-pile experiments for measurement of the rate of failure of oxide-coated particles as functions of temperature, temperature gradient, and internal oxygen potential (P_{CO}) should be set up. These experiments should be used to test the hypothesis that lowering and controlling P_{CO} by adding amounts of actinide carbides or other additives to the actinide oxides will reduce the mass transport rates.

The compositions and pressures of gases in both as-coated and irradiated particles should be measured. A knowledge of the variation of partial pressure of fission gas as a function of burnup and fuel isotope is important since this information is one of the primary parameters needed for both the mechanical and chemical failure models.

High temperature isothermal failure mechanisms (task 2.8.1.2)

If decay heat cannot be removed sufficiently rapidly, a particle will experience a high but uniform temperature at zero neutron flux.

The consequence of this should continue to be investigated, primarily experimentally, by annealing previously irradiated particles or fuel rods at various constant temperatures. When some understanding of failure under these conditions has been obtained, experiments simulating expected transients should be performed. Fission-product releases should be measured simultaneously in all experiments.

Formulation of fuel failure criteria (task 2.8.2.1)

Improved mathematical models of the carbon transport and/or kernel migration in oxide and perhaps carbide fuel systems should be established. Additions to fuel mechanical design codes to take account of carbon monoxide production should be made. As mechanical property data as a function of temperature become available, the STRESS II code should be used to analyze mechanical failure and hence devise abnormal operational criteria.

Validation of fuel failure criteria (tasks 2.8.3.1 through 2.8.3.4)

Fuel failure criteria should take the form of a set of mathematical relationships between fuel-coating survival time, temperature, specific power, and previous irradiation history.

At the long time end, validation should consist of observing the behavior of fuel removed from Fort. St. Vrain (task 2.8.3.3) or irradiation capsules (task 2.8.3.1). However, confirmation of the short-term behavior should require special experiments in which temperature, specific power, and burnup effects can be checked systematically. Furthermore, until the role of reactor radiation ($n + \gamma$) in carbon transfer reactions is better understood, such experiments would need to be performed in a reactor, perhaps using fuel previously irradiated in an HTGR to produce a suitable fission-product inventory and coating stress state.

Since large HTGR fuels differ from Fort St. Vrain fuels, they will need to be examined separately (task 2.8.3.4).

By contrast, the validation of a failure criterion in which the specific power is almost zero and the neutron flux is zero is a relatively simple hot-cell annealing job, again using fuel from operational HTGRs.

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10. PRIMARY COOLANT TECHNOLOGY (TASK AREA 3)

This task area is concerned with describing the effects of chemical reactions that can occur between impurities in the primary coolant and the coolant-circuit materials, including the core graphite, fuel, and structural metals. For the core materials, particular emphasis is given to an elucidation of the consequences of steam ingress into the primary coolant system, because this condition results in greatest impurity concentrations and could cause damage to the core. Insofar as consequences of steam-fuel interactions are concerned, relatively short-term effects are involved, whereas both long- and short-term effects need be considered in evaluating steam-graphite interactions, depending upon whether or not the graphite forms part of the renewable core or serves as a structural member. For both the fuel and the graphite, direct interaction with an oxidizing impurity is of concern, and one of the effects of this interaction is the introduction of other impurity forms into the coolant circuit. Generally, it is these impurity forms which can interact with the metallic structural materials, and the effects of this type of interaction are of long-range consideration.

This task area is divided into three task groups. The first of these, task group 3.0, involves a continual assessment of the adequacy of primary coolant technology as regards reactor safety and safety-related issues. Task group 3.1 deals with steam-core reactions and coolant composition, and primary emphasis is given to describing effects on fuel and graphite which result from oxidant ingress. Task group 3.2, on the other hand, is less concerned with the introduction of oxidants directly, but rather with the compatibility of other coolant-circuit materials with other impurity forms, particularly those forms that result from the oxidation of core materials.

10.1 AREA OBJECTIVES

The objective of this task area is to ensure that a sufficiently detailed understanding of the primary system chemical technology exists, so that accurate quantitative evaluations of safety-related consequences of exposing HTGR primary systems to adverse chemical environments are possible. Both long-term exposure at low contaminant levels and relatively brief exposures at high impurity concentration are of concern. The technology is likewise expected to provide a basis for the evaluation of safety margins that are associated with particular system features.

In more specific terms, methods of analysis and input data are required:

- (1) to predict moderator and matrix carbon removal rates, fuel damage and fission-product release, and resultant coolant composition as a consequence of steam ingress into the coolant circuit,
- (2) to provide criteria to assess the adequacy of response of instrumentation to detect steam ingress and of safeguards and cleanup systems to contain the effects of steam inleakage,
- (3) to evaluate incompatibility effects between coolant impurities and circuit structural materials,
- (4) to predict the effects of introducing flammable carbon oxidation products into the secondary containment,
- (5) to evaluate the introduction of air into the reactor core or into spent fuel storage vaults of HTGR plants, particularly with regard to oxidation of the graphite and fuel components, and the release of radioactive species.

The overall objective is thus to guarantee that these methods of analysis and the input data are available.

10.2 SCOPE

The scope of task area 3 is generally limited to a consideration of the nature of chemical attack on fuel, graphite, and metallic reactor components and of the direct consequences resulting from the introduction of impurities into the primary coolant circuit of an HTGR. Moreover, although the causes for the initial introduction of contaminants are outside the scope of this task area, attention is also focused on the likelihood of developing other sources for contaminant ingress as a consequence of the initiating event.

Similarly, consideration is given to fission-product transport only to the extent that these species affect catalysis or inhibition of the chemical reactions of interest. However, considerations of fuel-kernel hydrolysis which are included here do provide source-term data for fission-product migration, which is a concern of task area 2. Additionally, the determination of coolant composition and the specification of relevant surface characteristics, which are objectives of this task area, likewise provide input for establishing fission-product behavior in the coolant circuit. This

too is a concern of task area 2. Moreover, the definition of contaminant effects on metals aids in the development of materials technology (task area 6). Lastly, all of the concerns of task area 3 are necessarily involved in the overall accident analyses which form the subject of task area 1.

10.3 STATUS AND NEEDS

There are five modes for the introduction of oxidizing species into the primary circuit. In four of these, the oxidant is steam or water converted into steam: (1) leakage from a main steam generator tube, (2) failure of a main helium circulator water bearing and seals, (3) rupture in an auxiliary heat exchanger (or leakage at low pressure conditions), and (4) leakage from a liner cooling pipe. In the fifth mode the oxidant is air; however, this mode requires depressurization of the primary circuit.

Another mode for the accidental oxidation of a graphite fuel element, but one that does not involve the HTGR core, is the unintentional exposure of a fuel element to air while the element is located within a spent fuel vault.

The oxidation of graphite by air is exothermic and is much more rapid than corrosion by steam under similar conditions. As a result, in those cases of interest, the rate of oxidation of the fuel elements (and hence the rate of heat release) will most likely be determined by the rate at which air is brought into contact with the elements rather than by the chemical kinetics.

Corrosion by steam, on the other hand, is endothermic; so the reaction cannot be self-sustaining. But the potential for steam ingress is much greater than for contact of the fuel elements by air. In addition, the effects of steam ingress into the coolant circuit are more numerous and require more complex analysis. This is due in part to the flammability of the major reaction products, hydrogen and carbon monoxide, and the potential for carbon transport by a cyclic mechanism involving the CO-CO₂ equilibrium. This last circumstance may produce undesirable effects on structural metals.

Two types of steam inleakage are possible. One of these — continuous inleakage at operating conditions — can only result from defects in the steam generators. The effects of such inleakage, if sufficiently small (and carefully monitored), are of concern because of economic factors, such as the frequency with which components of the core or cleanup system must be replaced or regenerated. However, inleakage of this nature could lead to carbon transport and

carburization of the steam generator tubes and ultimately result in failures that could affect reactor safety.

The second type of steam leakage is of a pulsed nature, in which a large quantity of steam is introduced into the core of the HTGR over a brief period of time (the sources of ingress of this type have already been identified). The effects of large steam leakages include:

- (1) saturation or blockage of the helium cleanup system,
- (2) removal of carbon from graphite structural members,
- (3) removal of carbon from graphite matrix,
- (4) generation of hydrogen and carbon monoxide, which may lead to fire hazards if these gases are allowed to mix with air,
- (5) alteration of properties of coolant-circuit surfaces,
- (6) hydrolysis of carbide fuel, with concomitant increases in fission product release,
- (7) redistribution of fission products by formation of more volatile species (e.g., oxyhydroxides and carbonyl compounds).

At least six computer codes have been developed to account for corrosion of graphite by steam, but three of these have been developed outside this country and information regarding these codes is limited. Of the three U.S. codes, two have already been employed for analyses of steam leakage into a large HTGR, whereas the third code, STEAMCAR, would require extensive modification if adopted for this purpose. None of the codes considers all of the known phenomena associated with graphite oxidation in a radiation field, but it is not yet clear that all of these phenomena need to be considered in the first place. Furthermore, although an adequate experimental program which is designed specifically to test the computer codes has yet to be developed, steam injection experiments have been performed in the Dragon Project reactor, and a loop experiment is under way in France with which code validation tests can be conducted.

Input data regarding the kinetics of the steam-graphite reaction, including effects of internal geometry, need to be subjected to sensitivity analyses using an adequate, if simplified, computer code, and additional studies need to be made to refine these data as required. Similar extensions of previous investigations are needed on hydrolysis of carbide fuel particles, though these studies are likewise subject to an ordering of importance as a result of

modeling studies. Consideration should also be given to the likelihood of formation of volatile forms of fission products, such as oxyhydroxides or carbonyl compounds.

Fission-product inventory in the coolant circuit will in some measure depend upon the chemical composition and morphology of surface films. Studies of this problem are insufficient. Similarly, data are insufficient for a determination of the effects of carburization of metallic surfaces by graphite dust or carbon deposition, particularly in view of the temperatures involved, and these effects need to be understood if safety margins are to be defined accurately.

10.4 TASK AREA EFFORT AND SCHEDULE

The relative effort associated with work identified in task area 3 is presented in Table 10.1. Only four points require elaboration. First, the significance of radiolytic reactions in establishing the steady-state composition of the coolant has not been resolved. Resolution of this question is anticipated from coolant composition determinations during Fort St. Vrain startup and the subsequent analysis of the data with the aid of the GOP computer code. If radiolysis is shown to be important, additional effort beginning in FY 1975 and concluding in FY 1979 is necessary to determine the yields (G values) of the pertinent radiolytic reactions. This effort is identified as task 3.1.1.1h.

Second, additional work under task 3.1.1.1i appears to be unwarranted at this time. A simplified model should be developed to describe specific sequences, and extant rate data should be used to demonstrate the assumed insensitivity to the chemical kinetics aspects. However, should the model indicate that the rate of contact of the graphite by air is not the rate-controlling mechanism, then the model should be employed to determine the sensitivity of the results to the rate parameters and the adequacy of existent data reviewed under this constraint. The modeling studies are properly included in synthesis task 3.1.2.1.

Third, significant effort appears under task 3.1.1.1f and task 3.1.3.1. These expenditures are required for an in-pile loop and a large-scale loop with which experiments can be performed to validate the steam-ingress computer codes. If input can be gained with regard to the conduct of the French Pegase loop, and if the data obtained from the operation of the loop can be made available to the AEC, then a considerable savings can possibly result.

Table 10.1. Summary of identified safety and safety-related work in task area 3

Task groups or tasks marked with an asterisk (*) are included in the safety planning guide (Part II of this document).

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 3.0 Need assessment*	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5	0.5
3.0.1 Source task										
3.0.1.1 Assessment of the adequacy of primary coolant technology as regards reactor safety.										
Task group 3.0 totals	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5	0.5
Task group 3.1 Steam-core reactions and coolant composition										
3.1.1 Source tasks										
3.1.1.1 Kinetics of graphite oxidation.										
3.1.1.1a Estimate of graphite variability. Determination of surface areas, permeability, interdiffusion coefficient values, and porosities of reference graphites.	1.0	1.0	1.0	1.0						
3.1.1.1b Effects of elevated pressure. Determination of effect of total pressure and of high steam partial pressure on rate of reaction.	1.0	1.0	0.5	0.5						
3.1.1.1c Identification of reaction regimes. Determination of pressure and temperature ranges over which chemical reaction control, in-pore diffusion control, and gas-phase diffusion control are operative.	(Included in tasks 3.1.1.1b and 3.1.1.1d)									
3.1.1.1d Determination of rate constants. Includes burnoff and catalysis effects of all pertinent reference graphites.	2.0	3.0	3.0	3.0	3.0					
3.1.1.1e Variation of geometric parameters with extent of reaction. Determination of some parameters as in task 3.1.1.1a but as a function of burnoff.	(Included in tasks 3.1.1.1d and 3.1.1.3)									
3.1.1.1f Transverse flow effects. Determination of steam-graphite corrosion characteristics under conditions that simulate a single coolant channel and surrounding fuel compartments.		2.0	6.0 ^{b,c}	3.0	0.5 ^d	0.5				

Table 10.1 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
3.1.1.1g Catalysis studies. Determination of catalytic effect of impurities and fission products on graphite corrosion by steam.	(included in task 3.1.1.1d)									
3.1.1.1h Measurement of yields of radiolytic reactions. Determination of appropriate G values as required.	(additional work conditionally unwarranted)									
3.1.1.1i Air-graphite reaction measurements. Determination of rate constants for graphites of interest.	(additional work conditionally unwarranted)									
Task 3.1.1.1 total	4.0	7.0	10.5	7.5	2.5	0.5				
3.1.1.2 Reactions involving fuel kernels. Determination of rate of carbide fuel hydrolysis and passivation mechanisms.	0.5	1.0	1.0	1.0						
3.1.1.3 Structural strength of graphite. Determination of graphite strength as a function of burnoff.		1.0	1.0	1.0						
3.1.1.4 Reactions involving fission products. Determination of likelihood of formation of volatile forms by interactions with steam or carbon monoxide		1.0	1.0							
3.1.1.5 Flammability studies. Determination of flammability limits of H ₂ -CO-He-H ₂ O-air mixtures.	(additional work conditionally unwarranted)									
Task 3.1.1 total	4.5	10.0	13.5	9.5	2.5	0.5				
3.1.2 Synthesis task										
3.1.2.1 Modeling studies of effects of steam ingress. Comparison of current codes, use of GOP in Fort St. Vrain startup, perform parameter sensitivity analyses.	2.0	2.5	2.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5
3.1.3 Validation task										
3.1.3.1 Validation of steam ingress computer code. Intercomparison of code results with large-scale and inpile loop experiments.			0.5	1.0	7.0	3.0	3.0	1.0		
Task group 3.1 total	6.5	12.5	16.0	11.5	10.5	4.5	3.5	1.5	0.5	0.5

Table 10.1 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 3.2 Primary coolant-component compatibility										
3.2.1 Source task										
3.2.1.1 Reactions of coolant impurities with HTGR structural materials. Determination of rates of oxidation, carburization, etc., of primary circuit materials in impure helium. Study of microstructural stability and surface films.	0.5	3.0	3.0	3.0	3.0	2.0				
3.2.3 Validation tasks										
3.2.3.1 Examination of components from Peach Bottom HTGR. Examination of components from primary circuit for evidence of oxidation, etc.	0.2	1.0	4.0	1.3						
3.2.3.2 Examination of specimens from Fort St. Vrain HTGR. Examination of metallic components (e.g., control rods).		0.1	0.1	0.2	0.2	0.2	0.1	0.1		
Task group 3.2 total	0.7	4.1	7.1	4.5	3.2	2.2	0.1	0.1		
Task area 3 total	8.2	17.6	24.1	17.0	14.2	7.7	4.1	2.1	1.0	1.0

^aIncludes construction of a large-scale loop that will also be employed in validation experiments.

^bNecessity of experimental loops and/or extent of experimental work depends upon the nature and the availability of Pegase loop data. Estimates assume that the Pegase data are unavailable or unacceptable for AEC use.

^cIncludes construction of an in-pile loop.

Fourth, validation tasks 3.2.3.1 and 3.2.3.2 list effort only for specimen examination and do not include costs for removal or transportation of the components.

None of the source tasks are strongly interdependent, and only two critical paths of any consequence exist. One of these involves source task 3.1.1.1, synthesis task 3.1.2.1, and validation task 3.1.3.1. Initially, the source task and the synthesis task can proceed independently. The continuance of task 3.1.1.1f beyond FY 1977, however, depends upon the extent to which the modeling studies are prepared for validation experiments (and subject to developments regarding the Pegas loop). Similarly, construction of an in-pile loop in FY 1979, under validation task 3.1.3.1, is dependent upon the successful performance of the task 3.1.1.1f large-scale loop and the ability of the computer model to describe the results.

The second critical path concerns task group 3.0 and synthesis task 3.1.2.1. Work associated with these two activities is interrelated; in particular, progress in task group 3.0 would be seriously hampered if an operational computer code were not available before FY 1976 for use in the technology assessment task.

10.5 ASSESSMENT OF PRIMARY COOLANT TECHNOLOGY (TASK GROUP 3.0)

The purposes of this task group are (1) to explore the consequences of steam ingress into HTGR primary coolant circuits in order to delineate safety and safety-related aspects of such occurrences, (2) to describe in detail the information required to establish the behavior of the system insofar as identified safety concerns are involved, and (3) to assess the adequacy of the technology to provide the needed information.

10.6 STEAM-CORE REACTIONS AND COOLANT COMPOSITION (TASK GROUP 3.1)

This task group contains several source tasks, a single synthesis task, and a rather broad validation task. The most prominent of these is task 3.1.2.1 - the development of models to describe the consequences of steam ingress into the coolant circuit of an HTGR. The source tasks are intended to provide data which will be used initially to facilitate the modeling process and later to permit validation of the models that are developed. Task 3.1.3.1 (validation of computer codes) is listed as the concluding phase of task group 3.1; its completion marks the attainment of all of the objectives of task group 3.1.

10.6.1 Modeling studies of effects of steam ingress (task 3.1.2.1)

Justification and scope

Prime objectives of task area 3 are to provide both the data and the methods to describe the effects of steam ingress into the coolant circuit of an HTGR in as much detail as is required and for all relevant conditions. This can only be achieved in a practical sense by extensive reliance on mathematical models of the system. Additionally, such studies aid in adapting certain constraints inherent in the system to an ordering of the significance of input data. This process thus provides justification for the elimination of unnecessary studies of contributory phenomena.

However, the task area 3 modeling studies do not include such factors as the cause of the steam ingress, the neutron flux distribution in the core, or the stress patterns on graphite structural members. All of these factors are to varying extents required for the specification of the consequences of steam ingress, and similar modeling studies are assumed to be conducted in these areas, so that the necessary input to the steam ingress computer code will be available when required.

Status

There are six operational computer programs that describe the corrosion of graphite by steam. A brief description of each is presented below.

OXIDE (General Atomic Company). The OXIDE code takes into account the rates of steam-graphite, oxygen-graphite, and steam-carbide reactions; diffusion of the steam into the graphite members; regional coolant flow and heat transfer within the bulk moderator and the fuel rods; and potential flammability hazards in the secondary containment due to the graphite corrosion products. The program utilizes a finite time-step approach to analyze the transient characteristics of core temperatures and graphite corrosion. The geometric model employs two-dimensional heat transfer and diffusion in the graphite and fuel instead of one-dimensional models as used in other codes. Fission product release due to fuel hydrolysis is also simulated.¹

STEAMCAR (Oak Ridge National Laboratory). The STEAMCAR program is a steady-state analysis of graphite corrosion by steam. The transport and rate mechanisms considered include the steam-graphite rate of reaction, diffusion of steam through the graphite, and turbulent convection in the coolant channel as the mechanism for contacting the graphite with steam.²

GOP (Battelle Northwest Laboratory). The GOP code considers the extent of chemical reaction as the coolant circulates through multiple core regions. Diffusion is described in steady-state form, and no provision is made for burnoff effects. The rates for the pertinent chemical reactions appear to be calculated in a simplified manner, with no regard for variable catalyst concentrations, but radiolytic as well as thermally induced reactions are taken into account. The code also assumes a flat temperature profile in web graphite.³

RICHTER (United Kingdom Atomic Energy Research Establishment, Harwell). Only limited information has been obtained on the RICHTER code. The following aspects are included in the computational programs: (1) reaction rates for the steam-graphite, carbon dioxide-graphite, and the water-gas shift reactions; (2) steady-state diffusion of the steam through the graphite; (3) inclusion of a bulk flow mechanism in addition to diffusion; and (4) variation of the steady-state diffusion parameters to simulate effects of burnoff.⁴

LIGHTER (United Kingdom Atomic Energy Research Establishment, Harwell). The limited information available on the LIGHTER code indicates it to be a transient code that can be used to calculate core corrosion under conditions of rapid steam ingress, as in the case of a burst heat exchanger. Cylindrical geometry is used, and diffusion (in the radial direction only) is the sole mechanism of transport in the graphite. An analysis of the entire core region can be made, but unlike RICHTER, attack of the graphite by CO_2 and the effects of the water-gas shift reaction are not included. Although the program allows, as input, the time derivative of temperature as a function of mesh point within the graphite, in practice the derivative is set equal to zero, thus forcing a fixed temperature distribution within the system. Comparisons of core corrosion determined by the steady-state code RICHTER and the transient code LIGHTER have shown remarkably good agreement.⁵

TUBER (Dragon Project). Information on this code is likewise limited. A transient analysis of the corrosion of graphite by steam is performed. Some of the noteworthy aspects of the code include the use of a two-radial-zone heat transfer model that divides the core thermal capacity between the bulk moderator and the coolant channel walls, the use of an Arrhenius-type chemical reaction model to integrate the corrosion rate over the entire core (with reactivity factors that can vary from coolant channel to coolant channel), and a bulk transport model to describe steam penetration into the core graphite, but with no provision for Fick's Law type of diffusion.⁶

None of these codes has been systematically tested experimentally, though the results obtained from an application of OXIDE have been compared with laboratory-scale experiments⁷ and some steam injection experiments have been performed in the Dragon Project reactor for purposes of code testing.

In addition, the Pegase loop experiments in France may permit further testing of the OXIDE and the GOP codes, but the results may be regarded as proprietary. Startup of the Fort St. Vrain reactor offers a unique opportunity to determine the significance of radiolytic reactions and thus a test of the GOP code.

Needs and identified work

Thus far no detailed intercomparison of the codes has been made, nor have any of the codes been independently examined with regard to the assumptions involved, the mathematical derivation and numerical techniques employed, the inherent limitations of the techniques used, or the inclusion of all pertinent phenomena. Such an analysis and intercomparison must be performed, and any objections which may be raised must be resolved.

Sufficient experimental data exist to be used in parameter sensitivity analyses by extant or simplified computer codes. However, the analyses attending much of the experimental work are open to question. This is not a serious deficiency insofar as parametric sensitivity studies are concerned, but additional data are needed to define graphite oxidation levels more precisely.

The purpose of the sensitivity analyses is to establish guidelines for the required accuracy of code input data, and logical consistency requires that the analyses be attuned to and consistent with certain natural limitations on the system. For example, the rate of corrosion of the graphite by steam depends on the internal geometrical properties of the graphite, and some variation will

occur with respect to axial and radial position within the fuel element, from fuel element to fuel element of the same batch, from batch to batch of the same graphite type, and from graphite type to graphite type. The extent to which these variations occur should be established, and the accuracies desired of input data should reflect this natural constraint.

Additional experimental data should be sought to test the code, to provide more accurate input data, or both; laboratory-scale experiments should be used. Final validation of the computer model will require large-scale experiments, however, as well as an in-pile experimental program. These experiments should be few; they will sogma; the attainment of the research objectives.

The French Pegase experimental loop can in principle provide valuable data with which to examine the computer programs; an effort should therefore be made to obtain these data. Moreover, startup of the Fort St. Vrain reactor offers a unique opportunity to determine the significance of radiation-induced reactions. Such a determination should be made with a suitably modified version of the GOP code. In like manner, an attempt should be made to test OXIDE with Fort St. Vrain reactor startup data.

10.6.2 Kinetics of graphite oxidation (task 3.1.1.1)

Justification and scope

This multifaceted task has a threefold objective: (1) to provide input data for a computer code for analysis of steam ingress into an HTGR, (2) to ensure that the modeling studies omit no significant phenomena, and (3) to provide data for code validation. Most of the effort involves laboratory-scale experimentation. Significant effort should be expended to assess the variability of internal geometries of the fuel elements, because this aspect provides an inherent limitation to the reliability of predicted system response to steam ingress.

Status

Although considerable effort has been spent to elucidate the kinetics of the steam-graphite reaction, a treatment free of empiricism has not been attained. As a result, much of the data obtained to date are based upon physical analyses of the problem which are open to question; moreover, in many cases those parameters that have a significant influence in determining the rate of reaction (e.g., internal surface area) or the mechanism (e.g., the porosity-tortuosity ratio) have simply not been determined.

Two factors control the rate of corrosion of graphite by steam - the rate at which the steam molecules strike the available graphite surface and the fraction of steam-graphite surface contacts that result in chemical reaction. However, because of the porous nature of the graphite, the interaction of these two factors produces three different kinds of reaction control. At sufficiently low temperatures, the probability of reaction and the rate of gas-surface collisions are not coupled, and one can thus study the reaction kinetics directly. This circumstance is frequently designated as "reaction-controlled." At somewhat higher temperatures, sufficient depletion of steam occurs as a result of reaction to develop a gradient in steam concentration in the interior of the graphite specimen, and the supply of steam to sustain the reaction is limited by the rate at which the steam molecules can diffuse into the pores within the graphite. This condition, known as "in-pore diffusion control," results in a coupling (i.e., an interdependence) of the probability factor and the gas-surface collision factor, so that the two effects must be decoupled (with another mathematical model) to obtain information on the reaction kinetics. At still higher temperatures, the probability of reaction becomes so large that no penetration of the interior of the graphite by steam occurs, and the rate of reaction becomes controlled solely by the rate at which the steam molecules strike the available graphite surface. This condition is designated as "gas-phase diffusion control," and once again the two factors that regulate corrosion become decoupled, though obviously no information can be obtained in this case with regard to the reaction kinetics.

Because of the three different types of reaction control, extrapolation of experimental data beyond the temperature range over which the measurements were made can be invalid. Moreover, no experiments have been performed to establish directly the temperature regions over which the three types of corrosion control are operative. Such a set of experiments should be conducted.

All of the various forms for the rate expression that have been proposed thus far can be considered to be degenerate forms of the Hinshelwood generalized expression

$$R = \frac{K_1 P_{H_2O}^m}{1 + K_2 P_{H_2}^n + K_3 P_{CO}^q + K_4 P_{H_2O}^r},$$

in which R is the rate of graphite corrosion (in appropriate units), the P_i are partial pressures of the corresponding species i , and the rate constants K_j have the form

$$K_j = k_j \exp (-E_j/RT) .$$

One set of parameters employed to analyze effects of steam ingress into an HTGR⁸ is

$m = 1$	$k_1 = 2.67 \times 10^9$ (%/hr-atm)	$E_1 = 40.9$ (kcal/mole)
$n = 3/4$	$k_2 = 1.66 \times 10^{-2}$ (atm ^{-3/4})	$E_2 = -28.6$ (kcal/mole)
$q = 0$	$k_3 = 0$	$E_3 = 0$
$r = 1$	$k_4 = 5.31 \times 10^{-2}$ (atm ⁻¹)	$E_4 = -27.5$ (kcal/mole)

In these units the rate of corrosion is expressed as percent of graphite removed per hour. These parameters, which resulted from a recent analysis of experimental data (but which did not include data for H-327), appear to provide a fair representation of experimental data for H-327 graphite.⁸ By contrast, a second set of parametric values which has also been proposed⁹ to account for steam attack is

$m = 1$	$k_1 = 2.63 \times 10^{14}$ (%/hr-atm)	$E_1 = 68$ (kcal/mole)
$n = 1$	$k_2 = 4.51$ (atm ⁻¹)	$E_2 = -14.5$ (kcal/mole)
$q = 1$	$k_3 = 3.66$ (atm ⁻¹)	$E_3 = 14.5$ (kcal/mole)
$r = 0$	$k_4 = 0$	$E_4 = 0$

These values present a rather extreme example of divergence from the previous set. Moreover, in this formulation the parameters k_j actually vary inversely with temperature; the numerical values presented above correspond to 1000°K. In short, caution should be exercised in applying rate data determined under one set of conditions and with one type of graphite to describe the behavior of another type of graphite or other sets of conditions.

Factors that have a possible influence on the graphite corrosion rate include:

- (1) Dependence upon steam partial pressure. Virtually all of the studies conducted thus far have been performed at low steam pressures and low total pressures, so that the reliability of extrapolations to elevated pressures is subject to confirmation. Moreover, even the form of the dependence at low pressures has not been unambiguously defined.¹⁰
- (2) Inhibition by reaction products. Malinauskas and Blood¹¹ have clearly demonstrated the inhibiting effect of the reaction products on the rate of corrosion of H-327 graphite. The effect had in fact been observed previously by other workers with other graphite types.¹⁰ However, the functional dependence upon hydrogen and carbon monoxide partial pressure remains to be established.¹⁰
- (3) Temperature dependence of the reaction rate. The activation energy characteristic of the overall reaction specifies the temperature dependence of the reaction rate. For most simple reactions, the activation energy can be specified within about 5 kcal/mole. For the steam-graphite reaction, however, the literature values¹⁰ of the activation energy range between about 50 and 80 kcal/mole. Part of the problem is due to the mechanisms that give rise to inhibition by the reaction products, as these also involve activation processes. Another factor has been the lack of concern for the geometrical aspects; thus, for example, the characteristic activation energy for in-pore diffusion control is about one-half that for chemical control,¹⁰ and one must ascertain the nature of the rate-controlling mechanism in order for the activation energy value to be meaningful.
- (4) Effect of burnoff. Most investigations of graphite corrosion by steam use specimens that have been pre-oxidized to about 5% initial weight loss.¹⁰ Because no effort was made to characterize the geometrical nature of the sample in this state and because burnoffs exceeding 5% are most unlikely, such data should be used with care to specify effects of steam inleakage in an HTGR. Moreover, the effect of burnoff has been clearly shown to be a manifestation of alterations in internal geometry of the graphite specimens.^{12,13} A mathematical model to describe these alterations is lacking, although an empirical approach yields satisfactory agreement with experiment for H-327 graphite.⁹
- (5) Transverse flow effects. Effects of forced flow of helium-steam mixtures through graphite involve no new phenomena; so this condition could be as accurately modeled as the case involving in-pore diffusion control. Additional input data

are required, however, on permeability coefficients and changes in permeability with extent of reaction. Such data are in hand for H-327 graphite,¹³ but a model to describe the variation with extent of reaction has yet to be developed. Additionally, verification of the postulated behavior of corrosion rate under forced flow conditions remains undone.

- (6) Effects of catalysts. The qualitative effect of enhancement of reaction rates due to metallic impurities is well known;¹⁰ however, the quantitative modifying effects appear to be confined mostly to the description presented by Burnette and Zumwalt.¹⁴ Fortunately, most impurity catalysts can be eliminated in the manufacture of the fuel elements, and catalysis by fission products appears to be confined to barium and strontium.¹⁴ However, it is important to note that migration of catalytic impurities has been observed.^{15,16} This can result in the transfer of impurities to purified fuel elements from other sources within the system.
- (7) Spatial internal geometrical effects. The variation of geometrical factors that have a direct bearing on the rate of reaction (surface area) or on the rate of transport of steam through the graphite (permeability, porosity, porosity-tortuosity ratio) will most likely limit the reliability with which a given sequence of events attending steam inleakage can be described. The extent of these variations should be investigated. The experimental work involved is straightforward and uncomplicated, but tedious. Study of this aspect has been very limited thus far and has involved materials that are not very closely related to HTGR candidate graphites.¹⁷
- (8) Effects of graphite irradiation. The somewhat limited amount of data indicates that such effects are negligible.¹⁸
- (9) Radiolysis reactions. Although one group of investigators contends that radiation-induced reactions will present no significant problem in the coolant chemistry of helium-cooled steam generating HTGRs,⁹ this position is questioned on the basis of current information.³ In fact, computer studies that have employed available reaction data indicate that radiation-induced processes significantly affect the overall chemistry of the coolant.³

Needs and identified work

As indicated above, many factors must be considered in providing an accurate description of the physicochemical aspects of steam-graphite corrosion. To some extent these factors can be used as a basis for differentiating between the various research activities, but several of these are so closely interrelated that they involve the same experimental facility. The activities are identified by affixing an alphabetic character to the common task number identification.

Estimate of graphite variability (task 3.1.1.1a). A natural limitation on the accuracy with which the consequences of steam ingress can be described is imposed by the variability of the graphite materials with regard to corrosion. Some estimate must be had of the reactivity (independent of coolant composition and temperature considerations) along the length of a given fuel channel in a given fuel element, between channels in a given fuel element, and between like channels of different fuel elements. If it can be demonstrated that the reactivity differences are due solely to geometric factors, and not to variations in catalytic impurities, then the extent of variability can be investigated using simpler, nondestructive geometrical characterizations (surface area, permeability, and interdiffusion measurements) rather than direct corrosion determinations. This is an important consideration, since a large (but not unwieldy) number of samples are necessarily involved.

In like manner, it is entirely possible that the fuel elements, the reflector blocks, the graphite block floor, and the support columns will be of different graphite types; and the possibility of comparing their relative corrosion behavior on the basis of geometric factors should be explored. The reactivities of the matrix material and the pyrocarbon coatings of the fuel particles should also be considered in attempting the correlation. Should such a correlation prove possible, actual corrosion investigations could be drastically reduced to only a few confirmatory tests.

Effects of elevated pressure (task 3.1.1.1b). There is a paucity of experimental corrosion data under the pressure conditions that are postulated for the reactor. Although difficult to obtain, data at high helium and high steam pressures can aid in the clarification of two important factors. Experiments should be conducted at helium pressures in the range 0 to 40 atm and at some convenient but fixed steam pressure and temperature to establish unambiguously the dependence of the corrosion rate on total system pressure. In this way it should be possible to ascertain whether in-pore

diffusion or chemical reaction is the rate-controlling mechanism at that temperature as well as to determine the likelihood of there being a transition from one mode to the other as the pressure is increased.

The first of the two sets of rate constants presented earlier indicates that the rate of reaction becomes independent of steam partial pressures at sufficiently high steam concentrations. This is an important factor; if verified, this circumstance would impose an upper limit on the range of steam partial pressures that have to be considered and likewise cause some reduction in the experimental program.

Identification of the reaction regimes (task 3.1.1.1c). It is absolutely essential that the rate-controlling mechanism be identified before any attempt is made to determine rate constant values from experimental data. This is best established by performing the experiments at various helium pressures at different temperatures over the range 700 to 1200°C.

Determination of rate constants (task 3.1.1.1d). A complete set of rate constants should be determined using the reference fuel element graphite. These constants are best evaluated by performing corrosion experiments in which temperature and steam, hydrogen, and carbon monoxide partial pressures are varied systematically.

A limited number of experiments should be conducted with all of the graphitic materials identified earlier in an attempt to correlate corrosion rate with geometric parameters. Should such a correlation prove unlikely, more detailed reaction rate measurements should be delayed until the computer code sensitivity analyses indicate the additional data to be desirable.

Variation of geometric parameters with extent of reaction (task 3.1.1.1e). As already indicated, the observed variation of reaction rate with burnoff has been demonstrated to be a manifestation of changes in internal geometry. Although a purely empirical approach has proved to be fruitful in this regard for H-327 graphite, a theoretical description of the processes involved is most desirable. So long as the appropriate geometric characterizations are made in the course of the corrosion studies, this activity requires no additional experimental work.

Transverse flow effects (task 3.1.1.1f). A limited number of experiments should be conducted with a geometrical configuration which simulates a single coolant channel and the surrounding fuel compartments (but under isothermal conditions) to examine the extent and the effect of transverse flow on the corrosion characteristics. Initial experiments should be conducted with unloaded but blind fuel compartments, whereas later experiments should simulate matrix, shim, and fuel particles in the fuel compartments. These experiments would likewise provide data for code testing purposes and for later, more complex, code validation experiments.

Catalysis studies (task 3.1.1.1g). Variations in corrosion behavior from graphite type to graphite type or within batches of the same type may more properly reflect catalyst impurity concentration changes rather than internal geometry. If so, this aspect should be further evaluated and an appropriate experimental program developed. In like manner, the subject of catalysis by fission products should be re-examined, and whatever experimental work that is required to resolve the problems that may arise should be performed.

Measurement of yields of radiolytic reactions (task 3.1.1.1h). One of the objectives of the analysis of Fort St. Vrain reactor coolant during startup, from the standpoint of the GOP program, is to resolve the question of the significance of radiolytic reactions in determining coolant composition. Should these reactions prove to be significant, the phenomenon will be incorporated in the computer code to be used for steam ingress analysis and an assessment made of the advisability of additional measurements. If further measurements are required, an experimental program should be developed, and the yields of the appropriate radiolytic reactions should be determined.

Air-graphite reaction measurements (task 3.1.1.1i). Further experimentation in this area is presently felt to be unwarranted. Should accident analyses using available data indicate the desirability for further refinement, the necessary data should be obtained. This poses no experimental difficulties.

10.6.3 Validation of steam ingress computer code (task 3.1.3.1)

The concluding phase of this work should involve the operation of a large-scale loop that can simulate coolant flow conditions and temperatures. This loop should be operated primarily to validate the steam ingress computer code, but by proper design it can also

provide information on coolant-metal interactions of interest to task group 3.2. The extent to which validation tests with the large-scale loop are necessary will depend in part on the manner in which the Pegase loop is operated as well as on the extent to which the French data are made available to the AEC.

A lesser number of in-pile experiments should also be performed as an added check on the effects (or lack thereof) of radiation on the corrosion behavior. These experiments should be of a much smaller scale than the code validation experiments noted above, but, like these, the extent to which the in-pile experiments need be performed depends upon the nature and availability of the Pegase results.

10.6.4 Reactions involving fuel kernels (task 3.1.1.2)

Justification and scope

Hydrolysis of exposed carbide fuel can occur if steam contacts the fuel as a result of transport through the fuel-block graphite. Exposed carbide fuel could result from failed fissile (UC_2) particles and from failed fertile (ThO_2) particles exposed to high enough temperature to convert ThO_2 to ThC_2 . During full power operation, steam can permeate the graphite and react with exposed fuel only in the upper, relatively cool portion of the core where depletion by reaction with graphite will occur to a lesser extent than in the hotter core regions. During shutdown conditions, on the other hand, water can diffuse throughout the core graphite, thereby subjecting all exposed carbide fuel to possible hydrolysis reactions.

Status

Hydrolysis of carbide fuel can cause swelling, release of stored fission gases, and increased steady-state fission-gas release (R/B). A study to determine the effect of swelling on graphite fuel-block integrity indicated that up to 50% hydrolyzed heavy metal in a fuel rod can be tolerated.¹⁹ In a large HTGR, only about 5% of the heavy metal in the initial core is in the form of carbide (UC_2); the remaining 95% is ThO_2 . The maximum (design) failure fraction is 10%. The effects of swelling, therefore, would appear to be of little consequence.

A study of fuel hydrolysis and concomitant release of stored fission gases indicated that only noble gases are released to an appreciable extent during hydrolysis.²⁰ In a recent study at

GAC²¹ an irradiated fuel rod (containing all carbide fuel with 4% exposed carbide) was exposed to moisture at 900°C and subsequently annealed in dry helium at 1100°C. The overall increase in R/B was less than a factor of 3.

Hydrolysis of exposed carbide fuel under shutdown conditions (following a steam inleakage accident, for example) is expected to be of little consequence because the rate of hydrolysis of carbide fuel at shutdown temperature ($\leq 300^\circ\text{C}$) is greatly reduced after irradiation. This passivation effect is believed to be the result of a thin impervious oxide coating which forms in the early stage of hydrolysis.

Needs and identified work

The effect of carbide fuel hydrolysis on swelling, release of stored fission gases, and steady state R/B needs to be studied further to verify existing information and to gain a better understanding of the hydrolysis process. Hydrolysis temperatures should range from 200 to 1100°C. Water concentrations should range from 10^{-4} atm (equivalent to 2 vpm water in the primary coolant at 50 atm) to 1 to 2 atm (equivalent to water concentrations that could develop in the event of a large steam leak). Particular emphasis should be placed on studies of the effect of irradiation on the rate of hydrolysis at $\leq 300^\circ\text{C}$ (reactor shutdown temperatures).

10.6.5 Structural strength of graphite (task 3.1.1.3)

Justification and scope

This task provides information on the effect of oxidation on the strength of graphite. Accurate predictions of levels of graphite oxidation and consequent loss of strength in graphite components are needed for use in establishing maximum oxidant levels in the primary coolant and for use in designing graphite components. Information is needed for fuel-block graphite, for lower-reflector graphite, and for core-support graphite. Strength loss information on the latter two graphites is of particular importance, since they remain in the reactor over its 40-year lifetime.

Status

It is known that oxidation reduces the strength of graphite and that the extent of strength loss depends on the manner in which oxidation occurs. At low temperatures, for example, oxidation

occurs uniformly throughout the material, and the loss of strength with burnoff is relatively high. At high temperatures, oxidation is not uniform but is instead restricted mainly to the surface, so that relatively low loss of strength with burnoff results.

The effect of burnoff on graphite strength has been studied. In one study,²² graphite samples were oxidized at low pressure at 900°C; at this temperature the burnoff was observed to be relatively uniform throughout the samples, thus maximizing the effect of burnoff on strength. In another study,²³ the graphite was oxidized at 1000°C; as anticipated, lower strength losses with burnoff were found.

Needs and identified work

The effect of steam oxidation on graphite strength needs to be studied for the fuel-block, lower-reflector, and core-support graphites that will be used in the large HTGR. Because all of the parameters that affect the rate of oxidation of the graphite also determine the degree of strength loss with oxidation, to some extent the studies can be performed along with the rate investigations of task 3.1.1.1.

10.6.6 Reactions involving fission products (task 3.1.1.4)

Justification and scope

Steam ingress into the coolant circuit of an HTGR, and the resultant reaction with the graphite core, introduces at least three new species — H_2 , H_2O , and CO — that can alter fission-product transport and distribution as a result of chemical reaction with the fission products. Thus, for example, the determination of whether some of the fission products are in oxide, carbide, or elemental form will depend upon the H_2/H_2O ratio, and the manner and extent of the migration of the fission products, in turn, will depend upon their chemical form.

The scope of this task is in many respects closely related to activities of task area 2 but is limited to a consideration of the effects of steam, hydrogen, and carbon monoxide on establishing the chemical forms of pertinent fission products.

Status

Previous studies performed in connection with the Light Water Reactor Safety Program indicated an enhanced volatility of certain fission products (notably iodine, ruthenium, and tellurium) as a result of exposure to steam.²⁴ In the case of the latter two fission products, the enhanced volatility may result from the formation of a gaseous oxyhydroxide compound.²⁵ The existence of gaseous $\text{TeO}(\text{OH})_2$ has in fact been postulated to account for the observed volatility of TeO_2 in the presence of steam.²⁵ Furthermore, because the oxyhydroxides are formed at elevated temperatures and appear to undergo decomposition at lower temperatures,²⁵ this phenomenon provides a mechanism for transport of certain fission products out of the hot reactor core reaction.

In contrast, carbonyl compounds decompose readily (sometimes explosively) at elevated temperatures.²⁶ These compounds, which appear to be restricted to the transition elements of Groups VI [such as $\text{Mo}(\text{CO})_6$], VII [$\text{Re}_2(\text{CO})_{10}$], and VIII [$\text{Ru}(\text{CO})_5$]²⁶ may thus limit deposition of certain fission products that would normally tend to concentrate in cooler regions.

Needs and identified work

A limited number of transpiration-type experiments should be performed to assess the significance of oxyhydroxide, carbonyl, or other compound formation on fission-product release and transport. These experiments can be conducted by exposing irradiated coated particle fuel compacts (having extensive particle failure) to various $\text{H}_2\text{O}-\text{H}_2-\text{CO}$ mixtures by means of a helium sweep stream. Additional studies on specific fission products should follow these scoping type experiments only as the results of these experiments indicate the desirability of additional data.

10.6.7 Flammability studies (task 3.1.1.5)Justification and scope

Information on the lower flammability limits of mixtures of gases containing air, H_2O , H_2 , and CO is used in the analyses of hypothetical accident situations, involving mixing of containment-building air with H_2 and CO formed by the reaction of steam (or air) with core graphite. Accident situations are (1) venting of helium and steam-graphite reaction products (CO and H_2) through a pressure relief valve following a steam leak and moisture-monitor failure, (2) convection of building air into the PCRV following

a penetration failure in the top of the PCFV, and (3) occurrence of two simultaneous or consecutive accidents, a steam-leak accident and a depressurization accident. The work in this task involves laboratory-scale experiments, and the needed data consist of flammability limits obtained under representative reactor conditions.

Status

Literature data exist on the flammability limits of H_2 -air and CO-air mixtures. These data have been used in the Le Chatelier equation to estimate flammability limits for H_2 -CO-air mixtures. Experiments have been carried out at GAC to determine the limit of flammability of H_2 and CO mixtures (1:1) in air and helium. The experiments were carried out at room temperature using 2-in.- and 3-in.-diam combustion tubes. The work yielded lower limits of flammability which are slightly higher than theoretical values of the lower limit calculated using the Le Chatelier equation.

Needs and identified work

Experiments are needed to establish the effect of gas-mixture temperatures up to $300^\circ C$ on the flammability limits for mixtures containing air, H_2O , H_2 , CO, and He, since the lower limit is known to decrease with increasing temperature. Experimental work is also needed to establish the effect of volume (wall effects) on the lower flammability limit. Until the range of concentrations of flammable species is better defined, however, additional work in this area should be deferred.

10.7 PRIMARY COOLANT-COMPONENT COMPATIBILITY (TASK GROUP 3.2)

Task group 3.2 is directed toward the acquisition of detailed information on safety-related effects of coolant impurities on structural metals in HTGR environments. As this is a common goal of the three tasks of task group 3.2, it is convenient to discuss some of the aspects of the three tasks as a unit and to limit the individual task discussions to differences in detail.

10.7.1 Scope and justification

The potential problem of corrosion of structural materials by the small but finite level of impurity gases present in the primary coolant helium of an HTGR was recognized early in the development of this reactor system. More recently it has been realized that carburization of these materials is also a possibility.

Permissible amounts of impurity gases in the HTGR coolant are presently restricted to very low levels (e.g., HTGR technical specifications limit total oxidants to 10 vpm). Under normal conditions the impurities should be primarily H_2 , CO, and CH_4 . However, during startup, or in the event of minor steam leaks from the steam generator, H_2O may also be present. Relatively high levels of impurities will exist should an accident occur, but since these conditions are of short duration, the degree of reaction experienced during this period should not represent a significant hazard with respect to the metallic components.

The long service life (30 to 40 years) for which HTGRs are designed makes an improvement in understanding and confidence desirable in all areas related to coolant impurity reactions with the materials of construction. For example, while impurity levels can be restricted to small values by the use of purification systems, it is difficult to predict in advance the precise chemical balance that will exist between impurities (i.e., CO/ CH_4 / H_2 , etc.) in an operating reactor. In view of this, additional knowledge of the effects of impurities (singly and in combination) is needed to assure that the high-temperature metallic components of an HTGR will remain immune from significant degrees of oxidation, carburization, and other damage during their lifetime.

The emphasis in this task group will be on providing the necessary assurances of the above by performing experiments designed to examine the corrosion behavior (both surface and intergranular oxidation) of the primary loop structural materials and their potential for carburization and other reactions. In addition, as fission-product retention is influenced by the nature and character of surface films, these studies will provide valuable information for use in task groups 2.2 and 2.6 (fission-product distribution in the coolant circuit).

10.7.2 Status

Relatively extensive testing was performed in the early 1960's to examine the effects of coolant impurities on HTGR structural metals.²⁷⁻³² These early studies were conducted with the assumption that the principal coolant impurities would be equal amounts of CO and H_2 . Subsequent to these tests, operating data from Peach Bottom, Dragon, and AVR reactors have shown that this may not be entirely valid. For example, a finite level of CH_4 (1 to 2 vpm) has been observed in Peach Bottom during steady state operation. Since this was unanticipated, the effects of CH_4 (a gas with significant carburizing potential) were not included in the early studies. Further, carburization has been found in components removed from

this reactor. Neither the kinetics nor the mechanism of this phenomenon has been defined, but the extent of reaction observed is greater than would be predicted from earlier experiments.

The results of the earlier experiments demonstrated that ferritic alloys were relatively unaffected by CO and H₂ up to 600°C. Austenitic materials, on the other hand, carburized slightly (but not to the extent seen in Peach Bottom) at 600 and 750°C. Carburization experiments in CH₄-H₂ mixtures are in progress at GAC³³ at temperatures from 550 to 750°C. Additional studies on the effects of CO-H₂ mixtures are being performed for GAC by the Austrian Center for Atomic Energy Studies.³⁴ Finally, the effects of impure helium are being studied in connection with the Dragon project by the UKAEA. None of the studies referenced or in progress provides sufficient high-temperature (750 to 1000°C) data on the effects of helium impurities. Such information is necessary because some portions of the primary circuit will see local temperatures on the order of 950°C.

10.7.3 Needs and identified work

Reactions of coolant impurities with HTGR structural materials (task 3.2.1.1)

In these studies, structural materials should be exposed to various oxidizing and carburizing environments for periods up to several years at temperatures appropriate to their use in HTGRs. Materials to be tested should include alloys (Incoloy 800, Hastelloy X, 2-1/4 Cr-1 Mo steel, 1010/1020 carbon steel) and weldments (Incoloy 800 to Incoloy 800, Incoloy 800 to 2-1/4 Cr-1 Mo, Hastelloy X to Hastelloy X, 2-1/4 Cr-1 Mo to carbon steel) of current use and alloys of high potential for use in the near future (e.g., Hastelloy S and Inconel 617).

After exposure, the specimens should be examined by techniques including optical and electron metallography, electron probe microanalysis, x-ray diffraction, and Auger electron spectroscopy. These examinations would permit evaluation of (1) the kinetic/mechanistic aspects of corrosion, oxidation, carburization, etc., (2) the metallurgical or microstructural stability under these conditions, and (3) the chemical and morphological character of the surface films. In every instance the work in this task should be interfaced and closely coordinated with that in task 6.1.1.1 (mechanical properties and failure criteria for HTGR structural metals).

Examination of components from Peach Bottom HTGR (task 3.2.3.1)

Samples of steam generator tubing, the steam-generator support structure, hot ducting, metallic thermal-barrier materials, hot valve internals, and parts of the circulator from the Peach Bottom reactor should be examined by techniques similar to those described under task 3.2.1.1. Evidence related to corrosion, carburization, and metallurgical stability will be sought, and surface films should be identified. Correlations between reactor operating history and these observations should be sought.

Examination of specimens from Fort St. Vrain HTGR (task 3.2.3.2)

Metallic components exposed to the primary coolant helium in the Fort St. Vrain reactor should be examined as they become available. Control rods, for example, could be examined on a four- to six-year cycle as they are replaced. The method and purpose of examination are as described above in task 3.2.3.1.

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11. SEISMIC AND VIBRATION TECHNOLOGY (TASK AREA 4)

11.1 TASK DEFINITION AND OBJECTIVES

The tasks in this area have the common goal of providing analyses of safety-related behavior involving modes of failure and responses. The factors considered in planning these tasks are mechanical loadings induced by earthquakes and gas flow effects. The dominant discipline in this group is structural mechanics.

11.2 SCOPE

The task area is divided into the following six task groups:

4.0 Need assessment

4.1 Core seismic response

4.2 Component seismic response

4.3 Soil-structure dynamic interaction

4.4 PCRV support

4.5 Reactor internals vibration

The components concerned in these six task groups are all important to the safe operation of a reactor.

The first task group is generic in nature. It will evaluate needs for all aspects of concrete structure and component behavior under seismic and vibration loadings. Task groups 4.1-4.4 involve research aimed at providing information required to identify and when necessary reduce the response of HTGRs to seismic loadings. Included are studies of the core structure and the various operation and support structure components to seismic loadings. Task group 4.5 is concerned with defining responses of components subject to vibration induced by gas flow.

Certain HTGR system components have been conspicuously omitted from these task listings. Components such as control rod drives and circulators that should conceivably be included in these safety-related studies are already well developed and have been used in the Fort St. Vrain HTGR. The potential failure of every element of a reactor system must be considered; however, only those whose failure could lead to serious consequences should be selected for inclusion in the safety program.

11.3 STATUS AND NEEDS

The status and needs of each task group are discussed in general terms in this section. A more detailed discussion of many of these task groups is included as part of Sect. 11.4.

Current HTGR designs call for the core to consist of vertical columns of hexagonal elements supported on a graphite floor and keyed together horizontally at the top layer only. The columns have small clearance gaps between each other. Each hexagonal element is doweled to the element above and below it in the column, and the lowest hexagonal element in each column is doweled to the core support floor. Elements can rock on each other to the small extent permitted by limited clearances within the core. The circumferential boundary of the core is restrained by flexible lateral supports mounted on the inner surface of the PCRV.

Thus, the response of the core to the horizontal seismic input becomes a problem of many individual elements that can move in horizontal planes within the limits of the movement of the plenum layer and core support floor, the flexibility of the columns, the restraint of the dowels, and the confining proximity of the adjacent columns.

In addition to the core seismic response, the maximum impact threshold for chipping or cracking of the fuel element should be known, since fuel elements in the HTGR core would experience numerous collisions with elements in adjacent columns during earthquakes. If the collisions are sufficiently numerous and energetic, the graphite components might be weakened or might crack. There is a need for further understanding of this behavior as well as for analytical methods to determine the time-history dynamic response of core structures.

Under earthquake conditions, main steam generators must be shut down and core cooling maintained. This requires satisfactory operation of many components such as the rods, actuators, and auxiliary cooling-loop backflow prevention devices. Moreover, the integrity of other devices is necessary to prevent compounding of the plant contingency condition. Analytical methods are needed for describing the seismic responses of these vital components.

The effects of soil-structure dynamic interaction on the seismic response of nuclear power stations have long been recognized as being significant. The response of the overall plant structure in turn governs the input into all the dynamic subsystems such as the reactor core, steam generators, etc. The transmission of energy

from the base slab into the soil normally results in lower response levels than those predicted when the interaction effects are neglected. However, for certain ranges of parameters, increased levels can result, particularly when the effects of adjacent structures are considered.

Although theoretical formulations for nonlinear soil properties are in the development stage, they have not been completed. Some experimental studies have also been performed, but these tests have been confined to forced vibrations of rigid footings on soil or very low-level forced vibrations of an actual nuclear plant having a particular site condition. While these tests are informative, they do not provide design data for a range of possible ground conditions and acceleration levels. Consequently there is a need to conduct scale-model tests to provide sufficient data for general analysis verification.

The PCRV support method being applied to HTGR plant designs provides essentially a rigid connection between ground and structure. An alternative method would be to decouple the PCRV from ground seismic forces by combinations of flexible supports, dampers, and slip members, thereby reducing seismic loading on the structure and plant equipment. Any such reduction in seismic loading would enhance plant safety under earthquake conditions, particularly where the seismic intensities are high. There is need for a study of new types of support systems and the adaptation or development of analyses to be used in this work.

In addition to seismic loadings reactor internal structures such as the side-restraint structure, core-support structure, thermal-barrier cover plates, steam generator, and internal seals are subjected to vibrations created by circulation machinery and/or flow-induced turbulences. There is a need to determine specifically how the requirements of the AEC Safety Guide Standards (particularly those in Safety Guide 20) are to be met by a study of methods of predicting vibration effects and measuring vibrations on-site.

11.4 TASK AREA EFFORT AND SCHEDULE

The associated tasks and information flow of the task groups are shown in Figs. 11.1 and 11.2. Task groups 4.0 through 4.4 are shown collectively in Fig. 11.1 since they are concerned with various aspects of seismic effects. Figure 11.2 shows the same information for task group 4.5. Table 11.1 gives the relative effort of each task.

Fig. 11.1. Task groups 4.0 - 4.4. Seismic effects investigations.

Fig. 11.2. Task group 4.5. Reactor internals vibration.

Table 11.1. Summary of identified safety and safety-related work in task area 4

Task groups or tasks marked with an asterisk (*) are included
in the safety planning guide (Part II of this document).

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 4.0 Need assessment*	1.0	1.0	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5
4.0.1.1 Need assessment and technology review										
Task group 4.0 totals	1.0	1.0	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5
Task group 4.1 Core seismic response*										
4.1.1 Source tasks										
4.1.1.1 Seismic motions and effects		1.0	2.0	2.0	2.0	2.0	1.3	0.7		
4.1.1.2 Core scale model seismic tests	8	12	16	20	13	9	4			
4.1.1.3 Fuel element impact tests	1	2	2	1.5						
4.1.2 Synthesis tasks										
4.1.2.1 Analytical model of core seismic response	1	4	4	4	4	2.5	1.5			
Task group 4.1 total	10	19	24	27.5	19	13.5	6.8	0.7		
Task group 4.2 Component seismic response*	0.7	3.3	6.7	10.7	8	4				
4.2.1 Source tasks										
4.2.1.1 Seismic induced component motions and their effects										
4.2.1.2 Component seismic response tests										
4.2.2 Synthesis task										
4.2.2.1 Analytical models of component seismic response										
Task group 4.2 total	0.7	3.3	6.7	10.7	8	4				
Task group 4.3 Soil-structure dynamic interaction*	1	2	3.3	3.3	1.4					
4.3.1 Source tasks										
4.3.1.1 Soil-structure interaction										
4.3.1.2 Soil-structure seismic model tests										
4.3.2 Synthesis task										
4.3.2.1 Analytical models of soil-structure seismic response										
Task group 4.3 total	1	2	3.3	3.3	1.4					
Task group 4.4 PCRV support*		2.4	5.3	6.7	5.3	5.3	4	2.7	1.3	
4.4.1 Source tasks										
4.4.1.1 PCRV support system concepts										
4.4.1.2 PCRV support model tests										

Table 11.1 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
4.4.2 Synthesis task										
4.4.2.1 Analytical models of PCRV seismic response										
Task group 4.4 total		2.4	5.3	6.7	5.3	5.3	4	2.7	1.3	
Task group 4.5 Reactor internals vibration	1.3	1.3	2.7	2.7	2.7	1.3	1.3			
4.5.1 Source tasks										
4.5.1.1 Vibration excitation and component response										
4.5.1.2 Component vibration response tests										
4.5.2 Synthesis task										
4.5.2.1 Analytical methods for predicting component vibration										
Task group 4.5 total	1.3	1.3	2.7	2.7	2.7	1.3	1.3			
Task area 4 total	14	29	43	51.9	37.4	25.1	12.6	3.9	1.8	

The following sections give a general description of the proposed studies as organized under the appropriate task group.

11.5 NEED ASSESSMENT FOR SEISMIC AND VIBRATION TECHNOLOGY (TASK GROUP 4.0)

This task will provide a general safety assessment of seismic and vibration technology for HTGRs with particular emphasis on PCRV structure, core, and components.

11.5.1 Scope and justification

Efforts are under way to acquire experimental data and to improve methods of analysis for determining the effects of seismic loadings and operational vibrations on PCRVs. However, there is need for an independent review of the HTGR system with respect to its unique structural components and for an evaluation of the probable effects of vibrations occurring during both normal operation and seismic disturbances. An assessment could then be made of the adequacy of present methods of design and analysis employed for the various components. This assessment would provide the information needed to develop recommendations for further work leading to more clearly defined margins of safety.

11.5.2 Status

Considerable work has been undertaken to determine seismic and vibration response of the PCRV structure, core, and components as discussed in the following sections of this task group. Present procedures use general purpose and special types of analyses in conjunction with model studies. There is a need to evaluate the capabilities of the existing methods of analysis as well as the type and scale of model studies being employed.

11.5.3 Identified work

An overall independent assessment (task 4.0.1.1) should be made of the status of HTGR seismic and vibration technology to determine areas in which additional safety margins, features, and/or study may be needed to provide the required degree of safety. Areas of particular interest are core and component seismic analysis and design procedures. A critical review should be made of ongoing seismic work. This work should be coordinated with the task area 6 assessment study of HTGR core support structures.

11.6 CORE SEISMIC RESPONSE (TASK GROUP 4.1)

The purpose of this task group is to improve the capability to predict the response of the HTGR core to a seismic disturbance.

11.6.1 Scope and justification

The proposed core studies will be conducted to develop capabilities for testing reduced-scale core mockups to determine the seismic response of full-size HTGR core structures. The approach will be to test core mockups of reduced size and to develop the capability of scaling the results to actual size. Core integrity during a seismic disturbance is vital to the reactor shutdown and core cooling capabilities.

11.6.2 Status

A seismic research and development program is under way at GAC to ensure the structural integrity of the HTGR core under safe-shutdown earthquake conditions. The objective is to verify the design adequacy of the core and the core supports for at least 0.5-g peak ground acceleration in accordance with the AEC regulatory requirements. Additionally, the program will study the effect on core response of such parameters as core size, support configuration, core gap, and variation in earthquake characteristics and the effects due to the operating environment.

The seismic program is carried out in two parallel phases — an analytical phase and an experimental phase. The experimental phase consists of tests on scale models of the core designed to satisfy the requirements for dynamic similarity. By verification of the model scaling laws, HTGR design loads are obtained directly. In the analytical phase, mathematical models of the core will be developed and correlated with the results from the tests. Once accurate correlation has been achieved, HTGR design loads are determined analytically. The full-scale analytical model input parameters are obtained from use of the verified scaling laws or from full-scale component tests (i.e., full-scale collision test).

This work is confined to the components in the PCRV core cavity: the fuel elements, reflector blocks, core support blocks, and core support structures. Thus, the core structures have been isolated from the PCRV structure, an assumption that is justified by the fact that the PCRV is infinitely stiffer than the core structures, in comparison, and the weight of the core is only about 2.5% of the PCRV weight.

For test and analysis, the time-histories of synthesized earthquakes and actual earthquakes, such as the El Centro and Taft events, are used. The earthquakes used to obtain core and core-support design loads satisfy the latest AEC design specifications. These earthquakes are the PCRVR response at the core-support level obtained from seismic analysis with a linear model of the containment building and the PCRVR. This model is coupled to an elastic half-space that represents the appropriate soil conditions for the plant.

The experimental portion of the program is divided into three groups of tests:

Group 1. Functional tests on fuel elements and other components of the core and its associated structures. (The appropriate tests will provide acceptance criteria for these components. Forcing functions for these tests are derived from the full array tests of group 3.)

Group 2. Dynamic response tests of limited assemblies of components. (These tests are part of the preparation work for the tests in group 3. Also, the results provide basic input to the analytical core models.)

Group 3. Tests on scaled models of all or portions of the core. (The models are subjected to harmonic vibrations and/or random excitations corresponding to real and artificial earthquakes. The results from corresponding analytical models are correlated with these test results. Also, the results from the full array models subjected to forcing functions that satisfy the AEC criteria are employed as core design loads.)

In group 1, the strength of the standard fuel element under both static and dynamic loads has been determined. The dynamic tests were high-velocity impact of fuel elements with a near rigid mass. Based on subsequent two-body impact tests between fuel elements and control rod elements employing the time-history response of impact from typical earthquakes as derived from the full-array tests of group 3, fatigue curves are being established showing the number of cycles to failure as a function of relative impact velocity and impact force.

The following tests in group 2 are completed except for test 4.

- (1) Two-body collision tests between fuel blocks of 1/5, 1/2, and 1/1 scale were performed to study the collision dynamics of graphite, that is, to obtain the force/time relationship

of impact, coefficient of restitution, etc. Multibody collision tests between blocks (five blocks on rails) generated information about fuel-block impact and separation behavior. Such information provided basic input to the analytical core models and aided in the verification of the assumed model scaling factors.

- (2) Vibration tests were conducted in which sinusoidal excitations were imposed on the ends of single fuel columns of 1/4 and 1/8 scale. A 1/8-scale graphite column, a 1/4-scale graphite column, and 1/4-scale acrylic column were tested. An axial static force simulating the pressure force was applied to the top of the column. The input frequency ranged from 0 to 18 Hz, and the input displacement from 1/16 to 1/4 in. The data obtained were used to study the "column effects" on a planar array as part of the effort to assess how the three-dimensional core can be described by a two-dimensional model.
- (3) Hydromechanical shaker tests on 91 1/5-scale fuel blocks were conducted in a geometry representing a two-dimensional planar section of the core. This test was considered a forerunner to the 1/5-scale full-array tests of the HTGR core and core-support structures and provided a check-out for the hydromechanical shaker control system, instrumentation, and computerized data-acquisition system. These tests concluded an extensive instrumentation development program designed to provide displacements, velocities, and forces on elements in the core and displacements and forces measured at the core support structure.
- (4) Tests are being performed on two modified two-dimensional planar section models of the core with 1/5-scale and 1/2-scale fuel blocks. The test objectives are: (a) obtain the effect of change in core gap on core resonance, fuel block impact response, and boundary response; (b) correlate results from two-dimensional analytical models with test results; and (c) obtain further data in aid of verifying the model scaling laws.

In group 3, two tests have been undertaken:

- (1) Tests of a 1/4-scale model of a partial core are planned in cooperation with the French Atomic Energy Commission (Commissariat à l'Energie Atomique) at their research facilities in Saclay near Paris. The model is built of graphite and consists of 10 levels of 91 fuel columns in full (scaled) height.

- (2) Tests of a 1/5-scale model of the full core built in the United States have been carried out. The model is built of graphite and is designed to be a faithfully scaled reproduction of the real HTGR core and core support structure, incorporating all design details that may affect the dynamic response. It represents the core in its cold and fully irradiated state. Briefly the test objectives were to determine the core motion characteristics (force and deflection of lateral supports) and the core deflection and forces (number of repetitious loadings of fuel blocks). Core motion characteristics include core lumping, natural frequencies and mode shapes, and system damping; core deflections and forces include fuel column and reflector column deflections, fuel block impact forces, and fuel block dowel pin forces.

The earthquakes have been applied at equivalent ground intensities ranging from 0.075 g to 0.75 g. The core was tested uniaxially in two orthogonal directions, across-the-flats of a fuel element and across-the-corners. Two different side-support designs were studied, a soft spring support (20,000 lb/in. equivalent full-scale stiffness) and a combination soft spring and hard stop support, which represents the HTGR core-support system.

To comply with the new AEC licensing requirements, further tests with the 1/5-scale model (modified partial or full array) will be undertaken with simultaneous multi-axis input. The multi-axis input will include one horizontal axis combined with one vertical axis excitation.

Computer code development and core modeling have been carried out in parallel with the test work. Although large finite-element structural codes are available, it has been recognized that the complex modeling of the nonlinear collision dynamics of stacked hexagonal graphite blocks is not readily achieved by the techniques available in such codes. Hence, emphasis was put on developing specialized codes for this purpose.

Two codes, COSAM¹ and CRUNCH,² have been developed for modeling the core structures. These codes use different principles in mathematical formulation. COSAM is based on the second-order differential equations treating a spring mass system, and CRUNCH employs impulse-momentum equations to describe the core-block collisions. So far these codes have been developed to represent a planar section of the core in one or two dimensions. Continuous development is in progress, however, to expand this capability to include the treatment of a vertical core section in two dimensions and possibly in full three dimensions.

Two single-array core models have been constructed for COSAM and CRUNCH. These represent the masses of two reflector blocks on each side of the core, each supported by one lateral spring pack. The array across the core consists of six fuel blocks in sections, 27 deep (one reflector block covers a sectional area equal to about six fuel blocks). In representing the core as an array, the cross coupling of core motion is assumed to be negligible. This in fact is a good assumption according to 1/5-scale test results. The coupled motion of the reflector block is accounted for by a spring between the blocks which represents the circumferential stiffness characteristics observed during 1/5-scale tests. Further, the effect of column stiffness on the individual fuel block motions are considered by designing an appropriate spring from the element to the PCRV boundary. All elements are viscously damped to simulate system energy losses due to friction.

Good correlations with the 1/5-scale full-array test results have been obtained with these simplified COSAM and CRUNCH core models. Meanwhile, two-dimensional models are being constructed to handle multi-axis excitation. These models will be correlated with existing test data and subsequently with test data from the multi-axis tests.

11.6.3 Identified work

The seismic input data for the tests of the 1/4-scale and 1/5-scale models will consist of both sinusoidal and actual and synthesized time-history excitation. The results will be correlated with analytical models such as COSAM and CRUNCH. Required data for the core components include collision forces and frequencies, shear forces on the dowels, deflections or disarrays that could affect alignment, and extent of element lumping. For the lateral restraint structure, needed measurements include loads at each support point, deflection of the flexible support members, and number of load repetitions.

Fuel-element impact tests are also currently being performed to establish the resilience of the HTGR design with the reference material. The work identified in this task group consists of four main tasks — three source tasks and a synthesis task.

Source task 4.1.1.1 covers adaption to an HTGR design of information obtained from other studies aimed at describing the responses of the core to a given earth movement.

Source task 4.1.1.2 consists of bi-axial shaking tests on a 1/5- and possibly a 1/2-scale model. The 1/2-scale test would not be performed in the same place as the 1/5-scale test, but instrumentation would be the same. This task also provides validation of model predictions.

Source task 4.1.1.3 covers further work on the impact tests on fuel elements with particular emphasis on the effects of temperature and irradiation. Impact tests should be performed on blocks of the large HTGR core design. These blocks have a smaller number of large holes than the previously tested Fort St. Vrain blocks. The number of collisions of various energies required to initiate graphite web failure would be determined for both standard fuel elements and control rod elements. Compression strength changes due to repeated impacts should be determined. In the design of the new experiments, techniques for simulating the actual stress conditions within the fuel elements at reactor temperatures after some irradiation should be explored.

Synthesis task 4.1.2.1 covers development of analytical models.

11.7 COMPONENT SEISMIC RESPONSE (TASK GROUP 4.2)

The purpose of this task group is to adapt and validate analytical methods for describing the seismic responses of vital components with emphasis on the nonlinear aspects of the system.

11.7.1 Scope and justification

Under severe earthquake conditions, the reactor and main stream generators must be shut down and core cooling maintained. This requires satisfactory operation of many components such as the rods, actuators, and auxiliary cooling-loop backflow prevention devices. Moreover, the integrity of other devices is necessary to prevent compounding of the plant contingency condition. Consequently valid analytical methods must be developed or adapted to describe the seismic responses of vital components, with emphasis on the nonlinear aspects of the system.

11.7.2 Status

A number of large, general-purpose computer programs are available for analyzing the dynamic response of complex systems with many degrees of freedom. Programs such as NASTRAN,³ SHAKE,⁴ DSAP,⁵

and others are widely used but have little or no capability for handling nonlinear systems. Experimental investigations have clearly demonstrated the existence of strong nonlinearities in steam generators and other systems.

11.7.3 Identified work

This task group should investigate initially the feasibility of and then formulate a general-purpose solution technique for the time-history dynamic response of some selected nonlinear systems that exist in a large HTGR. The solution technique should be capable of treating both inherent nonlinearities, such as material yielding or opening and closing of clearances within the system, and discrete, and possibly large, sources of damping as opposed to model damping assumptions. This requires the numerical solution of the nonlinear differential equations of motion on a large digital computer. Primary effort should be devoted to developing a method for handling the nonlinear stiffness matrix and to selecting or developing the best numerical integration algorithm that is mathematically stable for the widest possible range of practical problems without sacrificing either numerical accuracy or efficiency.

Concurrently the supporting program should be planned. Later phases of the task group should include development of analytical models for specific component classes and the extensive test program. The work has been divided into two source tasks and one synthesis task.

Source task 4.2.1.1 performs the initial investigations that establish the theoretical basis for the response models. In addition, this task develops plans for the large-scale experimental program that will be required to generate empirical data for the analytical models and to validate the model results.

Source task 4.2.1.2 includes the experimental program itself, involving excitation of components by shake table or other means and observation of the responses. This task fulfills the aims of a validation task since many tests will be done on real components.

Synthesis task 4.2.2.1 provides analytical models.

11.8 SOIL-STRUCTURE DYNAMIC INTERACTION (TASK GROUP 4.3)

The purposes of this task group are to complete the development of analysis methods for nonlinear soil properties and to provide reliable test data for verification purposes.

11.8.1 Scope and justification

Both finite-element and elastic half-space methods of analysis should be studied. Experimental studies should consist of testing models under a wide variety of simulated foundation conditions. Such a program would provide experimental verification of the analytical model over a wide range of parameters including effects of soil properties, acceleration levels, layered media, and embedment.

These efforts toward establishing advanced models with additional capability concerning various significant ground effects should provide basic data for all large reactors, including fast reactors, and would therefore constitute scientific work of a general nature that would benefit all reactor development and safety.

11.8.2 Status

Recent investigations generally fall into either finite-element methods or those methods treating the soil medium as an elastic half-space, and both approaches have advantages as well as limitations. The finite-element method permits an easier treatment of nonuniform soil properties but is costly in terms of computer time and is restricted to very simple representation of the structure. Extension of an elastic half-space to account for nonlinear soil properties is not as straightforward as for the finite-element method but allows essentially an unlimited number of degrees of freedom for the structure modeling in addition to economical computer-run times. This approach has recently been extended to account for the effects of through-soil coupling on the dynamic response of adjacent structures founded on separate base slabs. In this model, the ground medium is represented by a homogenous, isotropic, elastic half-space that is coupled with up to three flexible, damped, three-dimensional structures on separate base slabs. This effort resulted in the successful development of a computer code, SOSDIT,⁶ that is capable of treating both steady-state and transient response problems.

Some experimental studies have also been performed, but these tests have been confined to the forced vibrations of rigid footings on soil or very low-level forced vibrations of an actual nuclear plant

with particular site conditions. While these tests were informative, they could not provide design data concerning the structural response for a range of possible ground conditions and acceleration levels under earthquake excitation.

11.8.3 Identified work

The theoretical formulation for nonlinear soil properties should be completed and a general purpose program developed which would be capable of treating nonlinear effects on the seismic response of reactor structures. This program should then be extended to account for additional considerations such as layered soils.

An experimental program is suggested to verify the analytical portion of the investigation. Because testing of full-scale reactors is impractical for other than extremely low acceleration levels and, even if accomplished, would yield only a single data point corresponding to a given site condition, a scale-model test program is identified which would utilize foam beds to simulate the soil media. The work is divided into two source tasks and one synthesis task.

Source task 4.3.1.1 extends current investigations of nonlinear soil properties. Source task 4.3.1.2 covers the model tests described above. Synthesis task 4.3.2.1 covers the production of analytical models.

11.9 PCRVS SUPPORT (TASK GROUP 4.4)

The purpose of this task group is to investigate alternative PCRVS support system concepts.

11.9.1 Scope and justification

The proposed work involves studying PCRVS support-system concepts for various degrees of flexibility and different seismic intensities, with the emphasis on the high seismic-intensity range. Other research and development work on components may be needed, such as supplementary dampers required for very flexible support. Development of analytical techniques that may be required in the study will be included. The development of a damping system or flexible coupling would provide additional resistance to seismic loadings.

11.9.2 Status

The current PCRIV support structure, which consists of six radial walls, supports the PCRIV above the containment mat. The PCRIV is cast integral with the support structure and containment mat.

11.9.3 Identified work

A test program including both static and dynamic model tests should be carried out to confirm the analysis and design. This work has been divided into two source tasks and one synthesis task.

Source task 4.4.1.1 investigates alternative PCRIV support-system concepts, considering effects of various degrees of flexibility, damping, and slip.

Source task 4.4.1.2 covers any model tests needed to confirm analytical predictions or to investigate new PCRIV support arrangements.

Synthesis task 4.4.2.1 includes development of analytical models for parametric studies.

11.10 REACTOR INTERNAL VIBRATION (TASK GROUP 4.5)

This task group is concerned with defining responses of components subjected to vibration induced by gas flow.

11.10.1 Scope and justification

Both experimental and analytical studies are in this task group. The budget proposed initially is only sufficient for some theoretical analyses and a very modest experimental program. Further work, particularly experimental testing, may be found to be needed. This program need not be given high priority until component designs become relatively firm. AEC Regulatory Guide 1.20 covers this subject; however, analytical and test methods must be developed to satisfy these requirements.

11.10.2 Status

A model of a partial Fort St. Vrain core was subjected to gas flow in order to detect any vibrational modes. Core-support blocks and posts were included, and both were instrumented with accelerometers. Similar tests on steam generator modules and a complete steam generator mockup have been performed.

Reactor components, including a steam generator, were instrumented during the Fort St. Vrain pre-operational tests. Tests were also carried out on typical thermal-barrier panels in the laboratory, with results indicating adequacy of the design.

11.10.3 Identified work

Measurements of vibration responses and those parameters that define the input forcing functions should be made. Duration of the tests should assure that all critical components are subjected to at least 10^7 cycles of vibration. The measurements should be sufficient to verify that the cyclic stresses in the components, as determined by analyses, are within the acceptable design stress limits. Lateral, vertical, and torsional amplitudes of vibration should be detected. Results of this task should be sufficient to enable a comparison between predicted performance and component failure conditions. A monitoring program, using sufficient vibration-measuring instrumentation, should be formulated to detect the predominant vibratory responses during the service life of the components.

The work has been divided into three tasks:

Source task 4.5.1.1 aims to investigate the vibration driving (input forcing) forces and the component vibration response modes.

Source task 4.5.1.2 is concerned with improving techniques for measuring component vibrations during pre-operational reactor tests.

Synthesis task 4.5.2.1 covers the development of mathematical models for predicting the vibration modes of specific components.

REFERENCES FOR SECTION 11

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12. CONFINEMENT COMPONENTS (CONCRETE REACTOR VESSEL AND CONTAINMENT STRUCTURES) (TASK AREA 5)

12.1 TASK DEFINITION AND OBJECTIVES

The design of prestressed concrete reactor vessels (PCRVs) for current HTGR systems is based on the integral concept; i.e., the entire pressurized gas circuit composed of reactor core, primary coolant system, and portions of the secondary coolant system is contained within a single vessel. PCRVs have been built for use either with or without additional containment; however, the more recent commercial HTGR designs include a concrete reactor containment structure.

The PCRV is, in essence, a spaced steel structure since its strength is derived from a multitude of linear steel elements made up of deformed reinforcing bars and prestressing tendons. The PCRV is designed to fail progressively only in small steps; consequently, a high degree of safety is realized, with catastrophic failure becoming an impossibility. PCRVs used for HTGR systems are generally massive, thick-walled, right circular cylinders having flat heads. They are either of a conventional single-cavity type such as was used for the Fort St. Vrain Nuclear Generating Station or of the newer multiple or multi-cavity type such as is proposed for the larger Delmarva Power and Light Company Summit Station. Subsequent discussion in this chapter will be based primarily on the multi-cavity type of PCRV to be used with reactor containment since it is the typical system being offered in this country for commercial HTGRs. Most of the research proposed here is basic in nature and should be relevant to any possible type of PCRV-reactor containment scheme applicable to HTGR systems.

In a multi-cavity PCRV, the reactor is located within a large central cavity that is surrounded by smaller cylindrical cavities containing the primary cooling system and other cavities housing auxiliary cooling loops or pressure relief wells. The steam generators and helium circulators are located in the primary cooling system cavities. These cavities are connected at the top and bottom to the central core cavity by radial ducts and are sealed at the upper end by concrete plugs that support the helium circulators.

A continuous welded steel liner is attached to the walls of each cavity. This relatively flexible membrane serves to contain the primary coolant while the concrete vessel supports the liner and provides resistance to the coolant pressure loading. A thermal barrier and network of cooling tubes are attached to the liner to maintain a specified temperature (about 150°F) at the liner-concrete interface during normal full power operation.

There are numerous penetrations through the PCRV heads for nozzles and openings for control rods, fuel elements, process piping, etc. Because the primary coolant circuit is contained entirely within the vessel, relatively large penetrations are required for withdrawal of helium circulators and heat exchangers. The most numerous penetrations, however, are for the refueling standpipes. All penetrations are sealed integrally with the liner to prevent loss of primary coolant.

The PCRV is constructed of relatively high strength concrete and is reinforced by a combination of conventional steel reinforcing bars and a steel post-tensioning system consisting of vertical tendons and circumferential wire-strand windings. The PCRV is anchored to the support structure and foundation mat by a combination of reinforcing bars and vertical tendons.

The reactor containment structure completely encloses the PCRV. It can be constructed of either reinforced or prestressed concrete or some combination thereof. Current preference is for a vertical right circular cylinder having either a hemispherical dome in the case of reinforced concrete or a shallow dome in the case of prestressed concrete. The inner surface of the concrete is continuously lined with welded carbon steel plate for leak tightness. Air purification and cleanup systems are provided for collecting and processing any radioactive materials that might be released from the PCRV during an accident.

Penetrations ranging in diameter from a few inches to as much as 20 ft are provided in the cylindrical walls to permit access of personnel, as well as passages for fluid piping systems, electrical ducts, and equipment removal. These penetrations must be provided with gas-tight seals that will act integrally with the containment liner.

The PCRV and reactor containments are designed to withstand the loadings occurring during construction and during normal and abnormal operating conditions, as well as assumed severe environmental conditions. These conditions are defined in the proposed *Standard Code for Reactor Vessels and Containments* (Proposed Section III, Division 2, Boiler and Pressure Vessel Code, American Society of Mechanical Engineers, 1973). The proposed code has been published and is being used in practice on a "trial and comment" basis before final adoption. Addendums will be published semiannually once the final version is adopted.

The construction of PCRVs and reactor containment structures presents no basically new concepts although it does call for far greater refinement and more detailed investigation and thought in both design and construction. Since the PCRVs greatest asset is the unparalleled

standard of safety achieved by containing the entire gas pressure circuit within a vessel that can fail only slowly and with ample warning, the factors contributing to this type of characteristic behavior must be the primary design objective. The development of highly reliable analytical procedures capable of predicting both short- and long-term structural behavior must be extended to permit a vessel to be designed with accurately known safety margins. Components vital to the structural integrity are prestressing tendons, wrapping wire strand, and thermal barrier.

The second basic design objective which is also applicable to the reactor containment structure is to achieve a high degree of leak tightness or pressure integrity; components vital to the pressure integrity are the penetrations and liners. The reactor containment structure must also be designed to withstand the combinations of forces considered credible that could result in a loss of coolant from the primary system.

12.2 SCOPE

Some of the work outlined under this task area is related to other task areas of this proposal. Task area 4, consisting primarily of seismic studies (see Sect. 11), involves investigations of PCRV support structures and soil-structure interaction. These are important aspects of both PCRV and reactor containment structural design. Moreover, the shear strength study proposed under the present task area relates directly to determining the resistance of both of these structures to seismic loadings.

The safety instrumentation studies described in Sect. 14 will provide information applicable to the proposed specialized PCRV instrumentation evaluation and development work, especially with respect to thermometry. In addition, the findings of the PCRV task area will provide information for the accident postulation and analysis studies discussed in Sect. 7.

The concept of prestressed concrete nuclear reactor pressure vessels was proposed almost two decades ago. Since that time, nine power plants incorporating concrete reactor vessels have gone into operation while an additional 15 PCRVs are known to be in various design or construction stages in Europe and the United States.

Considerable research has been done on plain, reinforced, and prestressed concrete throughout the world; however, a limited range of variables was employed in most of this work, and it is difficult to relate the numerous independent, and many times unrelated, research efforts to each other. For example, much of the available

concrete-property data were based on very limited laboratory tests undertaken on "model" concretes. Small size aggregates were frequently used, testing environments were not representative of the state of the material in the PCRV, and the tests were frequently conducted at such early concrete ages that their application to 30-year vessel designs is of marginal value. The development of good data is an especially difficult task since concrete is a non-isotropic, heterogeneous, viscoelastic-plastic material whose properties are dependent on many variables. Consequently, the mathematical formulation for the behavior of complex prestressed concrete structures presents great difficulties. Finally, the interaction of liner and thermal-barrier systems with the complex structure must be included to achieve total understanding of PCRV behavior.

Present designs require the temperature of the concrete to be restricted to approximately 150°F by means of an elaborate insulation and liner cooling system. This restriction is imposed primarily because of a lack of understanding of the long-term behavior of biaxially and triaxially stressed mass concrete at elevated temperatures.

PCRV designs in the United States are required to use unbonded prestressing tendons because of a lack of understanding of concrete creep and tendon relaxation, even though a PCRV with properly bonded tendons would exhibit a higher ductility and a more effective utilization of the strength capacity of both the tendons and liner.

The ultimate load analysis of thick head regions is difficult to perform and based largely on empirical test data. Consequently large safety factors must be used in the design of PCRV head regions. This work has been hampered by the lack of reliable information on the load-deformation properties of concrete under complex states of stress and, moreover, by the lack of a general failure criterion.

Since a number of HTGR systems employing multi-cavity PCRVs are in the design or review stage, there is a pressing need to evaluate and, if necessary, develop accurate, durable, and highly reliable instrumentation for the continuous monitoring of operating PCRVs. The data obtained would provide a much needed avenue for correlating PCRV performance with the analytical predictions obtained using laboratory test data. This information could lead to significant and systematic reductions in present performance limitations and remove a large area of uncertainty in design, thereby leading to safer and more reliable HTGRs.

From the safety standpoint, the most important design requirement for HTGR containment structures is the virtual elimination of any probability that an uncontrolled release of primary coolant could occur as a consequence of structural failure. It is therefore important to know the amount of reserve structural strength provided by the design. Factors influencing the accurate determination of reserve strength are material properties, applied loadings, and uncertainties inherent in the design. A series of hypothetical accidents will be systematically evaluated under another task area (see Sect. 7) to determine possible consequences of various vessel component failures. Data will be developed under the present task area to provide input to the relevant accident evaluation studies.

Considerable PCRV research and development work has been done in the past in France, England, and the United States and more recently in Sweden, Germany and Japan. Main emphasis of the Swedish and German research has been toward light-water and high-temperature reactor applications. These studies consist for the most part of basic materials-properties investigations and structural model tests.

In the United States, a PCRV research and development program was established by the USAEC in 1966. The program consisted of technology assessment studies, analytical methods development, materials investigations, and experimental models studies. Critical reviews of the available literature on primary² and secondary³ containments were prepared under this program.

A technology assessment study planned as the initial step in the proposed program will concentrate on information developed since publication of the two HTGR base program information reviews.^{2,3} The scope of the individual research tasks is contingent upon the findings of the technology assessment study which will evaluate the status of worldwide concrete materials research and model studies. Specific needs for basic materials research and model studies are to be identified, and the follow-on test programs planned.

12.3 STATUS AND NEEDS

PCRV technology has experienced 19 years of development during which considerable practical experience, advancement, and success has been achieved. This developing technology demands further attention because of the rigorous safety requirements of nuclear power plants and because of the continuing need for developing a yet-to-be-achieved thorough understanding of PCRV short- and long-term behavior.

According to Section III, Division 2 (Proposed) of the ASME Code, the integrity of a PCRV must be validated for all loading conditions that can be postulated to occur during the vessel service life. A total of six separate loading categories are specified: construction, normal, abnormal, severe, extreme, and failure load. Included in these categories are the routine types of loadings such as dead-weight and wind and such infrequent but severe loadings as those resulting from tornadoes, tsunamis, and earthquakes. Loadings considered under the failure-load category are pressurized cracks in the vessel wall (liner failure), temperature increases to 400°F in the mass concrete of the unpressurized vessel (thermal-barrier failure), temperature increase in the mass concrete to 600°F for a pressurized vessel (thermal-barrier failure at pressure), penetration closure failure, steam impingement on prestressing anchorage hardware, and failure of up to 50% of the prestressing tendons. To design adequately for this impressive listing of loading conditions, PCRV behavior must be clearly understood, and highly reliable analytical methods and material properties data must be developed and verified.

The construction of containment structures follows practices comparable to those used for any important large concrete structure. Because the liner is thin and flexible before concrete is cast, dimensional stability must be maintained by internal shoring. Probably the most difficult aspect of containment construction is concrete placement in regions congested by high concentrations of steel reinforcement. This difficulty has been eased considerably by the development of externally applied circumferential wire wrapping reinforcement. The method of applying and restraining the wire windings, as well as possible improved circumferential prestressing systems, deserves particular attention.

The requirement of maintaining adequate corrosion protection of prestressing before, during, and after construction is another area deserving particular attention. The methods used to test the various structural components, materials, and the completed vessel will be further assessed. Areas of current interest are shear strength tests of prestressed and reinforced concrete, tests to provide load-deformation properties of concrete under complex stress states, corrosion tests of prestressing, liner structural tests, containment convection studies, and thermal model tests. Also the type of proof testing that best suits the application needs further study. Development of a rational basis for proof testing would eliminate some of the questions concerning safety evaluation of containment structures.

Since a number of large PCRVs will soon be under construction, careful consideration must be given to an assessment of present in-service and post-operational inspection methods. A study is needed to review the present instrumentation for monitoring prestress forces, temperature gradients through the PCRV walls and heads, overall deformations and deflections, strain profiles, and corrosion of prestressing materials. There is also a pressing need for instrumentation capable of confirming leak-tightness of the entire structure during its lifetime.

Considerable PCRV research and development has been done in the U.S. by the principal HTGR vendor and under the separate AEC-sponsored program described in ref. 4. The major portion of the work outlined in this summary report has been completed and is discussed in the following section together with the relevant work undertaken at GAC and the current status of each task group.

12.4 TASK AREA EFFORT AND SCHEDULE

The task area is divided into a technology assessment study and ten tasks dealing with various aspects of structural and materials behavior, design, analysis, model studies, and specialized PCRV instrumentation. The tasks are organized where possible according to the various PCRV components consisting of concrete, prestressing, liner, and thermal barrier systems. Figures 12.1 and 12.2 show the basic organization of the task area as well as the interrelationship of the tasks. The PCRV surveillance study is an essential element in any comprehensive PCRV research program; however, it cannot be defined at this time. It is therefore included in task group 5.3 for completeness only. Table 12.1 is a task-by-task effort summary of task area 5 for the projected 10-year period.

The following gives a general description of the proposed studies. The initial effort is the preparation of a technology assessment report that will provide planning information to many of the proposed task groups. It comprises a literature search and consultations with concrete authorities and PCRV designers to determine the status of development in mechanical properties of concrete in the context of PCRV design. Supplemental investigations would then be planned to meet projected needs. The considerations of particular interest are: (1) moisture state and moisture migration; (2) strength-failure criteria; (3) long-term creep characteristics and determining factors; (4) properties variations with service conditions up through the accident range, e.g., elastic moduli, thermal conductivity, thermal expansion coefficient, and permeability; (5) radiation damage effects; and (6) advanced concretes for high

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Fig. 12.1. Effort for PCR technology (task area 5). Part 1.

Fig. 12.2. Effort for PCR technology (task area 5). Part 2.

Table 12.1. Summary of identified safety and safety-related work in task area 5

Task groups or tasks marked with an asterisk (*) are included in the safety planning guide (Part II of this document).

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task area 5 Confinement components										
Task group 5.0 Need assessment for confinement components*										
5.0.1.1 Initial technology assessment	1.0									
5.0.1.2 Continuing need assessment study		1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5
Task group 5.0 total	1.0	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5
Task group 5.1 Failure analysis										
5.1.2.1 Analytical PCR V failure studies*	0.5	0.5	0.5	0.5	1.0	1.0	1.0	1.0	1.0	0.5
5.1.2.2 Failure analysis methods development	0.5	0.5	0.5	0.5						
5.1.2.3 Three-dimensional finite element analysis improvement and evaluation	1.6	1.6	1.6	1.0	1.0	0.5				
Task group 5.1 total	2.6	2.6	2.6	2.0	2.0	1.5	1.0	1.0	1.0	0.5
Task group 5.2 PCR V head failure studies										
5.2.2.1 Concrete failure criterion development		1.0	1.0	1.0						
5.2.2.2 Simplified design methods development							0.4	0.4	0.6	
5.2.3.1 PCR V model failure tests		0.2	2.5	2.5	2.3					
Task group 5.2 total		1.2	3.5	3.5	2.3		0.4	0.4	0.6	
Task group 5.3 Concrete properties										
5.3.1.1 High-temperature strength		1.0	2.0	3.0	3.0	3.0	2.0	1.0		
5.3.1.2 Long-term creep and creep recovery		1.0	2.0	3.0	3.0	3.0	3.0	2.0	1.0	0.5
5.3.1.3 Reinforced concrete shear strength			1.0	2.0	2.0	2.0	0.5			
5.3.2.1 New concrete materials development		0.5	1.0	2.0	2.0	1.0	0.5	0.5	0.5	0.5
Task group 5.3 total		2.5	6.0	10.0	10.0	9.0	6.0	3.5	1.5	1.0
Task group 5.4 Prestressing components and materials										
5.4.1.1 Tendon corrosion and failure mechanisms		1.0	2.0	2.0	2.0	2.0	0.5	0.5	0.5	
5.4.1.2 Strip-wound prestressing systems		2.0	2.0							
5.4.2.1 Tendon installation criteria development					1.0			1.0	1.0	
Task group 5.4 total		3.0	4.0	2.0	3.0	2.0	0.5	1.5	1.5	

Table 12.1 (continued)

Task description	Relative effort by fiscal year (man years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 5.5 Liners and penetrations										
5.5.1.1 Liner and penetration materials and fabrication techniques		2.0	2.0	1.0						
5.5.1.2 Liner and penetration failure modes and criteria		1.0	3.0	3.0	2.0	1.0				
5.5.2.1 Liner and penetration design analysis methods		2.0	2.0	1.6	1.6	1.6	0.8	0.8		
Task group 5.5 total		5.0	7.0	5.6	3.6	2.6	0.8	0.8		
Task group 5.6 Thermal barrier										
5.6.1.1 Thermal barrier testing	1.5	5.0	4.0	1.0						
5.6.2.1 Thermal barrier materials development		2.5	4.0	1.5	0.5	0.5				
Task group 5.6 total	1.5	7.5	8.0	2.5	0.5	0.5				
Task group 5.7 Instrumentation										
5.7.2.1 Concrete instrumentation development		1.0	2.0	2.0	2.0	2.0	1.0	0.5	0.5	
5.7.2.2 PCRV component instrumentation development		1.0	3.0	3.0	3.0	2.0	1.0	1.0	0.5	
Task group 5.7 total		2.0	5.0	5.0	5.0	4.0	2.0	1.5	1.0	
Task group 5.8 Model studies and PCRV surveillance										
5.8.1.1 Simple concrete structure tests		0.5	4.0	2.0	2.0	0.5				
5.8.1.2 Complex concrete structure tests			0.5	6.0	4.0	2.0	0.5			
5.8.2.1 PCRV scale-model tests				1.0	6.0	6.0	6.0	2.0	1.0	
5.8.3.1 PCRV surveillance										
Task group 5.8 total		0.5	4.5	9.0	12.0	8.5	6.5	2.0	1.0	
Task group 5.9 Containment convection studies										
5.9.2.1 Containment forced-convection study	1.0	3.0	3.0	1.0						
5.9.2.2 PCRV natural convection study		1.0	1.0	2.0	2.0					
Task group 5.9 total	1.0	4.0	4.0	3.0	2.0					
Task group 5.10 Code rule development										
5.10.1.1 Data collection and interpretation				0.5	0.5	1.0	1.0	1.0	1.0	1.0
5.10.2.1 Data evaluation and code rule development					0.5		0.5	0.5	0.5	0.5
Task group 5.10 total				0.5	1.0	1.0	1.5	1.5	1.5	1.5
Task area 5 total	6.1	29.3	45.6	44.1	42.4	29.6	19.2	12.7	8.6	3.5

strength and/or high temperature. Although the technology assessment study will deal primarily with the aforementioned concrete aspects of PCR design, it will provide information on recent PCR system component developments as well.

12.5 NEED ASSESSMENT (TASK GROUP 5.0)

This task group will provide a continuing independent evaluation of the safety of PCRs and reactor containment structures based on existing and new technological developments as well as data obtained from operating HTGRs.

12.5.1 Scope and justification

This task group begins with a critical review and safety assessment of existing technology in order to identify possible areas needing further research and/or development. This initial study will be followed by a continuing review and assessment of HTGR concrete reactor containment technology in light of new information and developments. The primary purpose of this task is to obtain the information needed to determine the actual margins of safety of these complex concrete containment structures.

12.5.2 Status

A series of three survey reports consisting of a bibliography and critical review² of PCR literature and a review of reactor containment design and construction practices³ was prepared under the ORNL Concrete Program. Since the most recent of these documents was published more than four years ago, information on more recent innovations such as multi-cavity PCRs and reactor containments for HTGRs were not discussed.

12.5.3 Identified work (Tasks 5.0.1.1 and 5.0.1.2)

Information published in the literature and obtained directly from manufacturers, designers, vendors, and constructors should be collected and subjected to critical review, and an evaluation made of the safety aspects of concrete containment structures. The survey should concentrate primarily on information developed since completion of the earlier critical reviews (ref. 2 and 3). The evaluation of this information should both identify those areas of the technology having possible safety uncertainties and outline

the investigations required to eliminate these uncertainties. The design, construction, and operational aspects of the concrete, prestressing and conventional reinforcement, liners and penetrations, and insulation and cooling systems as well as the instrumentation required to measure stress, strain, time, and temperature behavior should be evaluated.

Task 5.0.1.2 should follow the initial task to provide an independent review of efforts to eliminate any identified deficiencies and of the influence of new technological developments on reactor safety. This task is intended to be a continuing evaluation effort to identify problem areas and specify the additional work required to assure the safety of HTGR concrete containment structures.

12.6 FAILURE ANALYSIS (TASK GROUP 5.1)

The goal of this task group is to extend the capabilities for predicting the mode of failure of PCRVs and for accurately predicting stress distributions and dimensional changes of a PCRv over its service life.

12.6.1 Scope and justification

PCRVs are currently designed using elastic analyses, creep analyses, and semiempirical failure or ultimate load analyses. The first two analyses are used to assure continued serviceability of the PCRv for its intended life. The elastic analyses are used to predict stresses in the unpressurized vessel at the beginning of design life and in the pressurized vessel at the end of its design life, while the long-term behavior of PCRVs is predicted using creep analysis. Of special interest is the prediction of prestress losses and the additional strains that concrete creep imposes on the steel liners. The overall stress distributions however will not be greatly affected by creep.

The ideal failure analysis is one capable of accurately predicting the initiation and propagation of internal cracks within the head region as well as the side walls of a concrete vessel.

This task group is closely coupled to the head-failure studies of task group 5.2 as well as the model studies and surveillance work in task group 5.8 which provide the data against which the analytical methods are compared. There is also substantial dependence on the development of physical properties of concrete and interaction of various effects (task groups 5.2 and 5.3). Once the analysis

techniques if an adequately verified, they could be used to conduct analytical studies of representative concrete containment structures and to identify potential problem areas. This would provide information to be used in code rule development (task group 5.9).

12.6.2 Status

There are a number of sophisticated methods for analyzing both short-term elastic and long-term time-dependent loadings of concrete containment structures. Computer program forms of finite-element methods have been developed which can account for three-dimensional, elastic, short-term deformation and axisymmetric creep and cracking effects. Short-term behavior can be analyzed with good accuracy by considering primarily elastic deformations and, where elastic limits are exceeded, some degree of plastic deformation. The three-dimensional methods of stress analysis cannot yet fully account for long-term time-dependent behavior, particularly of complex geometries. The necessary constitutive equations could be developed to describe these long-term effects, provided the basic materials behavior can be characterized. All of the resulting analyses must be thoroughly verified by appropriate model tests. Present strategy is to overdesign complex portions of the structure so as to ensure that any failure occurs in the more benign barrel region, thereby precluding catastrophic failure in other less well-understood regions of the vessel. The development of a satisfactory failure analysis has been hampered by the unavailability of reliable information on load-deformation properties under complex states of stress (see task group 5.2) and the lack of a general failure criterion. This particular weakness in the analysis needs correcting before the goal of developing reliable numerical models of PCRV head regions that can accommodate material behavior realistically can be realized.

12.6.3 Identified work (Tasks 5.1.2.1, 5.1.2.2, and 5.1.2.3)

The currently accepted and eventually the improved analysis methods should be used to conduct studies of the modes of failure of current and planned vessel configurations. The most satisfactory methods of analysis should be used to conduct evaluation studies of the PCRV liner, concrete prestressing, and thermal barrier for the purpose of identifying possible areas of distress.

The existing failure-analysis method should be modified and improved to provide the capability of predicting the initiation and propagation of internal inclined cracks in the vessel head region of both

single- and multi-cavity vessel configurations having multiple penetrations. The more generalized analytical methods development work would involve primarily expansion of existing finite element methods to include three-dimensional elements and creep-crack analysis capabilities and the development of special techniques to analyze complex geometries. Eventually consideration should be given to simplifying the analytical procedures to provide more practical and useful analytical tools.

12.7 PCRV HEAD FAILURE STUDIES (TASK GROUP 5.2)

The goal of this task group is to provide experimental data needed for the development and validation of an analysis method for predicting vessel head failures. These studies will form the basis for a better understanding of possible modes of failure of single- and multi-cavity PCRVs and their factors of safety.

12.7.1 Scope and justification

These studies would develop reliable information on load-deformation properties of concrete under complex stress states which is needed to formulate a general failure criterion. The resulting criterion could be used in the development of reliable numerical models of PCRV head regions (task group 5.1). This information is required because calculations of propagation of internal inclined cracks in the vessel head are sensitive to the assumed failure criterion. Tests should be conducted on models of multi-cavity PCRV head regions both with and without penetrations to provide information needed to understand the behavior of the multi-cavity vessels.

12.7.2 Status

An investigation was conducted at the University of Illinois on the strength and behavior of PCRVs subjected to monotonically increasing pressure. The critical component from the viewpoint of improving safety and economy for the particular single-cavity type of vessel studied was found to be the end slab, as shown by results of PCRV model tests conducted at ORNL under the AEC program and at other laboratories. Emphasis of the work was on shear strength of end slabs with and without penetration. The experimental results indicated that the reduction in strength attributable to the presence of openings in the end slab was considerably less than predicted using developed theories. The observed insensitivity of shear strength to the presence of large penetrations was reconciled using

available information on internal stress distributions and a failure criterion for concrete subjected to complex stress states.

A multi-cavity 1:20 scale model PCRV was constructed at GAC and tested to investigate ultimate load behavior (ultimate load in this context equals safety factor times design load). This test provided data showing that specified safety margins relative to design pressure were satisfied. The ultimate failure could, however, not be observed due to equipment limitations.

Previous studies of PCRV failures have been hampered by the unavailability of reliable information on load-deformation properties of concrete under complex states of stress and the lack of a general failure criterion.

Although a number of different theories of failure for complex stress states have been developed, none of these is of the specific type required for the development of a satisfactory failure analysis.

12.7.3 Identified work (Tasks 5.2.2.1, 5.2.2.2, and 5.2.3.1)

A series of small specimen tests should be conducted under task 5.2.2.1 to develop data on load-deformation behavior of concrete. The tests would consist of loading small, instrumented, plain concrete specimens to failure under various complex stress states. The resulting data could be used to formulate a general failure criterion that is required for further development of a general failure analysis (task 5.1.2.1).

The earlier experimental PCRV head failure studies were done on a single-cavity cylindrical vessel having flat heads; these studies should be extended to include multi-cavity vessel configurations. The tests consist of stressing PCRV scale models to rupture after appropriate concrete curing to observe failure modes and confirm predicted sequences and effects. Scale factors in some models should be altered to promote failures in regions of interest, e.g., the head and penetrations. The model tests would also investigate the effects of barrel stiffness on vessel head strength.

As test data become available, they should be used in the development of simplified analysis procedures to provide more practical analytical tools.

12.8 CONCRETE PROPERTIES (TASK GROUP 5.3)

This task group is concerned with the development of a better understanding of the basic properties of concretes that may be used as construction materials for PCRVs and reactor containment envelopes.

12.8.1 Scope and justification

The successful use of concrete as a PCRv construction material is based on its known capacity both to reliably resist compressive loadings and to redistribute stresses from areas of high concentrations to those of lower concentrations through its characteristic viscoelastic-plastic behavior. Because of these characteristics, the present generation of PCRVs has performed satisfactorily despite the existing lack of complete understanding of basic concrete properties. However, uncertainties can be introduced into analyses as a result of transient changes in concrete properties - changes occurring over long periods of time or where strength limitations are exceeded locally by amounts sufficient to cause irreversible damage, e.g., cracking in regions of high stress concentration.

The time-dependent changes in concrete properties are complex and depend upon the following individual factors as well as on various combinations thereof: basic mixture constituents, amount and state of water (whether it is free, bound, or chemically combined), temperature, microcracking (cumulative damage occurring primarily in the cement matrix in the form of small cracks), shrinkage, radiation, and state of stress. The interrelationships of these factors are often extremely complex, and equally complex analytical models will be required to develop rigorous descriptions of any resulting structural behavior. The PCRv responses to postulated accidents depend not only on the vessel condition preceding the accident but on the influence of the concrete factors. Once the interrelationships are understood, development of simplified descriptions should be feasible so that a practical but reasonably accurate analysis can be obtained. A more basic understanding of these material properties and the resulting improved analysis methods would provide an increased level of confidence in PCRv design and promote construction economies.

Much of the input for detailed planning of the work of this task group will depend upon information obtained from the technology assessment study. Division of this task group into the basic tasks needing further study is shown in Fig. 12.1. There exists a need for further understanding of the strength and creep behavior (also creep recovery) of plain concrete at elevated temperatures as influenced by radiation damage, free moisture, and multiaxial

stresses. The scope of these tests will depend upon availability of reliable data from other sources.

The safety of the present HTGR as well as that of higher-rated future HTGRs would benefit from concrete having the capability to withstand temperatures higher than the present limiting values. Aluminate cements offer promise from the standpoint of both resistance to elevated temperatures and dimensional stability. Chopped-wire fiber concretes appear promising from the standpoint of their higher thermal conductivities, significantly improved tensile strengths, and lower moduli of elasticity. A combination of these two materials might be desirable. The scope of this task group includes investigations of the properties of these and other special concretes that may be applicable to HTGRs.

12.8.2 Status

Investigations of plain concretes to describe basic material properties, moisture migration in concrete, and long term creep behavior under constant multiaxial loadings and temperatures up to 150°F have been completed and are in the final reporting stage. The tests attempt to define both isolated effects and effect combinations over ranges of conditions encountered in PCRV service; however, little if any significant progress was made in the study of properties of mass concrete (specimens sealed to retain moisture as is the case in large PCRV structures) at temperatures significantly above 150°F. Consequently there is substantial room for accumulation of more data. Some research is also being done to develop the special concretes mentioned previously, as well as to define their properties. However, additional work is needed for evaluation of their applicability to PCRVs.

12.8.3 Identified work (tasks 5.3.1.1, 5.3.1.2, 5.3.1.3, and 5.3.2.1)

Tests should be performed to provide additional data on the strength and creep behavior (including creep recovery) of plain concretes under conditions of interest for HTGR applications.

The strength tests would involve subjecting concrete test specimens to sustained temperatures and uniaxial or biaxial loadings for extended times at the end of which the specimen will be tested to failure. Test temperatures should range from room temperatures to 600°F, with loadings in the range of PCRV design limits. Other specimens should be subjected to thermal and stress cycling before testing to failure.

Long-term creep and creep recovery effects should be measured for small plain-concrete specimens under various multiaxial and thermal conditions. The number and range of variables should be selected to provide sufficient information for the development of constitutive equations. The stress range should be extended to include creep rupture. Because both the ultimate strength and creep properties are significantly affected by moisture level, development of specimen sealing techniques will be necessary to permit careful control of concrete moisture; this is an especially difficult task for temperatures above 212°F.

The experimental shear strength study would provide information needed to determine resistance of reinforced or prestressed concrete containment structures to seismic loadings. Loading systems should be developed to apply biaxial in-plane tensile loadings to reinforced concrete plates to simulate pressure loadings combined with in-plane or transverse shear forces to simulate seismic loadings. Satisfactory tests of this type would provide much needed design information. Shear strength of reactor containment structures would also be influenced by concrete moisture, radiation, and the multiaxial stress state, but to a lesser degree than in the case of the PCRV.

12.9 PRESTRESSING COMPONENTS (TASK GROUP 5.4)

The basic characteristics of the devices used for prestressing concrete structures are investigated under this task group.

12.9.1 Scope and justification

The vertical tendons and circumferential wire windings are essential elements of the PCRV structural system because they provide the means for maintaining compressive stresses in the concrete. Each tendon consists of a bundle of steel strands that are installed within a guide tube and anchored at the ends after tensioning. The circumferential windings are installed under tension in grooves around the barrel of the vessel. The prestressing is subject to possible stress corrosion; hence, protective measures are taken to prevent any contact with moisture that might promote such attack. The scope of this task group includes all aspects of the behavior of prestressing components that may have safety significance in HTGR applications.

12.9.2 Status

In general, the mechanisms of tendon corrosion are understood, and adequate protective measures appear to have been developed. Tendon and circumferential wire prestressing methods have been extensively investigated by the principal HTGR manufacturer in the U.S. However, tendon corrosion failures have occurred in European reactors under conditions different from those proposed for the HTGR which has precipitated further basic studies of stress corrosion. The maintenance of tendon quality during construction is a difficult problem. An attractive alternative to the circumferential wire strand is strip winding. Although preliminary studies have been made of strip tensioning materials, further information is needed.

12.9.3 Identified work (tasks 5.4.1.1, 5.4.1.2, and 5.4.2.1)

Investigations of potential tendon corrosion under specific conditions that could arise during PCRV construction are suggested.

Recently experienced failures of some axial prestressing tendons during the ORNL thermal cylinder model test provide evidence that the corrosion mechanism is not completely understood. The reasons for these particular tendon failure have yet to be determined.

A study should be made of the corrosion mechanism for the purpose of developing practical tendon installation and environmental control criteria. The studies should consider effects of moisture, radiation, and temperature on mechanical properties and the effectiveness of protective coatings and grouts in preventing corrosion. Possible problems associated with substituting strip winding for the present wire strand circumferential prestressing should also be studied.

12.10 LINERS AND PENETRATIONS (TASK GROUP 5.5)

The purpose of the liner system in a PCRV is to seal the inside concrete surface against helium leakage and to prevent excessive heating of the concrete to maintain conditions suitable for long-term service. Failure to perform these functions might lead to concrete degradation. Hence, the work in this task group is directed toward a better understanding of the safety-related characteristics of this system.

12.10.1 Scope and justification

In Fig. 4.6, the liner system is seen to consist of the main liner membrane, anchors, and cooling coils which follow channels on the inside surface of the concrete vessel and are welded to the liner plate. The system also includes penetrations through the PCRV walls and their closures.

The liner membrane and its anchors present the main uncertainties in the system design. Any long-term creep contraction of the PCRV dimensions following liner installation or any tendency for the liner to expand relative to the PCRV due to in-service heating induces biaxial compressive stress in the liner and shear stress in the anchors. If the stress is excessive and/or anchors fail, the consequence could be liner buckling. The main problems, therefore, resolve largely into materials considerations such as strength and ductility of mild steel, particularly in weldments, under long-term temperature, radiation, stress, and load cycling. Additional questions concern capability to withstand transient stresses and possible failure modes, such as under accident conditions or at the place where a local insulation failure induces a hot spot.

PCRV penetrations vary in diameter from a few inches to several feet. They also vary widely in complexity. Failure of a closure involves a breach of primary containment, loss of coolant pressure, and, depending upon the size of the breach, the imposition of varying mechanical loads on primary circuit components. Fission-product release to the reactor containment may also depend on the size of the resulting orifice.

Portions of the PCRV penetrations which are unbacked by concrete constitute steel pressure vessels and, as such, are designed according to the rules of Section III, Division I of the ASME Boiler and Pressure Vessel Code to ensure against failure in the ductile, fatigue, and fast fracture modes. However, many of the rules of this section of the Code are specifically oriented toward the problems of the relatively heavy section steel components of water-cooled reactor systems. Thus, these rules are not directly applicable to the penetrations in the HTGR PCRV which generally involve relatively thinner section materials. In such cases, current practice in design is to comply with the intent of the Code until such time as a sufficient data base is established to provide a comparable set of rules directly applicable to the HTGR situation.

12.10.2 Status

At GAC, a test program is in progress to demonstrate the ability of the liner and anchor design to accept the anticipated biaxial compression loading conditions without developing deformations that could adversely affect the performance of the liner, anchors, thermal barrier, or cooling tubes. A second test program has been performed to determine the shear-load carrying capability of the cooling-water tubes and anchor studs attached to the PCRV cavity and penetration liners and, in addition, to demonstrate that the cooling tubes and anchor studs can accept the static and cyclic shear deflections imposed on them by relative movement between the liners and adjacent concrete as the PCRV deforms due to loading or creep.

Radiation damage tests were performed by the Naval Research Laboratory at integrated neutron flux levels up to twice the level expected in the lifetime of the Fort St. Vrain reactor. Effects on nil ductility transition temperature and on the full shear energy were found to be generally tolerable. Additional design verification testing was performed on the cooling-water tubes and on the thermal insulation.

Factors to be considered in penetration safety analyses are:

(1) the choices of materials for liner, penetration anchors, closure and plug; (2) effects of PCRV head movement on penetrations; (3) the need for double closures; and (4) the practicality of reliable flow restrictors to control depressurization rate.

Penetration designs involving redundant double closures, single steel closures, and closures partially constructed of concrete have been proposed. The design of penetrations for steam lines is intimately associated with the design of the steam generator tubesheet and assessments of the likelihood of tubesheet failure. No single design basis has emerged to date.

12.10.3 Identified work (tasks 5.5.1.1, 5.5.1.2, and 5.5.2.1)

Further testing is suggested to provide a better understanding of modes of liner failure, failure conditions, and safety margins. The resulting information would lead to the development of improved analysis methods. Investigations should be made of reliability and failure-mode evaluation of the cooling coils, thermal stress limits, and safety margin for the liner material; corrosion failure limits and safety margins; effects of radiation on liner embrittlement at neutron levels typical of large HTGRs; and liner stress limits and failure modes due to dimensional changes in the PCRV under

accident conditions. Close coordination with task groups 5.1, 5.2, and 5.3 concerning PCRV behavior is required for the latter.

Widely different assumptions are currently being made by different organizations about the likelihood of a major penetration failure. Assessment of safety issues associated with penetration failure therefore requires that a more unified approach to the analysis of penetration function and design be developed. A further test program should be set up to study the reliability and the failure modes of the PCRV penetrations, with particular emphasis on identification of applicable procedures to guard against fast fracture. This study should be backed up by test work where necessary. The program should initially concentrate on major design parameters, rather than on specific design configurations, and should involve extensive fracture toughness characterization of relevant materials. Typical parameters to be considered are diameter-to-thickness ratio, anchorage requirements, and typical features such as attachment to liners and closure requirements. Consideration should also be given to requirements for flow restrictors, size limitations relative to blowdown rate, manufacturing limitations, and effect on other components such as the containment.

Task 5.5.1.1 is concerned with materials problems associated with the liner and penetration, including definition of limiting loads and failure criteria. The work should particularly emphasize fracture toughness characterization of several representative heats of PCRV forgings, plate, weldments, and heat-affected zones. Testing should include evaluation of material's properties using Charpy, tensile, drop weight, compact tension and instrumented Charpy specimens. The test program should follow the guidelines set up for toughness testing of water-cooled reactor materials by the Joint PVRC/MPC Task Group on Fracture Toughness Properties for Nuclear Components.

Task 5.5.2.1 would develop the analytical methods to be used in liner and penetration safety analyses. The work should emphasize use of conventional fracture toughness data and should identify fracture correlations between conventional fracture toughness test data, analytical methods, and real component behavior.

12.11 THERMAL BARRIER (TASK GROUP 5.6)

12.11.1 Scope and justification

The PCRV thermal barrier restricts heat flow from the primary coolant to the liner surface. In Fig. 4.6 the barrier is seen to consist of fiber insulation and a retaining cover plate, which is not completely gas-tight. High-temperature locations, such as the core lower or outlet plenum, require additional refractory insulating layers.

Failures of the thermal barrier, under either normal or accident conditions, may impose severe thermal loads on the PCRV liner and its cooling system. Hence, the integrity of the thermal barrier is a significant consideration in the safety of HTGRs. This task group considers short-term, extreme transient effects, as well as long-term effects at nominal operating conditions.

12.11.2 Status

The thermal-barrier system proposed for large HTGRs has been extensively tested for normal operating conditions and for accident conditions such as depressurization and steam leakage.

GAC has completed 5000-hr tests on thermal barrier resilience at temperatures up to 1800°F. Short-term tests on friction and wear between sliding parts have been completed for temperatures up to 1500°F and for the integrated relative motion expected over the reactor life.

Further investigation and testing are desirable in the following areas: (1) extended life testing under normal operating conditions to allow better extrapolation to the full 40-year reactor life, (2) determination of failure modes under abnormal temperature conditions, and (3) properties of higher temperature materials as possible substitutes for those currently used to provide higher safety margins.

12.11.3 Identified work (tasks 5.6.1.1 and 5.6.2.1)

The work should encompass the following tests:

- (1) Testing of the resiliency of the fibrous thermal barrier should be continued out to 30,000 hr at temperatures up to 1800°F.
- (2) Testing of friction and wear properties should be extended to assess the effect of time at temperatures up to 1800°F; higher bearing pressures, up to 5000 psi; and variables in the cycling pattern, e.g., periods of no movement under high pressure followed by periods of frequent cycling.
- (3) Tests of design installations to failure under high-temperature conditions should be initiated. Testing should be conducted to define more accurately the design properties of the fused silica materials used in the plenum beneath the core support floor, with particular emphasis on evaluation of the effects on these properties of long exposure to the reactor environment.

This work should include evaluation of the effects of gaseous impurities on crystallite formation, the effects of irradiation on devitrification and shrinkage, etc., and should include definition of the effects of such changes on the significant design properties of the material.

In addition, alternative high-temperature insulators with appropriate mechanical properties should continue to be sought. The most severe conditions occur in the core outlet plenum and the lower crossducts. Some candidate materials are silicon nitride, zirconia, fused silica, and graphite. Evaluation testing of these materials should be performed.

12.12 INSTRUMENTATION (TASK GROUP 5.7)

12.12.1 Scope and justification

A major PCRV safety consideration is the capability to monitor the vessel integrity over its lifetime. This involves the development of techniques and instrumentation to measure stress, strain, free moisture content, and temperature; much of the instrumentation must be embedded within the concrete and be capable of maintaining the required accuracy after subjection to the rigors of construction. In particular, the measuring devices must withstand the vibration associated with concrete placement without significant loss of integrity or sensitivity.

A variety of highly specialized instrumentation is required for the continuous monitoring of an operating PCRV. The various types are listed according to vessel component in Fig. 12.1.

12.12.2 Status

Commercial instrumentation has been adapted to the experimental measurement of stress and strain in concrete and also for similar monitoring of the Fort St. Vrain plant PCRV. However, effective methods of installing concrete-embedded instrumentation have not yet been established, and the long-term stability and reliability of the sensing elements remains to be determined. No satisfactory method for continuous measurement of concrete water content has been devised.

12.12.3 Identified work (tasks 5.7.2.1 and 5.7.2.2)

A comprehensive study should be made of the various types of commercial concrete-embedment instrumentation for PCRV applications which involves demonstrating the stability and reliability of sensors when used under PCRV construction and operating conditions. A basic development effort is required for the water-content monitor. The feasibility of acoustic emission techniques for identifying and locating localized cracking and incipient failures in concrete should be investigated by tests. This work has been initiated as part of the ORNL model studies. Because similar instrumentation is necessary for much of the concrete materials and PCRV model testing, the instrumentation development should proceed in conjunction with these testing programs.

Preliminary tests will explore the feasibility of acoustic-emission monitoring of tendon corrosion and liner system integrity.

Techniques will be developed to detect and locate (1) cooling coil leaks, (2) cooling coil blockage, and (3) liner membrane separation from concrete, i.e., onset of buckling. Since the liner and coils are not accessible, their conditions will have to be inferred from various indirect measurements. This task should be regarded as a scoping task.

In cases where new types of instrumentation will be developed, as will be the case for the concrete moisture gage, it will be necessary to develop and conduct tests of prototype gages to ensure that they will perform adequately. In most cases, these tests can be combined with one of the tests to be conducted under a related task group.

12.13 MODEL STUDIES AND PCRV SURVEILLANCE (TASK GROUP 5.8)

12.13.1 Scope and justification

Historically, structural model testing has played a prominent role in the development of much of the general technology of reinforced and prestressed concrete, and the use of experimental models is an established procedure for evaluating novel structural designs. During the period of PCRV development, models have been used to full advantage in the evaluation of short-term vessel behavior, and model studies are an important part of failure-analysis development as was discussed under task group 5.2. Models have been made of all of the PCRVs that have been constructed thus far.

The model studies to be performed in this task group are divided into relatively simple structural models, models representing some portion of a PCRV, and complex structural and PCRV models. The simple structural models are massive slabs, beams, and simple hollow cylinders that will be subjected to variable stress and temperature in a planned sequence simulating long-term PCRV operating conditions.

12.13.2 Status

A basic type of model study was conducted at ORNL to evaluate the suitability of using small scale ($<1/20$) mortar models and epoxy models to study elastic stress distributions, cracking, and failure modes of concrete vessels and to provide data for comparison with finite element structural analyses results. The tests included one relatively large concrete "prototype" model and five scale models. The structural behavior of the small-scale mortar models was found to closely duplicate that of the prototype, and reasonably good agreement was seen between analytical and experimental results.

A prestressed concrete thermal cylinder model was recently tested at ORNL. This model simulates on a reduced scale actual long-term PCRV thermal and mechanical loading conditions. Measurements of stress strain and moisture content have been obtained using available commercial instrumentation for the most part. Post-mortem examinations are presently being made on the dismantled and sectioned model.

12.13.3 Identified work (tasks 5.8.1.1, 5.8.1.2, 5.8.2.1, and 5.8.3.1)

Measurements of stress, strain, or creep and of water content should be made throughout the various models to confirm effects predicted by analysis. Each sequence should terminate in a transient stress to failure, seeking confirmation of anticipated failure modes and stress levels. The observed behavior could be compared with the predictions regardless of how well advanced the analytical capability is; accordingly these tests should be initiated as early as practical. The second series of tests is similar but should be applied to complex geometries simulating vessel head, penetrations, etc. These tests could be delayed at least until there was some indication that the simple form tests were indeed confirming the analytical predictions and until the test techniques themselves were well developed.

Guidance for further model testing should be based on the findings of the technology assessment study. Tests of one or more complex model may prove desirable once the required analysis methods and surveillance instrumentation have been developed to provide a final

validation before their application to prototype PCRVs. Every effort should be made to ensure that the reliable instrumentation and accurate analytical methods have been developed.

The ultimate validation of concrete materials and PCRV design technology should be sought through surveillance of PCRVs of operating reactor plants. This task is indicated in Fig. 12.1 for completeness, but there is no present basis on which to estimate when or at what level it should be conducted. Early funding should be provided for surveillance planning since design provisions for PCRV instrument installations or inspection access may be desirable.

12.14 CONTAINMENT CONVECTION STUDIES (TASK GROUP 5.9)

12.14.1 Scope and justification

This task group deals with both natural-convection heat transfer within the PCRV and forced-convection within the reactor containment structure. The objectives are (1) to evaluate the natural-convection heat transfer in several regions within the PCRV and (2) to develop analysis tools for the evaluation of forced-convection thermal and fluid dynamics during a helium or steam blowdown into the reactor containment. These studies are important in the safe design of the containment structures and related protective equipment.

Within the primary coolant loop of the HTGR, there are several regions of low helium flow or stagnant helium either during normal operation or during the loss of main circulator cooling. In these regions natural-convection flows and heat transfer can develop. Two such related cases are a natural-convection reversed flow from the core to the upper plenum during the loss of the main circulators and the subsequent natural convection of this hot helium through the upper crossducts to the auxiliary circulators before they are started. The degree of this natural convection relates to the safe design of the liner thermal barrier in the upper plenum and upper crossducts, as well as of the auxiliary circulator itself.

Prediction of the natural convection flow and heat transfer in the upper plenum and crossduct is difficult because experimentally based correlations or analytical tools are not available for these geometries and/or the high values of Grashof number encountered (which yield turbulent natural-convection flows). A similar situation exists in other enclosures within the PCRV.

Subsequent to a failure of a PCRV penetration or the rupture of steam loop piping, there is a rapid blowdown of helium or steam into the reactor containment. The computer code CONTEMPT is currently used to calculate the response of the containment pressure and temperature for such blowdown transients. To be conservative in calculating the peak containment pressure and temperature, it is assumed in the CONTEMPT code that the heat transfer to the containment walls and the equipment within the containment is solely by natural convection. Preliminary analyses indicate, however, that high velocity flows or jets may occur in portions of the containment during the initial stages of the blowdown when the flow at the rupture point is choked. These high velocity flows can cause forced convection heating of local portions of the containment liner, creating high thermal stresses or buckling, or can produce high temperature transients for plant-protective instrumentation or equipment. The high velocity flows can also produce large fluid forces on various pieces of equipment in the immediate vicinity of the rupture point, possibly causing flying objects.

An exception to the assumption of natural-convection heat transfer occurs when the saturation temperature of the steam exceeds a wall surface temperature. Here, condensation heat transfer is allowed. Unfortunately, prediction of such condensation in the presence of the containment atmosphere is difficult, and no reliable data exist.

To ensure the integrity of the containment liner and to provide for the safe design of plant-protective equipment for blowdown accidents is not a routine analysis/design effort. A knowledge of forced-convection flow patterns and surface heat transfer coefficients is first necessary, as well as the development of the analytical tools to predict these forced convection effects. For the steam leak accident, accurate prediction of condensation heat transfer is also required. The objective of the proposed program is to provide the needed knowledge and analytical tools.

As further justification for the proposed program, it should be realized that the containment design and the layout and design of much of the equipment within the containment is usually the responsibility of the architect-engineer for the particular plant. As individuals, these architect-engineering companies do not have the capabilities to develop the necessary information and analytical tools to evaluate forced-convection effects for blowdown accidents. This program will provide reliable and consistent data and tools for all the architect-engineering companies designing HTGR containments.

12.14.2 Status

GAC has attempted to provide analytical tools for evaluating natural-convection heat transfer in enclosures. A computer code for calculating two-dimensional, forced-convection viscous flows problems has been restricted to laminar viscous flows, and only recently has the state of the art been extended to certain classes of turbulent flow problems. GAC has also performed an extensive literature search to compile published information on natural-convection heat transfer in enclosures. Again, many of these correlations fail to satisfy the geometries and/or Grashof number values encountered with the HTGR.

Essentially no work has been done on evaluating forced-convection effects within the containment during HTGR blowdown accidents. GAC has developed the CONTEMPT code to evaluate the transient pressure and temperature response within the containment; however, it does not consider forced-convection effects. GAC has just completed a small preliminary study of forced-convection thermal effects for the helium blowdown accident. The study indicates that local heating is severe and that the reliable prediction of flow patterns and forced-convection heat transfer is difficult. The results of studies for the blowdown in LWR containments are generally not applicable to the HTGR problem.

12.14.3 Identified work

The study of containment forced-convection effects due to a helium or steam blowdown accident (task 5 9.2.1) should consist of the following subtasks:

- (1) Identify the worst possible helium and steam-leak flow patterns with regard to the integrity of the containment liner and the safe design of the plant-protective equipment.
- (2) Develop preliminary analytical models to evaluate the thermal responses of local regions of the containment wall and of equipment within the containment as well as the fluid forces on objects in the vicinity of the rupture. Besides providing the basis for the final analytical tools, these preliminary models should identify clearly where experimental data are needed.
- (3) Conduct an experimental program with a scaled-down model to verify flow patterns, to measure forced-convection heat transfer coefficients with both helium and steam, and to record containment pressure and temperature responses. These data could be used in the analytical models as well as to verify the overall results of analytical predictions.

- (4) Develop the final analytical models and a new blowdown analysis code incorporating forced-convection effects.

The study of natural convection of helium within the PCRV (task 5.9.2.2) should consist initially of the development of turbulent models for the numerical solutions of viscous flow problems. As these models become available, they should be adapted to natural-convection flows and incorporated in GAC's viscous-flow code. An experimental program should also be planned for simple but representative HTGR-type geometries and at Grashof numbers in the appropriate range to verify the turbulent models and the overall results of the natural-convection code. The specific reversed natural-convection flow into the upper plenum and the subsequent natural-convection flow through the upper crossducts during the loss of the main circulators should be analyzed.

12.15 CODE RULE DEVELOPMENT (TASK GROUP 5.10)

12.15.1 Scope and justification

The draft version of Section III, Division 2 of the proposed ASME Boiler and Pressure Vessel Code governs the design of both PCRVs and reactor containment structures. Two public hearings were held on the code which is currently in the trial-use period, and the prepared comments are to be evaluated for possible inclusion in the final version of the Code. Publication of the formal code is scheduled for late 1974. The code is to be revised where necessary and updated on a semiannual basis by published addenda. The need exists for the collection and evaluation of research and development data from ongoing programs and, when applicable, preparation of the resulting information in forms suitable for review and consideration by appropriate code bodies.

12.15.2 Status

The Concrete Pressure Component Research Committee was recently established under the Joint ACI-ASME Technical Committee on Concrete Components for Nuclear Service. The purpose of this committee is to provide a much needed means of communication between the code bodies and research organizations to ensure that the ongoing research serves a definite need and that the resulting data will lead to the development of more effective safety requirements for PCRVs and reactor containments.

12.15.3 Identified work (tasks 5.10.1.1 and 5.10.2.1)

As data become available from this task area and other relevant programs, they should be collected, evaluated, and correlated. Comparisons should be made between theoretical results and experimental data to evaluate the accuracy of the candidate analytical methods. The various practical aspects of PCRV construction, design, and proof testing should also be considered in light of information developed under this program and other technically related programs. In many cases the proven design methods could be used together with experimental data to develop design rules for PCRVs and reactor containments suitable for inclusion in the appropriate ASME Boiler and Pressure Vessel Code and RDT standards. This phase of the work should be done in close cooperation with a steering committee made up of practicing PCRV and reactor containment designers and development engineers as well as the Joint ACI-ASME Concrete Pressure Component Research Committee. These committees should identify and assign priorities to the developing safety-related research needs that will provide guidance to future studies in this task area.

REFERENCES FOR SECTION 12

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13. PRIMARY SYSTEM MATERIALS TECHNOLOGY (TASK AREA 6)

13.1 TASK AREA OBJECTIVES

Task area 6 is concerned with the technology of the materials used in the construction of the primary system of an HTGR including the core and steam generator. For several reasons, these materials are separated into three categories as follows:

- (1) control components (control rod absorber materials and burnable poisons),
- (2) graphite components (support blocks, reflectors, etc.),
- (3) primary loop metallic components (e.g., ducting, steam generator, and circulator).

The objective for each of these categories is the development of information that will permit more precise definition of material response and behavior during normal reactor operation and under abnormal conditions and that, thereby, will provide improved bases for the evaluation of safety.

13.2 SCOPE

Task area 6 is divided into four task groups corresponding to the need assessment group and the three categories defined above, each dealing with a different aspect of materials technology. Each task group will be discussed separately.

13.2.1 Need assessment for primary system materials technology (Task group 6.0)

This task group provides continuing assessment of primary system materials technology with the aim of identifying areas in which additional safety margins, features, and/or work could add substantial improvements in HTGR safety.

13.2.2 Control components (Task group 6.1)

Research in this task group will include study of the reaction of HTGR control materials with coolant impurities and the boron transport processes. Structural stability of control materials will be evaluated, and the integrity and compatibility of control rod cladding

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materials will be determined. Models for predicting the behavior of control materials will be developed and validated by tests on control materials removed from the Fort St. Vrain reactor.

13.2.3 Graphite components (Task group 6.2)

The thermomechanical properties of irradiated graphite under non-steady-state conditions will be determined as will the effect of partial annealing (at temperatures above design conditions) on these properties. These data will be used to construct time-dependent constitutive equations to assist safety-related analyses. Data from early HTGRs will provide substantiating information.

13.2.4 Primary loop metallic components (Task group 6.3)

Research will be directed primarily toward the mechanical properties (tensile, creep, rupture, fatigue, etc.) of structural materials and weldments with appropriate consideration of environmental effects (thermal, chemical, and radiological) and corrosion. Failure criteria will be determined, and analysis methods for safety evaluation will be developed and verified. This task group and task group 3.2 are complementary.

13.3 STATUS AND NEEDS

13.3.1 Need assessment for primary system materials technology (Task group 6.0)

Several factors contribute to a need for continuing assessment of the safety aspects of HTGR primary system materials. First, there is little long-time in-reactor experience with the materials of current HTGR system design. If some unexpected materials behavior is observed during reactor life, it must be assessed and related to safety. If design or operating changes are made, the effects of these on materials behavior and safety must be determined. Any new materials used in the future will also require safety assessment.

13.3.2 Control components (Task group 6.1)

Volatile boron oxides are known to be formed by reaction of B_4C with air and/or water vapor. The extent of such reactions in HTGR environments has not been studied. Further, data are needed on the swelling of boronated graphites of interest for HTGR control materials

and on the effects of thermal transients. Finally, the compatibility of Incoloy 800 with boronated graphite and the reactor environment needs to be demonstrated.

13.3.3 Graphite components (Task group 6.2)

Although significant advances have been made in the understanding of the behavior of reactor graphites, a number of matters need to be clarified. These include the effect of irradiation temperature on creep and elastic properties. At present, no creep data are being obtained on current types of moderator graphites, and mechanical properties are being measured primarily at room temperature rather than at actual operating temperatures.

13.3.4 Primary loop metallic components (Task group 6.3)

Detailed and accurate information on creep, fatigue, and other properties at elevated temperatures is needed as input to the formulation of structural design methods, failure criteria, and safety analyses. Analysis of the long-term behavior of HTGR metallic components must include consideration of effects such as corrosion, carburization, and thermal aging. Also, specific attention is needed on the properties and behavior of weldments. Although considerable amounts of relevant data are available for several HTGR materials, in no case is the data base satisfactory in all respects. In addition, several structural materials of potential use are not currently being studied in detail.

13.4 TASK AREA EFFORT AND SCHEDULE

Table 13.1 provides an overview of the relative effort associated with the work identified in this task area. The task groups are treated in summary fashion below and in more detail in the following sections.

A significant activity is the study of technology status and identification of needs as they apply to safety. This activity is expected to continue for the duration of the safety program and may result in the definition of additional safety work.

In task group 6.1, the three source tasks show activity continuing over a four-year period. The priority task 6.1.1.1 calls for intensive effort over this period. Tasks 6.1.1.2 and 6.1.1.3 would be expected to phase out as materials from Fort St. Vrain are examined in the validation task 6.1.3.1. The synthesis task 6.1.2.1 would require

Table 13.1. Summary of identified safety and safety-related work in task area 6
Task groups or tasks marked with an asterisk (*) are included in the safety planning guide (Part II of this document)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 6.0 Need assessment*										
6.0.1 Source tasks	1	1	1	1	1	1	1	1	1	1
6.0.1.1 Control materials safety technology needs										
6.0.1.2 Graphite materials safety technology needs										
6.0.1.3 Metallic materials safety technology needs										
Task group 6.0 totals	1	1	1	1	1	1	1	1	1	1
Task group 6.1 Control materials										
6.1.1 Source tasks										
6.1.1.1 Reactions of coolant impurities with HTGR control materials. Assessment of potential for boron movement due to water vapor, etc. Effect of temperature, impurity concentrations, B ₄ C content and distribution, and irradiation.	4	4.5	4.5	0.5						
6.1.1.2 Structural stability of HTGR control materials. Irradiation stability of full-sized compacts and effects of thermal transients.	2	2	2	1						
6.1.1.3 Integrity of control-rod cladding materials. Determination of compatibility effects on mechanical properties of cladding and thermal aging on shock absorber capabilities.	2	2	2	1						
6.1.2 Synthesis task										
6.1.2.1 Control materials modeling. Development of model for control material performance including structural stability and boron movement.	1	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
6.1.3 Validation task										
6.1.3.1 Fort St. Vrain control-rod materials surveillance tests. Examination and testing of control-rod absorbers and cladding, reserve shutdown system, and burnable poison.	0.2	0.5	1.5	1	1	1	1			
Task group 6.1 totals	9.2	9.5	10.5	4	1.5	1.5	1.5	0.5	0.5	0.5

Table 13.1 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 6.2 Graphite components										
6.2.1 Source tasks										
6.2.1.1 Effect of temperature, fluence, and materials properties on creep coefficients.	4	2	2	2						
6.2.1.2 Determination of physical properties above the normal operating temperatures of an HTGR.	4	3	3							
6.2.1.3 Effect of thermal history on irradiation behavior.		3	2	2	2					
6.2.1.4 Annealing effects on graphite properties.	1	2	2	2						
6.2.1.5 Effect of cyclic loading on the stress-strain curve.			1	1	1					
6.2.1.6 Determination of thermomechanical properties under reactor operating conditions.				2	2	1				
6.2.2 Synthesis task										
6.2.2.1 Development of time-dependent constitutive equations.	1	1	1	1	1	1				
6.2.3 Validation task										
6.2.3.1 Reactor surveillance.					2	2	2	2	2	
Task group 6.2 totals	9	10	11	10	10	4	2	2	2	
Task group 6.3 Primary loop metallic components										
6.3.1 Source tasks										
6.3.1.1 Technology status report. Evaluation of the status of technology of HTGR metallic structural materials.	10	15	20	20	15	10	5	2		
6.3.1.2 Mechanical properties and failure criteria for HTGR structural materials. Mechanical property information needed to develop safety evaluations for HTGR primary circuit structural materials and weldments will be obtained.										
6.3.2 Synthesis task										
6.3.2.1 Design analysis methods for HTGR safety evaluation. Formulation of models of material behavior to permit safety analyses under normal and accident conditions.	Effort in 6.3.1.1									

Table 13.1 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
6.3.3 Validation tasks										
6.3.3.1 Validation of design analysis methods (simple geometries). Simple structural tests to verify the capability of analysis methods for material behavior.	Effort in 6.3.1.1									
6.3.3.2 Design analysis and failure-mode validation (complex tests). Large-scale tests to examine the response of HTGR primary circuit components to service and accident conditions.				2	6	8	4	3		
6.3.3.3 Examination of components from Peach Bottom HTGR. Removal of primary circuit components from Peach Bottom HTGR. Conduct of complete examination including mechanical property determinations.	1	5	5	2						
6.3.3.4 Validation of alternative steam generator materials. Demonstration of the feasibility of employing high alloy materials in the low-temperature regions of the HTGR steam generator.	2	3	4	4	2					
Task group 6.3 totals	13	23	29	28	23	18	9	5		
Task area 6 totals	32.2	43.5	51.5	43	55.5	24.5	13.5	8.5	3.5	1.5

an early effort to set up the control material model followed by a lower continuing effort to update the model as a result of the other tasks and to provide guidance for the source tasks. Validation of the model would be through task 6.1.3.1 where Fort St. Vrain control materials will become available starting in FY 1976 and continuing periodically for six years.

Figure 13.1 shows the interaction of tasks in task group 6.2. The source tasks are initiated sequentially and in order of priority to flatten the associated effort over the early periods. Task 6.2.1.6 may also receive support from other than safety considerations. The synthesis task 6.2.2.1 could be initiated immediately to help guide the experimental program, since many of the required experiments are expensive. Validation experiments would initiate with operation of the first large HTGR, about 1980.

In task group 6.3, because source tasks 6.3.1.1 and 6.3.1.2, synthesis task 6.3.2.1, and validation task 6.3.3.1 are so closely interrelated (i.e., feedback and interaction are likely among these tasks), the effort related to these has been collected under a single total in the table. Estimates shown for 6.3.3.2 (design analysis and failure-mode validation) are speculative at present since the need and scale of such experiments must be determined from results in other of the 6.3 tasks — especially from 6.3.3.1 (validation of design analysis methods).

The effort associated with the examination of Peach Bottom HTGR components will be strongly dependent on component removal costs, which are not included in this tabulation. The schedule of effort may move in time, depending on the shutdown date of Peach Bottom.

Task 6.3.3.4 (validation of alternative steam generator materials) is independent of all other tasks listed under 6.3. Likewise, except that results from 6.3.1.1 (mechanical properties and failure criteria for HTGR structural materials) will be useful in interpretation of observations, task 6.3.3.3 (examination of components from Peach Bottom HTGR) is also independent in nature. The interaction of all other 6.3 tasks is shown in Fig. 13.2.

13.5 NEED ASSESSMENT FOR PRIMARY SYSTEM MATERIALS TECHNOLOGY (TASK GROUP 6.0)

13.5.1 Scope and justification

As changes in design and materials are made in HTGR systems, the performance and behavior of primary system structural materials

Fig. 13.1. Effort for task group 6.2.

Fig. 13.2. Effort for task group 6.3.

must be accessed relative to safety and safety margins. Further, knowledge obtained in base program studies and through reactor operating experience also forms a basis for needed safety assessment.

13.5.2 Status

The design of HTGR systems has necessarily included assessments of the type being discussed. Additionally, similar assessment has been performed in the course of developing this document.

13.5.3 Identified work

Need assessment for control materials (task 6.0.1.1)

The need for improved control materials or further testing of present materials may become apparent as a result of the efforts associated with source tasks 6.1.1.1 and 6.1.1.2 and/or related activities in engineering and technology program. Proposed changes in control materials will need to be evaluated to determine if further safety or safety-related testing or analyses need to be performed. The performance of boronated graphite control materials is strongly related to the graphite component. For this reason, a continuing assessment of information generated on graphite components (task group 6.2) is appropriate for control materials as well.

Need assessment for graphite materials (task 6.0.1.2)

The properties and behavior of graphite employed as core supports, reflectors, and as a moderator in the HTGR primary system will be assessed by this task to identify any safety-related areas that should be examined further.

Need assessment for metallic materials (task 6.0.1.3)

The metallic materials used in the steam generator, thermal barriers, and circulators have important safety significance. Any safety-related problems involving the behavior of these materials will be identified in this task through continuing overview of materials technology and operating experience.

13.6 CONTROL MATERIALS (TASK GROUP 6.1)

The reference neutron absorber materials employed in GAC HTGR control rods, reserve shutdown compacts, hexagonal reflector blocks, and lumped burnable poison consist of discrete particles of B_4C

dispersed in a graphite matrix. Each material has unique characteristics that reflect its operating service requirements. The boronated graphite used in control rod compacts is made up of B_4C granules from 38 μm to 250 μm homogeneously dispersed in a graphite matrix made from near-isotropic graphite filler. The fabricated compact containing 40 wt % B must have a compressive strength greater than 3000 psi, anisotropy between 0.8 and 1.2, and less than 1 wt % of the boron in the oxide form. The compacts are contained in an Incolloy 800 sheath to position them in the control rods and to protect them from oxidation. The sheaths have vent holes to equalize internal and external pressure. The control rods and reserve shutdown compacts are inspectable, and a periodic replacement of the control rods is planned.

The reserve shutdown compacts, which are 40 wt % B, are fabricated with B_4C granules from 38 μm to 300 μm dispersed in a graphite matrix to form spheres or pellets 9/16-in. in diameter. The compacts must withstand impacts resulting from a drop through the height of a reactor core. During standby, the reserve shutdown pellets are also separated from direct circulating contact with the coolant helium by a metal barrier. In use, the pellets would fall into blind holes where they would not be in the path of the circulating coolant. B_4C particles 38 μm to 250 μm dispersed in a near-isotropic graphite rod make up the 25 wt % B hexagonal reflector shield compacts. The 7.3-in.-long, 0.42-in.-diameter rods are sealed in Incolloy 800 or stainless steel cans and placed in holes in the hexagonal reflector blocks. The lumped burnable poison compacts consist of B_4C granules sized less than 20 μm homogeneously dispersed in a near-isotropic graphite matrix. The boron loading varies from 1 wt % B to 5 wt % B for different rod types. The rods are loaded into fuel blocks that are replaced on a four-year cycle.

13.6.1 Scope and justification

The reference neutron-absorber materials (boronated graphite) employed in HTGR control rods, the reserve shutdown system, hexagonal reflector blocks and burnable poison consist of discrete particles of B_4C in a porous graphite matrix. The oxidation behavior of the boronated graphites is significant for all applications because oxidation rates, if high, could lead to redistribution of unprotected boron in the core through transport of volatile B_2O_3 . The extent of this oxidation and the potential for boron transport need to be determined for long-term normal operating conditions with low concentrations of oxidizing impurities (including oxygen and water vapor) in the coolant and for accident conditions in which concentrations of such impurities could increase significantly above normal levels (<10 vpm).

The integrity of the control rod containment and shock absorber during reactor operation must be maintained. The radiation-induced dimensional changes in control-rod compacts and burnable-poison compacts must be well known so that components can be designed to avoid mechanical interaction between the control material and its containment. The effects of thermal transients on the control materials should be determined. Thermal aging, irradiation damage, and effects on materials of long-term exposure to boronated graphite and low-level coolant impurities should be determined.

Surveillance of neutron-absorber-containing components is required to verify the adequacy of designs.

13.6.2 Status

The formation of a volatile boron oxide caused by exposure of B_4C to air and/or water vapor has been observed.¹ The oxidation behavior of Fort St. Vrain burnable poison compacts was studied at several temperatures and oxidant concentrations representative of the most severe HTGR conditions. The investigation concluded that the oxidation rate would be acceptable under Fort St. Vrain operating conditions including postulated accidents. However, similar tests performed, on encapsulated, but vented, control rod materials have not been performed, and no information exists on whether this condition is aggravated by neutron irradiation.

Radiation induced swelling of boronated graphite² containing 5 and 7 wt % boron has been observed, and dimensional changes have been determined for Fort St. Vrain control rod materials to conditions beyond lifetime service requirements.³ However, no data have been reported for Fort St. Vrain burnable poison materials or full-sized control rod compacts. Similarly, no data have been reported on the effects of thermal transients on such irradiated material.

Information on the compatibility of Incoloy 800 with 40 wt % boron-containing graphite is not documented. Of particular interest would be the effect of low boron oxide concentration. Similarly, the impact capability of the thin-gage Incoloy 800 shock absorber after long-term exposure to the reactor coolant has not been reported.

13.6.3 Identified work

Reactions of coolant impurities with HTGR control materials (task 6.1.1.1)

The potential for movement of boron from boronated graphite under possible accident conditions, with high concentrations of water

vapor and other oxidizing impurities in the helium coolant, should be assessed. The potential for boron movement under long-time exposure to low concentrations of these impurities should also be assessed.

Boronated graphite should be tested under various conditions of temperature and impurities to determine reaction rates and the nature of the reaction products. Reaction rates should also be determined for boronated graphites with different B_4C contents and B_4C particle sizes. The behavior of irradiated material should be determined to see if the potential for boron movement is accelerated by irradiation-induced swelling and associated porosity of fragmentation. The effect of ionizing radiation on rates should also be determined. These data would provide input to task 6.1.2.1.

Structural stability of HTGR control materials (task 6.1.1.2)

Irradiation-induced swelling of boronated graphites should be determined through a program of irradiation testing and examination. Of particular interest would be the stability of irradiated material to thermal transients representing conditions appropriate to interruption-of-cooling accidents. Thermal transient testing should be conducted on materials with various fluence and irradiation temperature histories.

The products resulting from these tests would provide samples for the oxidation stability tests in task 6.1.1.1. The results of these tests would be factored into task 6.1.2.1.

Integrity of control rod cladding materials (task 6.1.1.3)

The compatibility of boronated graphite control materials with the sheathing material should be evaluated. In addition to metallographic examination to determine the extent of visually observable interaction, the effect of any interactions on the mechanical properties of the sheath material should be determined. The effects of long-term exposure to low-level coolant impurities on the mechanical properties of cladding materials should also be determined.

In addition, the effects of thermal aging, irradiation damage, and exposure to low-level coolant impurities on the ability of the control-rod shock absorber to perform its function should be evaluated in impact tests.

Control materials modeling (task 6.1.2.1)

A model for control materials and associated containers should be developed which will address the potential loss of boron from oxidizing impurities in the helium coolant as well as irradiation swelling and the superimposed effects of thermal transients. Input would be received from all other tasks in this task group. The model would be useful in predicting control-rod material behavior under accident and normal operating conditions, as well as for various proposed changes in control materials.

Fort St. Vrain control materials surveillance tests (task 6.1.3.1)

Periodically a high-burnup control rod should be removed from the Fort St. Vrain reactor and examined to determine the conformity of behavior with design requirements and the control materials model developed in task 6.1.2.1. Examinations would include determinations of radiation-induced dimensional changes and tests of metallographic and mechanical properties of sheathing. Segments of the control rod should be exposed to oxidizing helium atmospheres to test the ability of the control materials model to predict boron losses for this particular capsule and its vent geometry.

The condition and dimensional changes of the burnable poison should be examined as part of the Fort St. Vrain fuel-element surveillance program.

The reserve shutdown system should also be examined periodically to determine the reliability of the release mechanism, ball movement, and validation of the boron content of the balls.

13.7 GRAPHITE COMPONENTS (TASK GROUP 6.2)

13.7.1 Scope and justification

Graphite is employed in several ways in the HTGR, including use as: core support, permanent reflector, removable reflector, and moderator. Conditions in the support and permanent reflector regions are much less severe than those in the removable reflector and moderator regions and impose no critical requirement, other than the obvious one of strength, on the graphite. Radiation damage to the removable reflector blocks requires their periodic replacement, but conditions in this region are less severe than those for the moderator region. The moderator blocks require major safety considerations. Not only are these subject to the most severe conditions, they must also serve as a container for the fuel sticks.

Of the various accident modes considered, the effects on the graphite fall into two categories — temperature increases (either rapidly due to interruption of cooling or reactivity excursion or prolonged due to partial channel blockage) and increased stresses due to redistribution of thermal and flux gradients or to externally applied loads.

These considerations are compounded by the fact that graphite possesses a "memory" (i.e., its state at any given time depends on its past history). This sensitivity is manifested in a number of ways, such as radiation-induced creep, fatigue and hysteresis under cyclic loading, and partial recovery of physical properties under even modest annealing conditions.

The analyses of the consequences of an accident and the ability to restart the reactor after an accident require definitive data on off-design behavior and memory effects. The general objectives of this task group are, therefore, (1) to determine the thermomechanical properties of irradiated graphite as a function of non-steady-state history; (2) to determine these properties at temperatures above design conditions and after partial annealing; (3) to systematize the data into time-dependent constitutive equations that permit analyses of accident-induced conditions; and (4) to survey early operating reactors to provide substantiating information.

13.7.2 Status

A review of the current status of information on graphite for HTGR applications is given in ref. 4, which also includes an extensive bibliography of recent research and development. This review reaches the following conclusions:

"A complete understanding of the irradiation behavior of reactor graphites at high temperatures and high fluences has not been attained, but a number of significant advances have been made during the past six or seven years. The irradiation data have been extended to fluences of 4×10^{22} n cm⁻² and temperatures into the range 900 to 1400°C. The irradiation of reactor graphites has been paralleled by similar studies on as-deposited pyrolytic carbons and annealed pyrolytic carbons which approach single crystals in structure.

New insights have been made into the mechanism of irradiation damage to crystallites through electron microscopy and the nature and mobility of defects in

single crystals is now better understood. However, direct observation of the neutron-induced defects in reactor crystallites continues to be retarded by experimental difficulties in sample preparation and the poor diffraction qualities of the reactor graphites. Progress has been made in analyzing dimensional changes and certain property changes such as thermal conductivity and Young's modulus in reactor graphites in terms of defects observed in the annealed pyrolytic carbons and single crystals, but again the direct observations of damage in reactor graphite crystallites would be helpful to confirm these interpretations. Microstructural changes in reactor graphites have been measured and explained, but only in qualitative terms. Better quantitative experimental techniques of measuring crystallite and pore morphology changes are required in this area.

The effect of irradiation temperature on the dimensional change rate of reactor graphites in the range 900 to 1400°C needs to be clarified. There appears to be a temperature range from 900 to 1200°C where structural degradation is severe, whereas above about 1200°C lower dimensional change rates have been reported. However, data are not abundant above 1200°C and anomalies exist. More experiments are needed at 900 to 1400°C and improvements in thermometry and dosimetry above 1200°C will be required.

Mechanical properties of nuclear graphites have not been measured extensively. Systematic studies of changes in modulus and creep have been reported, but very little has been done to determine the effect of high-temperature high-fluence irradiation on strength, and the existing data are in conflict. There is dire need for stress-strain curves of irradiated graphites. Irradiation-induced creep constants have been determined as a function of irradiation temperature. However, current data are not abundant or precise enough to reveal whether there is a structural effect in addition to a temperature effect."

Much of the data called for in this summary are currently being obtained both in this country and Europe but are subject to sharp restrictions: creep data are not being obtained on current types of U.S. moderator graphites, and other mechanical properties are being largely measured at room temperatures rather than at actual operating conditions.

The thrust has also been almost exclusively toward design data. A fair body of data exists on annealing effects for graphites irradiated at 550°C and below, but very little is available on temperatures above this. Further, experiments were begun several years ago to examine the effects of complex thermal histories on radiation damage in graphite. Due to funding limitations, these experiments were discontinued before reaching definitive conclusions.

A statistical basis for treatment of graphite properties is being obtained by General Atomic Company and need not be considered as a safety-related problem. The tasks proposed below are meant to supplement other existing programs and include only those areas basic to safety considerations.

13.7.3 Identified work

Of the following eight tasks, the first six are generally listed in order of priority; the last two must follow the earlier tasks. This relative priority has been assigned on the basis of two considerations: the intrinsic importance of the data to the core behavior and the relevance of the data to a particular accident type and its severity.

Effect of temperature, fluence, and materials properties on creep coefficients (task 6.2.1.1)

The existence of a nonthermal, radiation-induced creep in graphite is essential for its nuclear application. Radiation-induced distortion would quickly develop stresses beyond the capability of the material if these stresses were not relaxed by creep. The ability of graphite to creep in response to stress appears to reach a minimum at about 500°C. Creep coefficients under both tension and compression are required in the range 600 to 1400°C for the isotropic-type graphites to be used in large HTGRs. Moreover, these measurements must be carried out to full-life fluences.

Determination of physical properties above the normal operating temperatures of an HTGR (task 6.2.1.2)

Thermal expansion coefficients are required at low and high fluences for graphites irradiated in the 900 to 1400°C range to determine the thermal stresses developed under transient or off-design conditions. Young's moduli in tension and compression are different and need to be determined for the same fluence and temperature conditions. Data on Poisson's ratios, thermal conductivities, and

ultimate strains are equally important. This task is differentiated from the creep experiment (task 6.2.1.1) in that creep requires an in-reactor measurement, whereas the other thermomechanical properties can be measured after irradiation.

Effect of thermal history on irradiation behavior (task 6.2.1.3)

There appears to be a distinct difference in the dimensional behavior of graphites irradiated under varying temperature histories, depending on the order in which the temperature changes are introduced. In terms of load following and transients, this may have a significant effect on the core response to accident conditions. Insufficient information exists to assess the importance of thermal history on behavior, and at least some definitive experiments are required.

Annealing effects on graphite properties (task 6.2.1.4)

For lightly irradiated graphites, significant annealing of physical properties can result from even as small as a 100°C temperature excursion. As damage proceeds, the amount of potential recovery decreases steadily. Very little information on annealing effects exists for materials irradiated above 600°C to significant fluences. The degree of recovery as a function of annealing temperature to 2000°C and of fluence is required for immediate postexcursion analysis; it is equally important in assessing whether or not the reactor can be safely restarted.

Effect of cyclic loading on the stress-strain curve (task 6.2.1.5)

As mentioned above, not only are the compressive and tensile properties of graphite different, but there is a distinct hysteresis effect on even one single loading cycle. The hysteresis is a function of load and number of cycles. Further, a fatigue effect exists on repeated cycling which becomes significant as the load approaches the ultimate strengths. These effects have not been investigated on radiation-damaged graphites and are obviously of concern for load-following operation, transient effects, and accident response.

Determination of thermomechanical properties under reactor operating conditions (task 6.2.1.6)

This task is design as well as safety related, but elevated temperature measurements are expensive and have been carried out only to a limited degree. This task is given a relatively low priority because extrapolations from low temperature measurements can probably be extended to high temperatures during at least the first half of the core life. This requires substantiation, as well as a definition of the fluence at which extrapolation becomes unreliable. If the latter restriction is within the potential life of the graphite, further data must be obtained.

Development of time-dependent constitutive equations (task 6.2.2.1)

This task ties the previous six tasks together. Stress analysis of the graphite at any point in time requires that the thermomechanical properties and radiation-induced distortion be known as functions of time, temperature, and fluence. Details of accidents and their effects are not necessarily required and can probably be calculated as limiting cases both during and subsequent to the accident.

Reactor surveillance (task 6.2.3.1)

The graphite to be used in the 770- to 1100-MWe reactors will be different from that currently employed in the Fort St. Vrain reactor. The Fort St. Vrain graphite is anisotropic and quite similar to AGOT, the graphite generally employed in the Hanford reactors. Surveillance of the Fort St. Vrain graphite is therefore, of limited interest. The isotropic grades being considered for the larger reactors have superior response to radiation, develop smaller strains, and have longer mechanical integrity. Provision should be made for surveillance of graphite from an actual operating reactor, with sufficient material included to provide for critical property data. This would permit testing of both the constitutive equations and their data basis.

13.8 PRIMARY LOOP METALLIC COMPONENTS (TASK GROUP 6.3)

13.8.1 Scope and justification

Many of the metallic components in the primary circuit of an HTGR are exposed to temperatures at which their behavior may be highly time-dependent (e.g., creep strength or long-time thermal stability

may be limiting factors in their use). In addition, during abnormal or accident conditions, many of these materials will be subjected for short periods of time to temperatures considerably above those experienced during normal reactor operation. The objectives of this task group are, therefore, to provide (under conditions above) additional information necessary to provide an improved basis for (1) determining the response of these materials to loading, (2) formulating failure criteria, and (3) identifying the influence of environment (i.e., temperature, time, and coolant impurities) on behavior. In short, this task will provide improved bases for evaluation of the safety of design as related both to normal and abnormal conditions.

At present, metallic components forming the pressure boundary of reactor primary circuits are designed to the rules of Section III of the ASME Boiler and Pressure Vessel Code. In addition, when such materials are required to operate in the creep range, the elastic-plastic and creep behavior of these materials are required to be considered in detail (e.g., as in ASME Boiler and Pressure Vessel Code Interpretation Case 1331-8). Design and safety criteria and analyses, whether performed on the basis of ASME or other rules, must incorporate knowledge of such phenomena.

All of the above require detailed and accurate information on the elevated temperature creep, fatigue, etc., of structural materials. This information provides the input to the formulation of structural design methods, failure criteria, and safety analyses. Assurance of the applicability of these formulations requires their validation through multi-axial experiments and tests of scale-model structures.

Analysis of the long-term behavior of HTGR metallic components must also include consideration of any effects on mechanical properties of impurities contained in the coolant. (e.g., the effects of oxidation, carburization, etc., on creep and fatigue). Additionally, it is important to ascertain any deleterious effects of long times at temperature on properties. For instance, is aging or thermal embrittlement occurring and how might this affect the performance of materials during normal operating and accident conditions?

The properties of weldments should receive special attention. The behavior of material in the weld deposit and in the heat-affected zone is quite likely to differ somewhat from that in the base metal. Studies and evaluations similar to those performed on the base materials are necessary to assess differences in properties. In addition, weldments between dissimilar base metals (e.g., Incoloy 800 to 2-1/4 Cr-1 Mo steel) must be evaluated.

Finally, use of austenitic or other high-alloy materials in the lower temperature (evaporator) region of the HTGR steam generator may be desirable in the future (e.g., to limit tritium permeation). However, substitution of the low alloy 2-1/4 Cr-1 Mo ferritic steel (having relatively high tritium permeability) with Incoloy 800 or a higher alloy steel will require validation of the corrosion resistance and surface film properties of the chosen material under the expected conditions of use.

This task group is not intended to satisfy all needs for design data and methods. The primary aim is to provide specialized information to permit safety-oriented analyses.

13.8.2 Status

Of the structural materials of interest for the HTGR primary circuit, Incoloy 800 has probably received the greatest attention. Several studies relative to its creep, rupture, and fatigue properties have been published,⁵⁻⁸ and it is qualified under ASME Code Case Interpretation 1331-8 for use in nuclear service to 1400°F. However, additional information — including the effects of coolant impurities, long-time exposure to temperature, and properties at temperatures above 1400°F — is desirable. The effects of impurities on creep properties have been studied by the UKAEA.⁹ Lower rupture strengths were found in impure helium than in air under some conditions. Limited study of Incoloy 800 is also in progress in connection with the LMFBR steam generator.

2-1/4 Cr-1 Mo ferritic steel is also qualified for elevated temperature nuclear service under 1331-8. However, only limited information has been reported on fatigue, creep-fatigue interactions, environmental effects, etc. Some work on environmental effects is being sponsored by the UKAEA.¹⁰ In addition, GAC is sponsoring a program of creep fatigue testing at Battelle. An LMFBR-sponsored program of considerable scope on 2-1/4 Cr-1 Mo for steam generator application is in progress at ORNL.

Other primary-circuit structural materials of current interest (e.g., Hastelloys, low-carbon steels, and Inconels) are not being studied in great detail. The same is true for weldments.

No specific program directed to the development of design analysis methods for evaluation of HTGR component safety, nor validations of such methods, now exists. The rules and codes provided by Section III of the ASME Code and interpretations thereof (e.g., 1331-8) are applicable in many cases, but even these are in a transient and formative stage. However, structural design technology being

developed by ORNL for LMFBR systems should provide useful background and tools for use in HTGR evaluations.

13.8.3 Identified work

Technology status report (task 6.3.1.1)

The current status of the technology of HTGR metallic structural materials should be evaluated. This would include examination of the mechanical properties of alloys used in the primary system and in the steam generator and auxiliary cooling systems over temperature ranges consistent with normal operation and abnormal conditions. Consideration should be given to the effects of time and environment on properties for each material and application.

Mechanical properties and failure criteria for HTGR structural metals (task 6.3.1.2)

Mechanical property information needed to develop safety evaluations for HTGR primary circuit components should be obtained. The materials to be tested include those of direct, immediate interest to HTGR technology (i.e., Incoloy 800, Hastelloy X, 2-1/4 Cr-1 Mo steel, and 1010/1020 carbon steel) and others (e.g., Hastelloy S and Inconel 617) currently believed to have significant potential for application. Study of weldments should include Incoloy 800 to Incoloy 800 with both Inconel 82 and 182 filler metal, Incoloy 800 to 2-1/4 Cr-1 Mo steel with Inconel 82 filler metal, 2-1/4 Cr-1 Mo steel to low carbon steel, and Hastelloy X to Hastelloy X.

Basic mechanical property tests should include (where appropriate) conventional tensile, uniaxial creep and creep-rupture, stress relaxation, and cyclic stress-strain. The temperature range investigated may be different for each material (e.g., 900 to 1800°F for Incoloy 800, 600 to 1100°F for 1010/1020 low carbon steel) but should encompass those temperatures expected both during normal operation and under accident conditions. Tests should be performed in both air and helium to assess environmental effects. Work on the effects of helium impurities on mechanical properties should be closely coordinated with that described under task 3.2.1.2 (reaction of coolant impurities with HTGR structural materials).

Since material from any one heat or lot cannot be considered typical or representative of the "average" heat, a number of heats representative of the range expected in commercial practice should be screened on the basis of relatively short-time properties determined on both virgin material and aged specimens. More extensive testing

should be performed on a selected number of these heats (e.g., creep and stress-rupture to 1000 hr or greater in both the aged and non-aged condition). A heat judged to be average should be tested in great detail and for long periods of time (more than 10,000 hr) in creep. A similar philosophy, but of necessarily different detail, should be employed in the testing of weldments.

Finally, complex tests involving load, strain, and temperature cycling should be performed to permit evaluation of (1) creep-fatigue interactions, (2) cumulative damage effects, and (3) material behavior under conditions expected during postulated accidents.

Design analysis methods for HTGR safety evaluation (task 6.3.2.1)

Mathematical models of material behavior under varying multiaxial loading and variable thermal conditions should be formulated. These models will be designed to permit safety analyses under both normal and accident conditions.

Validation of design analysis methods (simple geometries) (task 6.3.3.1)

Tests designed to simulate the thermal, mechanical, and environmental history of HTGR primary circuit components should be performed on relatively simple structures to validate the analysis capability developed under task 6.3.2.1. The conditions (e.g., stress state) imposed in any given test should be based on those expected for similar structures in the reactor.

Design analysis and failure-mode validation (complex tests) (task 6.3.3.2)

The results of the validation experiments on simple structures (task 6.3.3.1) should be used to assess the need for larger, more complex experiments.

Examination of components from Peach Bottom HTGR (task 6.3.3.3)

Components representative of those used in large HTGRs should be removed from Peach Bottom and examined. These should include the steam generator tube bundle and support structure, the hot and cold ducts, thermal barriers, valve internals, and parts of the circulator. Various nondestructive examinations should be performed

on-site before and during removal of the components. Disassembly, sectioning, and examination should be made of all components removed. Specimens should be examined for evidence of reaction with impurities (task 3.2.2.1). Various mechanical property tests should be conducted on selected specimens to evaluate the consequences of service (i.e., the effects of any chemical reactions, thermal aging, etc.) on material behavior.

Validation of alternative steam generator materials (task 6.3.3.4)

The purpose of this task is to demonstrate the feasibility of using high alloy materials in the low temperature (evaporator or economizer) regions of an HTGR steam generator. Before this can be done, however, it must be demonstrated that (1) corrosion damage in the dryout region would be negligible in the event of accidental feedwater contamination, (2) weldments in the evaporator region would have adequate corrosion resistance, and (3) deposits normally accumulated in the evaporator region can be removed safely by standard chemical cleaning processes.

The above should be explored through a series of corrosion-loop and service tests. Tubular specimens of Incoloy 800 or other suitable materials (e.g., 9 Cr-1% Mo ferritic steel) should be exposed to conditions of boiling heat-transfer in the evaporator section of once-through dynamic corrosion loop. Representative conditions (in terms of flow, temperature, pressure, heat flux, and other variables) should be set in the loop, and all test specimens should be either prestressed or stressed in-place. Chloride and oxygen contaminants should be added to high-purity feedwater (pH 9.3, ppb impurities). Contaminant levels should be progressively increased until either specimen failure is observed or unreasonable levels are reached.

In the service tests, tubing of appropriate materials should be installed in the evaporator sections of operating steam generators with relevant water chemistry and heat transfer conditions. These tests, to demonstrate satisfactory service under actual operating conditions, should be continued for a minimum of 10,000 hr.

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14. SAFETY INSTRUMENTATION (TASK AREA 7)

14.1 DEFINITION OF TASK AREA AND OBJECTIVES

The subject of this task area is the development and testing of instruments and instrument systems required to assure the safety of HTGR plants. The task area includes instrumentation for surveillance, monitoring, or trouble diagnosis which aids in the safe operation of the plant, as well as instrumentation which is a part of the plant protection system used to detect potentially unsafe conditions and initiate appropriate protective action. Instrumentation in the first category can be related to the first level of safety¹ which requires design for a high level of safety in normal operation. This demands optimization of measurement, display, and control features so that the plant can operate with a low challenge rate to the plant protection system. Instrumentation identified as part of the plant protection system addressing the second level of safety concern receives the major emphasis in this task area. Instrumentation with accurate and timely response is necessary for the measurement of significant plant parameters if unavoidable plant upsets occur, so that appropriate action can be automatically initiated to restore the plant to a safe condition and to actuate engineered safeguards for the prevention of subsequent damage and release of radioactivity. A third level of safety concern is defined as the failure of some protection systems simultaneously with the occurrence of an accident. The role of instrumentation in these circumstances is not well defined, but studies are proposed to determine what instruments might be designed to remain operable under accident conditions. Overlap of these three areas of concern is unavoidable, and a strict categorization of instruments into the three areas is not possible.

14.2 SCOPE

The scope and specific objectives of this task area are largely limited to instrumentation problems that are unique to HTGR plants or which involve special adaptation for use in HTGRs because of the different environment or other conditions. Safety instrumentation that has become well developed for other reactor types as well as for HTGRs and that is suitable for use without special adaptation, is not included in this plan, although verification of suitability may be necessary. In some cases development needs may be cited which are also identifiable with other reactor types but which are nonetheless important for gas reactors. Such items are included in the plan, with the recognition that parallel or complementary

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efforts may be planned or under way as part of other reactor development programs. Instrumentation for surveillance of PCRVs is treated separately in Sect. 12.

14.3 STATUS AND NEEDS

Brief summaries of the current status and anticipated development needs for safety instrumentation are included as part of the specific tasks which have been identified. For the most part, all these areas involve instrumentation and techniques that are presently deemed to be adequate to assure plant safety but which are perhaps not optimum. Recent advances in technology may allow for substantial improvement in safety or reliability, and additional advances are believed to be attainable. While all the work explicitly identified in this section is closely related to safety, it is also important to satisfactory performance of the plant under conditions that do not involve the health and safety of personnel. Therefore it is identified with the engineering and technology development effort.

14.4 TASK AREA EFFORT AND SCHEDULE

Because of the variety of subjects treated in this task area, the identified work has been divided into nine task groups, each dealing with a particular problem class. Most of the tasks are presented as source tasks even though the work includes synthesis and validation efforts. Many of the subjects identified for consideration are also treated in the safety planning guide (Part II) as topics for which additional assessment is required of the technology needs as they apply to safety. The work that is discussed in this section relates to needs that can be identified on the basis of current understanding of the state of the technology. There is considerable overlap in the scope, justification, and status between the need assessment tasks (task group 7.0) and the technology tasks (task groups 7.1 through 7.8) as given in this section. Those portions are not repeated for task group 7.0; that information can be obtained from Sect. 8 and is also partly included in the scope, justification, and status subsections of the development tasks below. All of the tasks are listed in Table 14.1, which summarizes the estimated relative effort that is involved.

Task group 7.0 deals with safety instrumentation assessment. The task group has been divided into several separate assessment tasks associated with different disciplines and covers assessment of HTGR thermometry, in-core flux monitoring, surveillance and

Table 14.1. Summary of identified safety and safety-related work in task area 7

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 7.0 -- Safety instrumentation assessment										
7.0.1.1 HTGR thermometry. A study to determine thermometry needs for safety and the consequences and probabilities of thermocouple failure adversely affecting the safety status of an HTGR. Includes identifying needs for on-line surveillance.	2	2								
7.0.1.2 In-core flux monitoring. Assess HTGR safety needs for high-temperature in-core neutron detectors; this also involves assessing the state of the art of existing detectors.	0.5									
7.0.1.3 Surveillance and diagnostics by noise analysis and dynamic testing. Identify incipient failures that require early detection and process variables that might be monitored or dynamically perturbed for analyses related to adequate safety assessment.	1									
7.0.1.4 Reactor coolant moisture monitoring. A detailed study of safety requirements.	1	1								
7.0.1.5 Study to identify additional safety instrumentation needs. A comprehensive assessment of status and requirements of all instrumentation related to the safety of the plant.	3	2								
Task group 7.0 total	7.5	5								
Task group 7.1 -- Development of failed-fuel location	3	2	2	2	1	1	1			
7.1.1.1 FAT-P system development. Continue development of system based on flux and temperature peaking.										
7.1.1.2 Sniffer system development. Continue study of gas sampling scheme to detect activity from specific regions.										
7.1.1.3 Source-induced perturbation development. Evaluate use of Cf-252 source to perturb local core regions.										
7.1.1.4 Delayed-neutron triangulation. Investigate feasibility for HTGRs.										
Task group 7.1 total	3	2	2	2	1	1	1			
Task group 7.2 -- HTGR thermometry development	2	2	2	4	4	2	2			
7.2.1.1 Testing of sheathed Chromel/Alumel thermocouples. Test to 1100°C for times of more than 10,000 hr in fluxes to 10^{14} n/cm ² ; evaluate service life.										
7.2.1.2 Qualification of improved thermocouples. Test Geminol, Nicrosil/Nisil, and Platinel under severe conditions.										

Table 14.1 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
7.2.1.3 Development of advanced temperature sensors. Develop single-point devices and sensors using gas as the working substance.										
7.2.1.4 Methods for assuring accurate and reliable thermometry. Develop methods for on-line testing, surveillance, diagnosis, and restoration.										
Task group 7.2 total	2	2	2	4	4	2	2			
Task group 7.3 — Development of in-core flux monitoring	2	3	2	1	1					
7.3.1.1 Testing of in-core flux monitors. Test detectors in simulated HTGR environments.										
7.3.1.2 Application of in-core flux monitors. Observe performance in operating systems.										
7.3.1.3 Development of in-core flux monitors. Extend current technology for fission chambers and self-powered detectors.										
Task group 7.3 total	2	3	2	1	1					
Task group 7.4 — Development of surveillance and diagnostics by noise analysis and dynamic testing	2	3	3	3	2	1	1	1		
7.4.1.1 Noise-analysis methods development. Feasibility studies followed by in-plant tests to demonstrate applicability to HTGRs.										
7.4.1.2 Dynamic-testing method development. Identify subsystems to be monitored and perform in-plant tests.										
7.4.1.3 Overall surveillance and diagnostic system design. Design a system to be integrated into HTGR plants.										
Task group 7.4 total	2	3	3	3	2	1	1	1		
Task group 7.5 — Development of reactor coolant moisture monitoring	2	3	1							
7.5.1.1 Evaluation of current capability. Compile existing information and identify other potentially applicable methods.										
7.5.1.2 Qualification of monitors for HTGRs. Develop systems that offer greatest promise.										
Task group 7.5 total	2	3	1							
Task group 7.6 — Development of data acquisition and processing (DAP) systems	1	2	2	1						
7.6.1.1 Identification of the status and completeness of a DAP system. Establish criteria for evaluating the system in the context of safety.										

Table 14.1 (continued)

Task description	Relative effort by fiscal year (man-years)									
	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Task group 7.6 total	1	2	2	1						
Task group 7.7 - Development of post-accident monitoring systems	1	2	2							
7.7.1.1 Range of accidents for consideration. Determine accidents and resultant conditions for monitoring.										
7.7.1.2 Identification of needed information. Define plant status data required for safety at all stages of accidents.										
7.7.1.3 Identify measures to obtain information. Define measurements to be made and development needs to ensure availability of data.										
Task group 7.7 total	1	2	2							
Task group 7.8 - Development of plant controllability	1	2	2	2						
7.8.1.1 Identify desirable plant control features. Identify features needed for power-plant control.										
7.8.1.2 Establish reasonable control schemes. Determine feasible and reasonable modes of control.										
7.8.1.3 Investigate effects on plant safety. Relate desired operating methods to safety.										
7.8.1.4 Investigate changes required to protection system. Determine needs to effect improvements.										
7.8.1.5 Develop performance criteria. Evolve criteria for optimum plant control.										
Task group 7.8 total	1	2	2	2						
Task area 7 total	21.5	24	16	13	8	4	4	1		

diagnostics by noise analysis and dynamic testing, reactor coolant moisture monitoring, and additional studies as needed relative to the instrumentation requirements for HTGR plant safety.

Task group 7.1 deals with development of methods for locating failed fuel in HTGRs. Existing techniques are considered adequate for locating relatively large failures but lack sensitivity for the timely and accurate location of small quantities of failed fuel. The development effort, as identified here, has two principal aspects - the refinement of the existing techniques to improve sensitivity and accuracy and the investigation of different concepts.

Task group 7.2 covers reactor temperature measurements, which has an important bearing on reactor safety. While thermocouples are available for use in the specific chemical, thermal, and radiation conditions of HTGRs, there are insufficient data to provide quantitative predictions of failure rate. Initial efforts could obtain reliability information, identify critical failure mechanisms, and establish quality assurance needs for immediate applications. Concurrently, investigative and development work could begin on improved thermocouples and advanced sensors suitable for the special conditions of HTGRs. On-line methods could be developed for determining remotely the accuracy and reliability status of temperature sensors in HTGRs and for diagnosing or improving their condition.

Task group 7.3 concerns development of in-core flux monitoring. Commercially available rhodium self-powered neutron detectors are planned for use in HTGRs. Additional qualification in the specific environments expected is necessary to assure adequate reliability for high-power applications. High-temperature fission chambers are highly desirable for use in gas-cooled reactors, and some development is underway in the LMFBR program. A cooperative effort is envisioned for this common need of all reactor types.

Task group 7.4 deals with development of safety surveillance and diagnosis by noise analysis and dynamic testing. These techniques are to be used to determine system operational characteristics, to detect incipient failures, and to characterize anomalous behavior and failures of plant components. Similar techniques are being applied to other reactor types and mechanical equipment and offer the advantage that only regularly available plant hardware is needed. Early development is desirable to have equipment ready during the initial phases of plant design and operation, so that characterization and qualification can be achieved as equipment becomes operable.

Task group 7.5 deals with development of equipment for reactor-coolant moisture monitoring. The presence of moisture in the coolant gas would indicate steam in-leakage from the heat exchangers, and reliable monitoring systems are needed. Acceptable systems have been used on early gas reactors, but none of these instruments will operate in large HTGR systems without modifications such as heavy-wall construction for pressure capability. All of the available systems have some disadvantages such as low reliability, complexity requiring high levels of maintenance, marginal response, or limited sensitivity which make further development highly desirable. An experimental program to investigate different sensors for high moisture levels probably should receive early attention because improved systems of this type will benefit all HTGRs.

14.5 SAFETY INSTRUMENTATION ASSESSMENT (TASK GROUP 7.0)

14.5.1 Scope, justification, and status

This task group provides for assessment of the instrumentation requirements for HTGRs from the viewpoint of safety needs. The assessment includes identifying or establishing plant requirements for thermometry, in-core flux monitoring, incipient failure detection, coolant moisture monitoring, and other instrumentation areas as needed. For more detailed information concerning this subsection, please refer to task group 7.0 in Sect. 8 where the scope, justification, and status are discussed specifically for each of the tasks in this task group. Related information is also given in the development task groups below.

14.5.2 Identified work

Assessment of HTGR thermometry (task 7.0.1.1)

The temperatures in all locations within the containment vessel for large HTGR plants should be estimated from prior experience with Peach Bottom and from calculations on models of the reactor system. The effects on safety of incorrect measurement of those temperatures should be assessed for normal, startup, shutdown and other transient conditions. The ability of existing and alternative temperature sensors to provide accurate measurements for prolonged periods — 10,000 to 100,000 hr — should be surveyed. Where information is not available and is needed for safety assessment, a plan for its acquisition should be formulated. On-line surveillance methods for determining thermocouple performance should be evaluated relative to the influence that thermocouple surveillance can have on plant safety.

Assessment of in-core flux monitoring (task 7.0.1.2)

Assessment should be made on HTGR requirements for in-core measurements. This includes determining the state of the art of self-powered neutron detectors and high-sensitivity fission chambers for high-temperature applications typical of HTGR needs. The study should include the characterization of the HTGR environments and operating modes in which the detectors must operate and should define the safety performance criteria of both types of detectors. Also included should be evaluation of necessary positioning devices. This study would be expected to define any safety development requirements for in-core neutron detectors.

Assessment of surveillance and diagnostics by noise analysis and dynamic testing (task 7.0.1.3)

A study of typical HTGR plants should be made to determine what types of failures are most important in terms of early detection and which process variables might be monitored, along with their frequency ranges of interest for noise analysis. Further, an assessment should be made as to which subsystems can be monitored by dynamic testing to give useful safety information and how the tests can be implemented.

Assessment of reactor coolant moisture monitoring (task 7.0.1.4)

Assessments should be made to determine the detailed moisture-monitoring requirements of HTGR plants from the viewpoint of safety; this includes assessing the present state of the art with respect to moisture monitoring capability. These studies should lead to a report which (1) identifies moisture-monitoring requirements, (2) compiles the Peach Bottom experience with moisture monitors, both operational and developmental, (3) identifies commercial systems and discusses their applicability to HTGRs, (4) describes foreign work, (5) discusses other U.S. research and development activity, and (6) identifies and discusses other concepts that have potential applicability but have not been investigated.

Study to identify additional safety instrumentation needs (task 7.0.1.5)

The principal effort of this task should be a comprehensive review of the status and requirements of all instrumentation

related to the safety of the plant so that additional safety-development needs can be identified. This may include further investigation of other specifically proposed tasks as well as a search to reveal unidentified problems.

14.6 DEVELOPMENT OF FAILED-FUEL LOCATION (TASK GROUP 7.1)

14.6.1 Scope and justification

The failed-fuel location system provides the means for determining the location of such material within the reactor system. This system is to be available, on call, for use when significant fuel failure has been detected. The ability to locate a particular region of fuel failure in the presence of other failed fuel is necessary since operation with some failed fuel is anticipated. Failed fuel constitutes an immediate source of primary loop contamination and a potential source of fission products that could enter the primary coolant rapidly in a postulated accident in which the core temperature rises abnormally. Knowledge of the location of failed fuel is necessary to plan for change-out. Accordingly, the object of task group 7.1 is to develop a reliable and accurate method for locating failed fuel.

14.6.2 Status

Currently there are two methods under study at GAC for identifying the region in an HTGR core that contains failed fuel. The current reference method is to use a combination of flux tilting and orificing of individual refueling regions (flux and temperature peaking method or FAT-P). This procedure raises the power in several regions by a special rod withdrawal pattern and raises the temperature of one of these regions by adjusting the orifice of that region. By monitoring the circulating activity as the power and temperature in each region is raised, a relative measure of the fraction of failed fuel in the regions throughout the core can be made. The major advantage of the FAT-P method is that no additional equipment or reactor modifications will be required. Disadvantages are the poor sensitivity for locating a limited amount of failed fuel or evenly distributed failures and the increased release of fission product necessary to accomplish failure location.

One alternative method under study is the use of an on-line "sniffer" system that would sample the coolant activity at the exit of each refueling region. Comparison of these activity concentration values would give a relative measure of the fraction of failed fuel in the

regions throughout the core. This technique could possibly be incorporated using a thermocouple sleeve present in each region. If this is not feasible, difficult mechanical-design problems may arise in providing "sniffer" access to each region. However, compared with the FAT-P method, the advantage of the sniffer system is potentially improved detection sensitivity, which might justify the added mechanical-design complexity.

Other methods have been proposed for the HTGR, such as introducing a ^{252}Cf neutron source to suspect regions of the core during shut-down and measuring the fission product release to the coolant. Tests of this concept are proposed in the Peach Bottom end-of-life program (see Appendix B). Consideration of LMFBR technology yields only the delayed-neutron triangulation technique which is under development at ORNL as a possibility.²

14.6.3 Identified work

The feasibility of identifying and locating excessive fission-product leakage from fuel elements in situ in the reactor core could be investigated along several routes. These presently include the FAT-P method, "sniffers," delayed-neutron detection/triangulation, and local core perturbations with a strong neutron source. Associated with exploration of these methods is the need to characterize HTGR failed-fuel behavior.

A study to characterize the behavior of failed fuel is identified as part of the effort in task area 2. This study would provide the expected release rates of fission products to the coolant for normal and off-normal operations of the plant. This information is vital to the feasibility and optimization studies of the various methods under consideration.

FAT-P system development (task 7.1.1.1)

Continued development of the FAT-P method is suggested. Measurement procedures could be developed consistent with available plant hardware and plant operational limitations. Analytical studies would be used to optimize techniques, and in-plant testing would be performed to define the locational accuracy and sensitivity of this method.

Sniffer system development (task 7.1.1.2)

Evaluation of the sniffer concept could proceed. The locational accuracy and sensitivity should be defined, and the mechanical design requirements identified. If the method proves desirable, then in-plant testing could be performed.

Source-induced perturbation development (task 7.1.1.3)

The concept of using a ^{252}Cf neutron source to perturb local core regions could be evaluated and the size of source required, source insertion schemes, and instrument requirements studied. If further development appeared warranted on the basis of the Peach Bottom tests, additional in-plant tests would be proposed to further evaluate this concept.

Delayed-neutron triangulation (task 7.1.1.4)

A study will be made to determine the feasibility of extrapolating LMFBR development results for this method to HTGR application. If further development appears warranted, then studies for optimizing the method could be performed and in-plant tests be conducted.

14.7 HTGR THERMOMETRY DEVELOPMENT (TASK GROUP 7.2)

14.7.1 Scope and justification

Designs of large HTGR plants^{3,4} incorporate sheathed thermocouples at the bottom of the core to measure core-exit-gas temperatures. Important reactor safety and control information is obtained from these thermocouples under both normal and accident conditions. Excessive core temperatures, which could damage fuel and release fission products into the primary coolant system, are determined principally from core-exit-gas temperatures. Conditions deduced from these temperature measurements during operation are: (1) presence of radial or azimuthal xenon oscillations, (2) excessive fuel temperature, (3) effectiveness of rod programming, (4) power/flow mismatch, (5) flux tilts and power imbalances, and (6) adequacy of flow orificing. If core cooling were interrupted, these measurements would also provide the first direct indications of excessive helium temperatures upon restoration of primary flow. Presently, steam generator inlet temperature provide the first indication.

The accuracy and reliability of reactor thermocouples are adversely affected by (1) improper design or installation, (2) thermal aging, (3) chemical contamination, (4) thermal neutron transmutation, (5) fast neutron damage, and (6) in some cases prompt thermal disequilibrium due to gamma or neutron irradiation. Calibration drifts, erratic output, and short service life may be avoided and safe, efficient reactor operation assured if the mocouples of high quality and known reliability are used. Improved thermometers and the use of on-line surveillance can extend the capability of existing thermocouples to provide instrumentation for recovery from off-normal operation.

This task group includes (1) a parametric determination of the failure rates of high-quality nuclear-grade Chromel/Alumel thermocouples under HTGR conditions; (2) failure rate determination of possible substitute thermocouple materials, such as Geminci and Nicrosil/Nisil, for higher temperatures and conditions detrimental to Chromel/Alumel; (3) study of methods for on-line surveillance and restoration of failed core-exit-gas thermometers; and (4) application of alternative temperature sensor systems, such as resistance, fluidic, pneumatic, or acoustic thermometers, to significantly increase the reliability of this temperature measurement. Effects of design, fabrication, and installation of the sensors on their reliability should be determined. Application of improved, high-reliability thermometers to in-core fuel and graphite temperature measurements can be examined. Development of reliable in-core thermometers would permit closer control of reactor operating conditions and improve the interpretation of experiments relating to fuel performance and fission product transport.

14.7.2 Status

There are not sufficient specific data to provide a quantitative prediction of the failure rates of sheathed thermocouples in reactors under the chemical, thermal, and radiation conditions for the HTGR. Mean-time-between-failure (MTBF) data are needed for design and for replacement scheduling of HTGR core exit thermocouples. Some pessimistic failure rates for Chromel/Alumel thermocouples have been obtained⁵ from capsule irradiations. The LMFBR thermometry program at ORNL has performed extensive out-of-pile tests on sheathed Chromel/Alumel thermocouples at temperatures comparable to those expected in the HTGR. In-pile tests on high-reliability sheathed Chromel/Alumel thermocouples have been made in a carbon-free environment, in a fast-neutron flux, and at lower temperatures (1200°F or 650°C) than the HTGR requires but showed only very

slight failure.⁶ Current testing in Peach Bottom and HFIR fuel-capsule irradiations and in the Fort St. Vrain reactor will provide some indications of thermocouple failure patterns and mechanisms.

Other base-metal alloys, such as Geminol and Nicrosil/Nisil,⁷ offer possibilities of higher temperature service than Chromel/Alumel permits but have not been tested under reactor conditions. The relative susceptibility of various thermocouple materials to carbon contamination also needs investigation.

The condition of a thermocouple during service can be determined by application of on-line surveillance methods being developed by the LMFBR thermometry program.⁸ Thermocouple reliability can be improved and imminent failure predicted by external measurements of the thermocouple's time constant, loop resistance, or insulation resistance. Other methods for diagnosis and location of failure can be used to determine whether temperature channel failures occurred in the sensor, leads, or external instrumentation. A method⁹ for restoring open-circuit failed thermocouples is currently being evaluated for reactor use.

A method has been suggested by the ORNL thermometry studies⁶ to "anneal out" drift in thermocouple output caused by fast neutron irradiation damage and thus to restore its calibration. This task should explore annealing methods, extend their use to HTGR thermocouple problems, reduce the methods to practice, and reconcile their use as safety functions with the normal reactor control function requirements.

Other temperature sensing systems have been proposed which would avoid some of the inherent limitations of thermocouples for gas exit temperature measurement. Fluidic thermometers, which sense gas temperature directly, were developed for UHTREX.¹⁰ Pneumatic probes were considered for nuclear engines.¹¹ Ultrasonic thermometry techniques have been studied¹² which use either solid temperature-sensitive delay lines or the sound-propagation characteristics of gases. These methods were applied to DRAGON project and HTGR fuel studies.¹³ Resistance thermometers have been tested and specified¹⁴ for certain out-of-core locations in the LMFBR program where their higher sensitivity and stability can be used to provide more accurate measurements than are possible using thermocouples for overall heat-balance determination at temperatures up to 660°C (1220°F). Johnson noise thermometers (JNT) have been investigated recently in NASA and AEC¹⁵ programs and evaluated in the European reactor program¹⁶ as absolute thermometers for nuclear reactors.

If successfully applied, the JNT would provide HTGR temperature measurements free of drift, aging, or transmutation effects. In-pile tests are scheduled as part of the HTGR fuel studies.¹³

14.7.3 Identified work

Development of core-exit-gas thermometers to meet HTGR needs with testing in appropriate thermal, chemical, and radiation environments is suggested. Tests would obtain reliability information, identify critical failure mechanisms, provide design and installation guidance, and establish quality assurance needs for ensuring thermometer integrity. Improved thermocouples and advanced sensors for measuring temperatures of coolant gas, core graphite, and fuel could be developed, tested, and evaluated. On-line methods could be developed for verifying sensor integrity, anticipating failure, detecting faults, and restoring calibration to extend thermometer reliability beyond conventional passive acceptance limits.

Testing of sheathed Chromel/Alumel thermocouples (task 7.2.1.1)

Reactor-grade, RDT-qualified,¹⁷ 1/8-in. and 1/16-in. OD stainless steel sheathed, MgO-insulated, Chromel/Alumel thermocouples are identified for testing to temperatures as high as 1100°C for times longer than 10,000 hr in inert and carbon-containing atmospheres and in neutron fluxes up to 10^{14} n/cm²-sec. Post-test and post-irradiation examination could be used to determine causes of thermocouple failures. A parametric MTBF study should be systematically followed to determine effects of sheath size, junction style, and external conditions on the failure rate. These thermocouples should be evaluated for their capability of exceeding a 100,000-hr service life.

Qualification of improved thermocouples (task 7.2.1.2)

Alternative base- and precious-metal thermocouples, such as Geminol, Microsil/Nisil, and Platinel, should be tested and evaluated to determine whether or not they will provide improved performance at HTGR temperatures from 850 to 1200°C in the presence of carbon-containing atmospheres and neutron radiation. Noble- and refractory-metal thermocouples, using elements with low neutron cross sections, should be evaluated for gas- and fuel-temperature measurements up to 1800°C. Alternative sheath materials and insulators should be tested to provide better chemical protection for the thermocouple elements and to minimize wire stresses due to differential thermal expansion and radial temperature differentials. The improved thermocouple

systems should be tested for long-term, high-temperature drift and for prompt and cumulative irradiation effects, and their reliability (MTBF) compared quantitatively with existing sheathed Chromel/Alumel thermocouples.

Development of advanced temperature sensors (task 7.2.1.3)

Combinations of thermocouples and resistance thermometers measuring a single temperature point should be designed for HTGR gas-exit application and tested to determine how much this combination can increase the reliability and accuracy of a given temperature measurement. Alternative temperature-measuring sensors that use gas as a working substance, such as fluidic, pneumatic, or acoustic thermometers, developed for other applications, should be tested in furnaces and in reactors to HTGR conditions and their reliability compared with that of thermocouples. Prompt radiation effects and application problems could also be investigated. Other advanced sensors, such as the solid delay line and resonant ultrasonic thermometers and Johnson noise thermometers, should be evaluated for core-gas-exit use in place of or in addition to thermocouples for improved reliability and accuracy and for the more severe in-core graphite- and fuel-temperature measurements. Optical, thermal radiation, and other non-contact temperature measurement methods should be evaluated for HTGR core-exit measurements. Accuracy, drift, and reliability data should be obtained for any advanced sensor that proves to be a practical candidate for HTGR plant use.

Methods for assuring accurate and reliable thermometry (task 7.2.1.4)

Methods of on-line surveillance of thermocouples (and other thermometers) in the HTGR should be developed for increasing the accuracy and reliability of temperature measurements in HTGRs. These methods should be tested and evaluated for their capability of increasing reliability and extending useful life of thermocouples. Some possible methods to be studied are presented below.

Methods of performance verification. Methods could be developed for verifying continuity, calibration, emplacement, and time response of thermocouples in situ by using such techniques as comparison with insertable standards, updating of calibrations based on isothermal conditions, and measurement of the transient response of the thermocouples to temperature fluctuations or loop current heating.

Methods of imminent failure anticipation. Methods for detecting the progressive deterioration of a thermocouple preceding its catastrophic failure could be studied: continual monitoring of thermocouple loop-resistance changes, monitoring of loop-to-ground resistance for isolated junction thermocouples, and routine analysis of thermal noise signals from thermocouples.

Methods for failure diagnosis. Methods should be developed for determining the cause and location of failures in thermocouples so that the faults can be remedied or the thermocouple replaced. These methods include measuring resistance of shorted loops and capacitance of open loops, using time-domain reflectometry techniques for locating electrical impedance discontinuities, and interchanging sensors and external circuits to isolate faults. A method for in situ detection of thermoelectric inhomogeneity produced by radiation damage should be investigated.

Methods for restoration of failed thermocouples. Thermocouples whose calibration has drifted beyond acceptable limits or that show intermittent operation may be restored to useful service by application of methods developed in this activity. These methods include: in situ annealing of fast-neutron irradiation defects in thermocouple wires by the application of electrical heating of the thermocouple wires, loop-to-ground short-circuit clearing by application of burn-out currents, and reestablishing electrical continuity by application of large current pulses (remote junction rewelding or "zapping"). These methods should be tested, and the reliability of thermocouples returned to service by these methods determined.

Methods for design and installation practices to increase reliability. Design alternatives, such as the use of multiwire thermocouples, or of combinations of thermocouple and resistance thermometers in a single sheath, should be studied. Emplacement and routing criteria for thermocouples should be studied for maximizing thermocouple reliability. Reactor control and safety system requirements should be considered, and any on-line surveillance methods derived from this study adapted to meet reactor requirements. The end objective would be to provide compatible, continual, unattended thermocouple monitoring to increase accuracy and reliability.

14.8 DEVELOPMENT OF IN-CORE FLUX MONITORING (TASK GROUP 7.3)

14.8.1 Scope and justification

Since proposed HTGR cores are very large, the out-of-core flux measurements do not accurately monitor local flux oscillations or peaking. In-core flux measurement capability at power, supplementing thermometry, would reduce the opportunity for undetected local overheating and fuel damage. Furthermore, the out-of-core measurements are not sufficiently sensitive for monitoring reactor status during refueling, maintenance, and startup. High-temperature in-core neutron detectors for flux measurements in the source range could afford additional safety by allowing the monitoring of reactor status during startup. At present, shutdown is monitored by inserting a chamber after the core has cooled down.

Accordingly, this task group is aimed at extending current investigations of self-powered neutron detectors for in-core power-range neutron flux mapping. In addition, high-temperature fission chambers for startup-range application in high-temperature locations are identified for investigation.

14.8.2 Status

Self-powered (rhodium) neutron detectors are available commercially for moderate temperature applications.¹⁸ High-temperature fission chambers are being developed for LMFBRs at ORNL and GE; they are planned for service at 1050°F and 10^6 to 10^7 roentgen/hr.^{19,20}

14.8.3 Identified work

Additional development and testing should be directed toward qualification of both the self-powered and the high-sensitivity neutron detectors for the HTGR operating environment.

Testing of in-core flux monitors (task 7.3.1.1)

Testing of the detectors should be performed in simulated HTGR environments in the laboratory and then in actual reactor conditions.

Application of in-core flux monitors (task 7.3.1.2)

The application of both of these detector systems could be effected through an integrated effort between the detector developers and the reactor designers. Detection systems should be designed that meet the monitoring needs of and are compatible with the reactor system design.

Development of in-core flux monitors (task 7.3.1.3)

Any development required to extend current technology to HTGR applications should be performed. This is not expected to be extensive for high-sensitivity fission chambers but may be more difficult for self-powered detectors.

14.9 DEVELOPMENT OF SURVEILLANCE AND DIAGNOSTICS BY NOISE ANALYSIS AND DYNAMIC TESTING (TASK GROUP 7.4)

14.9.1 Scope and justification

Advanced surveillance and diagnostic systems could improve plant safety, reliability, operability, and efficiency. Noise analysis and dynamic testing methods are emerging technologies that offer the opportunity of advancing considerably the reactor surveillance and diagnostics state of the art. These techniques are used to determine the system operational characteristics, to detect incipient failures, and to characterize anomalous behavior and failures of plant components while using only regularly available plant hardware.

Characteristic noise patterns ("signatures") are associated with the operation of a reactor and its heat removal equipment. Changes in these patterns signify developing wear of rotating machinery, weakening of vibrating parts, or changes in operating characteristics. Analysis of noise can and is being done on a variety of reactor plant process signals including temperature, flow, pressure, acoustics, and neutron flux.

Dynamic testing in reactors is used for stability assessment, model verification, and identification of design parameters. It can be used in many subsystems of the reactor plant: reactor core, heat exchanger, steam generator, pressurizer, etc. Recent dynamic testing advancements have been achieved by going to small signal perturbations ($\sim 1\%$) and two-level input signals which then allow on-line measurements using existing hardware and without interrupting normal operation.

14.9.2 Status

Considerable use of noise analysis methods has been made in non-nuclear systems, particularly in the aircraft industry. Nuclear applications have been under study for the past decade, and many different applications have been reported.^{21,22} However, a comprehensive system approach has not been taken in any of the investigations to date, and the proper implementation of the technique will not be effected until this approach is taken.

Small signal dynamic testing has been applied to several research reactors²³ and to several operating pressurized-water²⁴ and heavy-water²⁵ reactors recently. As in the case of noise analysis, the full benefit of this technique has yet to be realized because of a lack of systematic approach as part of the reactor operations.

14.9.3 Identified work

Development of surveillance and diagnostics systems for HTGRs based on noise analysis and dynamic testing methods employing regular plant hardware should be performed. Trial application of these methods should be made in operating plants or on equipment test stands to aid in designing the overall surveillance and diagnostic systems.

Noise analysis methods development (task 7.4.1.1)

A study of the overall plant should be made to determine the process variables to be monitored and their frequency ranges of interest. In-plant tests should be made to define plant hardware capabilities, normal background signatures, and signatures for anomalous behavior and component failures.

Dynamic testing method development (task 7.4.1.2)

A study is needed to determine which plant subsystems can be monitored by dynamic testing and how the tests can be implemented. Dynamic models should be studied to define the test requirements, and actual tests performed in operating plants to demonstrate feasibility.

Overall surveillance and diagnostic system design (task 7.4.1.3)

An overall surveillance and diagnostic system should be designed as an integral part of a reactor design based on the development results of tasks 7.4.1.1 and 7.4.1.2. This system could be implemented in the operating plant for evaluation as an overall system.

14.10 DEVELOPMENT OF REACTOR COOLANT MOISTURE MONITORING (TASK GROUP 7.5)

14.10.1 Scope and justification

The identified potential source of moisture input to the helium coolant of the HTGR is primarily leakage from steam generator tubes. Such leakage is likely to be quantized into one of two categories: (1) pin hole leakage or (2) tube rupture or breakage. The consequences of a pin hole leak are long term in nature, so an adequate defense is the off-line helium-impurities monitoring system. However, a tube break must be detected in about 10 sec so that the defective steam generator can be isolated and depressurized to minimize the quantity of steam-ingress to the coolant. To fail to do this correctly could lead to extensive oxidation of the graphite and an increase in primary system pressure with possible actuation of the PCRV relief valves and consequent release of fission products to the containment.

It should be noted that the need for moisture-monitoring instrumentation as an input to the plant protection system is unique to gas-cooled reactors. Although there are commercially available moisture monitors employing a variety of principles, none fully satisfy the HTGR requirements. For this reason, the moisture systems installed in the Fort St. Vrain reactor are an improved version of a commercial design with the improvements being made by the reactor vendor. An experience base for evaluating the applicability of any given design to the HTGR situation does not exist except with the reactor vendor.

14.10.2 Status

Test data have been obtained at the Peach Bottom reactor. The instruments in this test are standard moisture monitors with the exception of the thermal conductivity system which is a prototype system assembled at GAC. None of these instruments will operate in the large HTGR system without modifications such as heavy-wall construction for pressure capability. The Peach Bottom operating

experience has yielded valuable information on the advantages and disadvantages of the various operating principles used in the design of the instruments. The systems producing acceptable data at Peach Bottom are: (1) dew point hygrometer, (2) infrared analyzer, and (3) thermal conductivity trip unit.

Table 14.2 gives the advantages and disadvantages of the system presently being considered.

Most of the systems tested are somewhat complicated (i.e., employ a significant number of component parts) and therefore need periodic maintenance. It is highly desirable to minimize the maintenance required, especially on the instruments associated with the plant protection functions where at least three redundant instruments are required for each measurement. It should also be noted that the vulnerability to common mode failure can be significantly reduced if the moisture monitors used in the plant protection system are not all identical. This may require that more than one basic detector principle be qualified.

Some consideration is being given to using moisture trip instruments only on the plant protection measurements because these instruments are usually less complicated and require less maintenance as well as simpler on-line testing procedures.

Three of the five systems shown on Table 14.2 were tested at Peach Bottom and appear to have potential for large HTGR instrument systems. Three other systems, not listed, were used either on the reactor itself or on the test setup and found to be unsatisfactory.

An electrolytic hygrometer (P₂O₅ cell) was used as a plant system. This method yields an accurate moisture measurement and rapid response. However, the life of the electrolytic cells was very limited (approximately five weeks). The aluminum oxide method proved unsatisfactory due to a change in the calibration and response time after exposure to the dry helium environment. An infrared system with a short light path was tested and found unstable due to temperature sensitivity and the high amplification required.

14.10.3 Identified work

Evaluation of current capability (task 7.5.1.1).

A detailed study is required to determine the present state of the art with respect to moisture monitoring capability, for HTGRs. This study could lead to a report that (1) compiles the Peach Bottom

Table 14.2. Operating characteristics of large HTGR moisture monitors

Instrument type	Successfully tested at Peach Bottom	Relative accuracy	Response time	Temperature sensitive ^a	Pressure sensitive ^b	Flow sensitive ^c	Radiation sensitive ^d	Sensitive to solid particulate
Dew point hygrometer (indicating and trip)	Yes	Very good	Very good in trip mode ^e	No	Yes	No	Unknown	Yes
Infrared analyzer (indicating and trip)	Yes	Good	Very good	No	Yes	Yes - effects response time	Unknown	Yes
Partial pressure detector (indicating and trip)	No	Unknown	Unknown	Unknown	Unknown	Yes - effects response time	No	No
Thermal conductivity probe (trip only)	Yes	NA	Very good	Yes	Unknown	Yes - may cause false trip	Unknown	Unknown
Vibrating reed probe (trip only)	No	NA	Unknown	Unknown	Unknown	Unknown	No	Unknown

^aSample gas temperature changes from room temperature to 200°F.

^bFor operating pressure changes from 13 to 750 psia.

^cFor flow changes from 50% to 150% of rated flow.

^dFor long term exposure to 500 rads/hr.

^eTwo separate instrument systems are required for indication and fast response.

experience²⁶ with moisture monitors, both operational and developmental, (2) identifies commercial systems and discusses their applicability to HTGRs, (3) describes foreign work such as that of Bennet and Conibear,²⁷ (4) discusses other U.S. research and development activity such as the work of Billeter,²⁸ and (5) identifies and discusses other concepts that have potential applicability but have not been investigated. Such a report is a logical necessity to establish a basis for independent judgment with respect to the adequacy of systems and to identify direction for new work.

Qualification of monitors for HTGRs (task 7.5.1.2)

Following the completion of task 7.5.1.1, judgment would be required as to whether commercial devices were adaptable to the HTGR. Should they not be (the fact that the Fort St. Vrain monitors are not commercial makes this a strong possibility), the two most promising of the non-commercial techniques should be pursued in a research and development program. This program would include direct qualification under HTGR conditions.

A program to concentrate on the development of improved sensors that will operate in the trip mode only, for high-level detection, is suggested for prompt implementation. A vibrating-reed sensor and a differential pressure unit activated by a constant-temperature dew-point leg appear to offer promise at this time.

14.11 DEVELOPMENT OF DATA ACQUISITION AND PROCESSING (DAP) SYSTEMS (TASK GROUP 7.6)

14.11.1 Scope and justification

The design philosophy that separates the plant protection system from all other systems serves also to encourage a DAP design concept that may not address important factors relating other operating systems to the overall safety and productivity of the plant. Three of these factors, central to accident prevention, are: (1) accurate and comprehensive process status available to the operator, (2) timely and reliable annunciation of parameter-limit violations, and (3) uninterrupted control of auxiliary process systems.

1. Current status^{29,30}

The HTGR DAP system centralizes the gathering and portrayal of plant operating information through the application of identical, redundant computer systems designed for high-quality information presentation and essentially failure-free operation (via the redundant backup system). The DAP system displays and printout provide the sole annunciation of plant margin violations, and the plant operator is completely dependent on the displays and printout for such information as rod position and core-region orifice position and outlet temperature.

14.11.3 Identified work

Identification of the status and completeness of a DAP system (task 7.6.1.1)

This task involves the establishment of criteria for evaluating an HTGR DAP system in the context of safety and a study of the potential for such a system to give plant-condition diagnoses and guidance. The consequences of sensor or DAP-component inaccuracy or failure in terms of possible misleading information furnished the operator, which might cause him to make decision that would compromise the safety or productivity of the plant, should be studied.

14.12 DEVELOPMENT OF POST-ACCIDENT MONITORING SYSTEMS (TASK GROUP 7.7)

14.12.1 Scope and justification

After a major reactor accident, it is quite likely that many of the normal instruments would be unable to adequately monitor the status of the plant. Instruments could be damaged by the accident, the plant could experience conditions so far from normal that instruments could be off-scale, and many important readout instruments might be in inaccessible locations. Yet it is imperative in the minutes and hours following such an event that operating personnel have access to timely and correct information regarding plant conditions that normally receive little attention. Because of the magnitude and complexity of a modern reactor and power plant and the almost infinite number of courses an accident may take, it is necessary that the operating crew act decisively and correctly to minimize further damage and to limit the release of radioactivity to the environment. This action involves realignment of engineered

safety features, optimization of energy source utilization, and maybe even rescue and recovery operations. All such activity depends very strongly upon the availability of and confidence in operating instruments.

14.12.2 Current status

Although most postulated accidents are described in great detail in the plant's safety analysis report, it is fairly safe to assume that a major reactor accident will result in conditions never before seen by the operator and possibly not anticipated by the designer. To expect anyone to anticipate the innumerable combinations of circumstances that can occur is not reasonable, and even less reasonable is the expectation that proper operation of accident-mitigating equipment to occur automatically over the long term. There is currently under development an IEEE Standard and an AEC Regulatory Guide to address the requirements for instrumentation to assess water-reactor power plant conditions during and following an accident. A guide is under development for HTGRs.

O'Brien³¹ has reported on post-accident instrumentation with particular emphasis on rescue and recovery operations, with extensive references. Robinson and Duggins³² have reported a study of the monitoring problem in light-water reactors following a loss-of-coolant accident.

14.12.3 Identified work

The purpose of this task should be to develop a coherent and reliable system of instrumentation dedicated to the accumulation and display of information relative to the status of the plant and its vital auxiliaries following a major reactor accident. Of particular importance is information necessary for the alignment of defenses and utilization of resources to limit release of radioactivity. It is vital that such a system survive the accident and continue to operate for the duration of the emergency. Its readout should be available to personnel at a habitable (or at least temporarily accessible) location and preferably at a central control station. Very little work is being done in this area for any reactor type, and the proper priority is difficult to assign. The tasks to be accomplished fall into four broad categories as defined below.

Range of accidents for consideration (task 7.7.1.1)

A suitable range of accidents for consideration should be determined. These may include, but are not limited to, credible reactivity accidents, depressurization accidents, steam generator failure, fire, and the like.

Identification of needed information (task 7.7.1.2)

The basic bits of information relative to plant status during all stages of the identified accidents should be defined. These may include such items as reactivity status of the core, coolant-flow blockage, temperature and radioactivity distributions, availability of standby equipment, reserves of fuel and coolant, status of containment closures and boundaries, etc.

Measurements needed to obtain information (task 7.7.1.3)

Appropriate measurements needed to obtain information identified in task 7.7.1.2 should be defined. This task will include identification of the influence of the accident upon the measuring system as well as identification of development tasks that must be undertaken to assure availability of components that can withstand such influence.

14.13 DEVELOPMENT OF PLANT CONTROLLABILITY (TASK GROUP 7.8)

14.13.1 Scope and justification

Although current HTGR designs include compatible control and protection systems, a study would be desirable of the influence of improved controllability on overall safety and specific protection-system requirements. The purpose of this task is to determine those plant control features that should be included in HTGR systems for optimizing the utility of those reactors and to investigate the effect such features could have on plant safety and plant safety system requirements.

14.13.2 Identified work

Desirable plant control features (task 7.8.1.1)

By studying utility requirement and operating experience with nuclear power plants, those features of an HTGR control system that are most desirable for power plant operation should be identified.

Control schemes (task 7.8.1.2)

Experience with nuclear reactor control systems, both in power plant and engineering, test, and research reactors, should be studied to establish what is feasible and reasonable in HTGR plant control.

Effects on plant safety (task 7.8.1.3)

The direct effect of desired operating methods on plant safety should be investigated.

Changes required in the protection system (task 7.8.1.4)

Changes in protection system requirements that would be brought about by improved control methods should be investigated.

Performance criteria (task 7.8.1.5)

Performance criteria for an optimum plant control system for HTGRs should be developed, and the effects of such a system on plant safety documented.

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Appendix A

LETTERS RELATED TO THE NEED FOR AN HTGR SAFETY PROGRAM

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Dec 15, 1972

COPY

Milton Shaw, Director, Division of Reactor Development and Technology

HTGR SAFETY RESEARCH - REG:RSR-74

In our continuing review and evaluation of HTGR nuclear steam supply systems, we have identified several areas where additional research and development will be needed for safe design, construction, and operation as well as to provide an adequate basis for our safety assessment of HTGR's and their engineered safety systems. As you know, a listing of these safety issues and related research and development areas was provided to Gulf General Atomic Company in the enclosed letter from Mr. A. Giambusso, Deputy Director for Reactor Projects, L, to Mr. John Landis, President, GGA, dated December 4, 1972. It is expected that GGA will accommodate these identified safety R&D needs as well as others that may be identified. Regulatory will, of course, independently assess the results of these efforts relative to their fulfillment of our requirements. In this regard, we will have need for assistance in order to implement our assessment of certain areas.

For your information, and to prevent possible duplication of effort, enclosed is a summary of our current technical assistance projects to provide increased independent assessment capabilities as well as our understanding of the RDT funded programs specifically related to HTGR safety R&D. These efforts are providing valuable assistance to us and should be continued.

It should be recognized that we have not had access to technical details of large (1100 MWe) HTGR designs and it is anticipated that additional safety research and development needs may be identified in our continuing review of new plants. Also, as noted in the letter enclosure to GGA, we have requested that they identify safety issues and related R&D areas to the extent that they differ from ours. Such identification may lead to additional needed AEC funded R&D efforts to provide us with the required independent assessment capabilities to evaluate such areas.

As Regulatory identifies new HTGR safety research and development needs, we will make these needs known in order that they can be appropriately addressed in a timely manner. We would be happy to discuss our needs with your staff and provide additional specific details as may be required.

/s/ Ray G. Smith, for

Lester Rogers, Director
Directorate of Regulatory Standards

Enclosures - see next page

Milton Shaw

Dec 15, 1972 ^C_O_P_Y

Enclosures:

1. Letter, Giambusso to Landis dtd 12/4/72
2. Summary of REG Technical Assistance Contracts
and RDT Funded HTGR Safety Research Programs

cc: Manager, WRSPO
R. F. Fraley, ACRS (18)

bcc: E. J. Blech, DDR
S. H. Hanauer, DRTA
L. Rogers, RS
J. O'Leary, L
E. G. Case, L
J. Hendrie, L
F. Schroeder, L
A. Giambusso, L
R. Boyd, L
D. Knuth, L
R. Minogue, PS
R. G. Smith, RS
R. Clark, L
D. Eisenhut, L
J. Norberg, RS
A. Pressesky, RDT

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

copy

Dec 4, 1972

Mr. John Landis, President
Gulf General Atomic Company
P. O. Box 608
San Diego, California 92112

Dear Mr. Landis:

As a part of our preparation for performing the safety reviews related to the expected applications for permits to build and operate central power stations using large HTGR nuclear steam supply systems, we have reviewed the safety issues identified during the design and safety reviews of the Peach Bottom No. 1 and Fort St. Vrain HTGR's as well as those issues identified during discussions of the conceptual design of the large HTGR (1100 MWe). Based on this review, it appears that research and development will be needed in several areas in order to provide the information needed for safe design, construction and operation as well as to establish an adequate basis for review and evaluation of HTGR's and their engineered safety features.

The enclosed listing of design and safety issues and related research and development areas is provided for your consideration in developing information in support of specific HTGR applications that will be submitted for our review and evaluation.

The information may be included in PSARs or may be submitted as topical reports. The information should be provided in adequate detail so that our review can proceed without delay. In those cases where programs are underway or are planned to obtain information, but have not been completed, the research and development should be described in enough detail so that we can evaluate the adequacy of these programs as well as the schedule for their completion. Specific fallback or alternate positions, should the results of the R&D programs not be favorable or available in a timely manner, must also be identified.

The design details of a large HTGR plant have not been submitted for our review; consequently, the listing of design and safety issues and technical questions requiring further resolution either through R&D or

other optional approaches should not be considered complete and inclusive, but only as being representative of safety issues and the related research and development areas. In this regard it would be helpful to have your identification of safety issues and related R&D areas to the extent that they differ from ours.

By calling this matter to your attention at this time, we hope to facilitate the safety reviews of the pending large HTGR plants. If you have questions or desire to discuss these matters, please contact me.

Sincerely,

Original signed by:
A. Giambusso

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Enclosure:
List of Safety Issues and
Related HTGR R&D

HTGR RESEARCH AND DEVELOPMENT**I****Safety Issue: Reactor Pressure Vessel Integrity**

The safety of the HTGR is highly dependent on the integrity of the reactor vessel (PCRv) and its principal penetrations. Certain postulated failures of the reactor vessel could lead to unacceptable consequences.

RESEARCH AND DEVELOPMENT

1. The applicability and validity of the analytical methods, models, and computer codes used in the design of the PCRv must be established.
2. Safety margins in the design must be established.
3. Means for assuring the continuing integrity and functional reliability of the PCRv liner, thermal barrier, post-tensioning components, concrete, penetrations, and the liner cooling system must be provided.
4. The adequacy and reliability of the PCRv pressure relief system must be established.

II**Safety Issue: Integrity of Core Structure**

A significant safety consideration is the ability of the HTGR core structure to retain the functional capabilities of coolant plenums, coolant channels, control rod channels, reserve shutdown system channels and core supports during normal and accident conditions. Loss of all core cooling or loss of reactivity control could lead to unacceptable consequences.

RESEARCH AND DEVELOPMENT

1. Safety margins in the graphite and metals structures must be defined and the applicability and validity of material properties, load combinations, and allowable stresses established. Means for assuring the continuing integrity and functional reliability of core structures must be provided.

3. Adequate response of the core structures to accidentally imposed loads such as seismic events or blowdown must be established.

III

Safety Issue: Core Design

The neutronic stability of the HTGR core must be assured by design, by adequate monitoring, and by effective reactivity control. Potential xenon instabilities must be accommodated by the neutronic and thermal hydraulic design limits of the reactor. The core thermal and fluid dynamic design must provide adequate margin to assure that temperatures of the fuel and core structures are maintained well within established limits.

RESEARCH AND DEVELOPMENT

1. Safety margins in neutronic design of the core and reactivity control system must be defined and the applicability and validity of analytical methods, models, computer codes, and neutronic data established.
2. Safety margins in core thermal and fluid dynamic design must be defined and the applicability and validity of analytical methods, models, and computer codes established.
3. Means for assuring that core neutronics, thermal characteristics and fluid dynamics are within acceptable limits must be provided.

IV

Safety Issue: Accident Analysis

A critical area of the HTGR safety review will be the development of design basis accidents and their associated radioactivity source terms so that the adequacy of engineered safety features can be evaluated.

RESEARCH AND DEVELOPMENT

1. Performance of coated fuel particles (BOL & EOL) must be demonstrated for both normal and accident conditions.

2. Fission product concentration in the primary coolant and in deposits on primary coolant system components and pressure boundaries must be defined by establishing the applicability and validity of analytical techniques, methods, models and computer codes.
3. The applicability and validity of analytical methods, models and computer codes used to calculate temperature, temperature distribution, and fission product distribution for normal and accident conditions must be demonstrated.
4. The applicability and validity of analytical methods, models and computer codes used to calculate the consequences of the ingress of chemical impurities (other than H₂O) into the primary coolant must be established.
5. The applicability and validity of analytical methods, models and computer codes used to calculate the probability and consequences of introducing water into the primary coolant must be established. The reliability of the steam generators, water-seals and other components whose failure could result in water ingress, must be demonstrated. The quantity of H₂O (steam) released into the primary coolant via a failure(s) in the steam generator or other component and the extent of the reaction of water with graphite to form H₂ + CO must be considered and the acceptability of the related values used in your analysis must be demonstrated.
6. Techniques, equipment and instruments for detecting and monitoring chemical impurities in the primary coolant and for initiating protective action when impurity limits are reached must be provided and their reliability established.
7. The applicability and validity of analytical methods, models and computer codes used to calculate pressures, and temperatures in the containment structure following design basis accidents (e.g., rapid depressurization of PCRV, steam line break, steam generator tube break in conjunction with depressurization) must be demonstrated.
8. Techniques, equipment, and instrumentation for detecting, monitoring and controlling combustible gases in the containment structure must be provided and their reliability established.

Note: Logic diagrams (fault and decision tree analysis) and probability statistics would be helpful in identification of design basis accidents and in determining areas of safety significance requiring research and development.

V

Safety Issue: Operating Experience

The performance and reliability of many first-of-a-kind components are important to the safety of the HTGR.

1. Performance data for safety related components in operating HTGRs must be collected and evaluated to establish reliability and safety margins by comparison with designer specified performance and service life.

Enclosure 2

SUMMARY OF REGULATORY TECHNICAL ASSISTANCE CONTRACTS
AND
RDT FUNDED HTGR SAFETY RESEARCH PROGRAMS

REG TECHNICAL ASSISTANCE CONTRACTS

In order to develop and to implement a capability for independent analysis in the staff safety review and evaluation, Regulatory has initiated contracts with outside organizations. The purpose of these technical assistance contracts are (1) to develop a capability for review in these areas; (2) to provide additional expertise and manpower for the review of the initial applications; and (3) to develop guides that may be used by the staff and/or others in reviewing future applications.

The following contracts have been implemented:

(1) Modern Analysis, Inc.

"Probabilistic Study on Load Combinations for Concrete Reactor Vessel Design"

This program is directed toward establishing a capability for independently assessing the acceptability of the applicants' load selections and their combination. A method of estimating safety margins associated with the load combinations will be developed through which the safety of the PCRVR can be reviewed in a more consistent manner.

(2) Iowa State University

"Assessment of Behavior and Design of Steel Liners for Concrete Reactor Vessels"

This program is directed toward providing a capability for independently assessing the acceptability of the PCRVR liner. A comparative evaluation of selected methods of analysis will be made and a guide for the review and evaluation of a PCRVR liner design will be developed.

- (3) Battelle Memorial Institute
Pacific Northwest Laboratory

"Nuclear Graphite Evaluation for HTGR Technology"

The objectives of this contract are to provide an independent capability to assess the adequacy of the graphite and coated fuel designs proposed for the large HTGR. A report will be developed to summarize and assess all available information on the fabrication and properties of graphites for nuclear purposes and the behavior of various graphites in the reactor environment. A guide for review and evaluation of graphite components in large HTGR's will be developed.

- (4) Battelle Memorial Institute
Pacific Northwest Laboratory

"HTGR Metals Technology"

The objectives of this program are to develop an independent capability to review and evaluate metallic components proposed for use in the high temperatures and helium environment of the large HTGR. A report will be developed to summarize and assess the metals degradation processes and the mechanical properties of metals in the HTGR. A guide for the review and evaluation of major metal components in the large HTGR will be prepared.

- (5) Oak Ridge National Laboratory

"Analysis of Tritium Distribution and Release in HTGR"

Evaluation of possible safety problems associated with tritium during long-term operation of an HTGR.

- (6) Oak Ridge National Laboratory

"Evaluation of HTGR Behavior Under Abnormal Conditions"

Development of the capability to calculate the temperatures and temperature distributions in the HTGR core for conditions of flow reduction (e.g., channel blockage). The resulting temperatures will be utilized to evaluate the migration and transfer of fission products throughout the reactor.

Evaluation of reactor criticality for a variety of abnormal conditions such as loss of boron due to high core temperatures and redistribution of fission products and fuel. This program will also evaluate a number of computer programs used by GGA to estimate reactor behavior (e.g., FIPER, OXIDE, and STRESS II).

RDT FUNDED HTGR SAFETY RESEARCH

The research and development programs funded by DRDT are also important to our safety review and we plan to incorporate the results from these efforts into our review of the large HTGR. It is our understanding that the following programs are being supported by DRDT during FY 73:

(1) HTGR Fission Products Surveillance Program

This program at ORNL will monitor the fission product release in the Peach Bottom No. 1 and Fort St. Vrain HTGR plants during the course of their routine operation. Periodically the concentrations of all gas borne fission product isotopes will be determined. Collected and plated out solid fission products will be assayed, and fission product distribution in certain fuel elements will be measured.

(2) HTGR Fuel Chemistry and Fission Product Behavior

ORNL will conduct studies directed toward the chemical aspects of coated fuel particle design, the fission product release mechanism, and fission product transport study. These studies will evaluate the stability of the fuel particle kernel and coatings as a chemical system; investigate kernel and coating interactions, and seek to establish mechanisms controlling fission product and fuel loss from the coated particles.

(3) Safety Studies for HTGR's

This ORNL program encompasses several major tasks as described below.

(a) Fuel Integrity for HTGR's

This task at ORNL is directed toward a better understanding of the thermal stability of coated particle fuels operating at abnormal temperatures (> 1500°C).

(b) Fission Product Behavior in the Coolant Circuit

This task will study the characteristics of plated out fission products on reactor surfaces and the displacement from the surface by increased coolant velocity and/or changes in coolant composition.

(c) Consequences of Water Ingress

This task is directed toward the study of the reactions of steam with various graphites and with the pyrolytic coatings of the fuel particles.

(d) System Safety Analysis

This task will collect information on post-accident changes in important conditions, such as fission product distribution, coolant pressure flow and composition, and system temperature. They will also assess the probability and possibility of accident courses using logic (fault tree, diagrams and other analytical methods. The ultimate objective of this program will be to provide a description of the endpoint of a given accident sequence and to assess the changes of the endpoint being reached. The results from these studies could prove to be extremely valuable as a means of identifying areas for safety research and development as well as determining priorities for technical assistance programs.

GULF GENERAL ATOMIC COMPANY

April 19, 1973

COPY

P.O. Box 602
San Diego, California
Tel: (714) 453-1000

Mr. A. Giambusso
Deputy Director for Reactor Projects
Directorate of Licensing
U.S. Atomic Energy Commission
Washington, D.C. 20545

Dear Mr. Giambusso:

With reference to your letter on HTGR safety issues and related R&D, we want to express Gulf's appreciation for the attention and interest you have given to the HTGR and the opportunity afforded us to provide the Regulatory Staff with necessary and adequate information such that reviews of PSARs when submitted by Gulf's customers may proceed without undue delay. We are pleased now to send you the enclosure which provides substantive response to each of the issues in your letter.

The information on these, necessary to establish an adequate basis for review and evaluation, are provided in the PSARs being submitted. Extensive lists of reports and documents are being referenced in the PSAR. In addition, the PSAR is being supplemented with in-depth topical reports. One of these topical reports will cover safety related research and development and will be updated periodically with supplements as appropriate.

A Table summarizing the coverage we are giving to your listed items is also enclosed. The numbers appearing in the Topical Report column refer to the list of topical reports immediately following the Table. Items of R&D currently taking place or planned which will be covered in the R&D Report are noted in the Table against items in your list to which they are related.

We hope that the answers given in the enclosure and the information covering the safety issues and related R&D will provide an adequate basis for your review and evaluation of HTGR construction permit applications. Please let us know of anything that we could do to facilitate your efforts in this regard.

Sincerely,

/s/ J. M. Waage

J. M. Waage
Director
Engineering Division

JMW:vh
Enclosure

RESPONSE FROM GULF GENERAL ATOMIC TO LETTER FROM MR. A. GIAMBUSSO,
DEPUTY DIRECTOR, AEC'S DIRECTORATE OF LICENSING
TO HIS LIST OF HTGR SAFETY ISSUES
AND RELATED RESEARCH AND DEVELOPMENT

The information establishing an adequate basis for independent safety review and evaluation of the large HTGR is provided in PSARs being submitted. This includes a summary description of the plant facility, the design bases and their relation to the principal design criteria, and an analysis and evaluation of the design and performance of plant structures, systems and components. The information you require on the margins of safety during normal operation and transient conditions anticipated during the life of the plant and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents have also been provided. Those safety features or components requiring further research and development to confirm the adequacy of their design are identified and the related program are described in the PSAR. The PSAR coverage of the list of safety issues you identified (Attachment 1) is shown in Table 1.

In addition, the PSAR is being supplemented with in-depth topical reports providing additional details not appropriate for inclusion in the PSAR. The list of topical reports which we plan to provide is given in Table 2. The numbers appearing in the Topical Report column of Table 1 refer to this list. We also plan to provide a report and supplements as appropriate on the status and the information generated in the R&D program referred to in the preceding paragraph. Items covered in the R&D program that are related to the list of safety issues are also noted in Table 1.

The specific response to each of the listed issues is provided below. The numbers correspond to those of the list included in the letter.

I. Reactor Pressure Vessel Integrity

Considerable effort has been expended throughout the world to establish and ensure the integrity of prestressed concrete reactor vessels (PCRVS) as primary reactor pressure vessels. Research and development programs have covered analytical methods, structural model tests, safety margins, and the reliability, performance and redundancy of major components. The PCRVS has gained world wide acceptance and adoption as the reactor pressure vessel for gas-cooled nuclear power plants.

1. The applicability and validity of the analytic methods, models and computer codes used in the design of the PCRV have already been established.

At GGA, analytical techniques for the PCRV have generally employed the SAFE series of finite element computer codes. The codes have been proven on many structural analysis problems. In particular, the correlation between predicted and actual test results for scaled models and the actual Fort St. Vrain proof pressure tests have demonstrated the validity of such analytical methods.

The model tests conducted at GGA are supplemented by exchange agreements for similar data with design and construction organizations in France, England, Australia, Sweden and the USA. In fact, in support of the PCRV's in service or under construction, over 100 models have been constructed.

2. Safety margins and reliability have also been established by test programs for all major components such as tendons, liner, and thermal insulation. The success of these numerous models and component tests, demonstrating the predictability and safety margin in the PCRV, has been a major factor in their acceptance.

Codes of practice for design and construction of concrete vessels are being established throughout the world. In the US, a draft code will be issued in April for trial use and comment. This code, covering both concrete primary and containment vessels, is under the auspices of the ACI-ASME Technical Committee. It will subsequently be issued as Division 2 of Section III for Nuclear Power Plant Components. GGA has actively participated in the preparation of this code together with many members of the AEC staff. GGA's design will meet such code requirements as a minimum.

3. In addition to allowance of adequate margins and redundancy in the designs, high quality construction and fabrication with stringent quality assurance and quality control procedures, in-service inspection have been incorporated to assure the continuing integrity and functional reliability of essential safety components and systems. These include the PCRV concrete and post-tensioning components, penetrations, and thermal barrier and the liner cooling system.

The existing ASME Code, Section XI, for Inservice Inspection of Nuclear Reactor Coolant Systems, is not applicable to HTGR

plants since it is written for light water reactor plants which are different from the HTGR, among others, in both the coolant system and the reactor vessel. An expansion of the Code to encompass rules of practice applicable to gas cooled reactors is in progress and should be completed in 1974.

Gulf has developed an inservice inspection program for the HTGR which fully meets the intent and purpose of the existing ASME Code. The program will be modified to meet all applicable provisions of the expanded Code when it comes into existence. Requirements of the Gulf program were first applied to the Fort St. Vrain PCRV and associated vessel components. Details of the program for the FSV plant and the large HTGR are given in applicable SARs.

4. The PCRV pressure relief system provides protection against primary coolant system pressure exceeding the vessel reference pressure. The same basic design as that of FSV is employed. Two trains are provided, either of which is adequate to prevent exceeding the reference pressure in the event of any credible overpressure accident. On the basis of the assumed design basis accident for the relief system, results of analysis indicate that there exists a further safety margin of over 100% in each train. In order to call into play the relief system, it is necessary to postulate the prior failure of all three levels of protective actions.

Because of the high pressure and the enormous flow required it is not practicable to test a complete train of the relief system. However, the key components such as the valve operator of the relief valve and the rupture disk actuation system are tested. The former is performed by the manufacturer and the latter once a year at the plant site. A testing program was established for FSV with the concurrence of the Regulatory Staff. The same program, modified to take into account any operating experience gained, would be employed. An extensive program of reliability study is in progress to provide a basis for the technical specifications and the requirements of maintenance and routine testing.

II. Integrity of Core Structure

1. Safety margins have been defined for stresses and dimensional changes in hexagonal graphite elements, and included in the PSAR for the large HTGR. These include limitations on primary stresses alone and in combination with thermal and irradiation-induced stresses. The same graphite material is being used at the FSV plant.

Two computer programs have been developed for analysis of stresses and dimensional changes in the fuel elements under thermal gradients and irradiation. One program is used to survey the core and locate regions of maximum stress and deformation, while the other is used for more detailed analysis of these regions.

An extensive test program is in progress to determine the physical and mechanical properties of the graphite material before and after irradiation over the full range of fast neutron fluence and temperature conditions in the large HTGR. Appendices E & F of the PSAR provide a complete discussion of the research and development programs being conducted on the core materials and components. Test elements in the Peach Bottom HTGR are demonstrating the capability of graphite to operate at stress levels about twice as high as those in a large HTGR. All data collected to date indicate that the stress design criteria employed for the large HTGR provide ample margin between the graphite peak stresses and the graphite strength.

The design of the large HTGR fuel and reflector elements, coolant plenums, control rods, and reserve shutdown materials is basically the same as that used in the Fort St. Vrain HTGR. The Design and safety analyses of these components is well documented in the FSV FSAR and amendments. The operation of the FSV reactor will demonstrate the total adequacy of the basic design for the large HTGR. A core surveillance program will be conducted to evaluate the performance of the fuel and reflector elements and control rods. This will provide added data to demonstrate the use of these basic designs in the large HTGR.

Loads in the coolant plenums during normal service and blow-down were shown by the analysis for the FSV reactor to be less than one-third of the yield strength of the steel structures. The plenum elements for the large HTGR will be designed to the same conservative levels.

Graphite has also been selected as a structural material for the core support floor on the basis of its compatibility with the reactor core and its strength up to very high temperatures. The design is basically similar to that employed on the Fort St. Vrain. Large graphite blocks support each fuel region. Each block is supported on three graphite posts. Proof load tests on prototype components have indicated a safety factor of about 8 on the block and 12 on the posts for the maximum vertical load imposed by dead weight and gas pressure drop differential.

There is a negligible increase in vertical loading due to blowdown or seismic condition. The effect of the operating environment, including severe temperature transients, has been investigated and found to have inconsequential effects on the structure.

The lateral support structure performs primarily a passive function to restrain and align the core. It performs a safety function only during the seismic conditions as described below. This structure is cooled by gas bypassing the core. It is constructed from low-alloy nickel-based steels. The design is in accordance with the proposed draft Section NG "Core Support Structures" of the ASME Code Section III, Division 1. The quality assurance requirements also meet code specifications.

2. To assure the integrity of the core and its support structure during seismic disturbance, an extensive research and development program has been undertaken. This program is intended to demonstrate the adequacy of the HTGR core response to high seismic loadings. The program involves both experimental and analytic studies of HTGR cores under conditions experienced during earthquakes. The overall objectives of the program are to develop and validate analytical methods predicting core response to seismic disturbances, to demonstrate structural and functional integrity and design margins of the core and core structure components.

Details of the program are given in the PSAR being submitted. Tests giving the inter-element collision forces and frequencies in the core assembly, and the shear forces on the core element dowels, have been satisfactorily completed. Other parts of the program are still in progress. The completed program will be covered in a topical report.

III. Core Design

The items delineated in this area are design related rather than specific "safety" issues. In the design activities at GGA to-date these items have been recognized and considered in the core nuclear design. Analysis and evaluation of the neutronic stability of the HTGR, covering shutdown margins, control rod worths, reactivity coefficients, xenon stability, and applicable uncertainties, have been defined to assure that the core operates within established limits and that the design objectives are met. Future design and analysis will continue to focus on these design issues.

1. The validity of our current analytical tools, models, and basic nuclear data has been tested through evaluation and analysis of the HTGR critical assembly experiments, the Peach Bottom HTGR and the HTLTR experiments. A further test of our codes will be the forthcoming analysis of startup experiments and subsequent operation of Fort St. Vrain, which will provide an excellent check of both analytical tools and basic data. In addition, we are continuing to analyze existing experimental data to further establish their validity with actual test results.

Consistent with the status of the nuclear design, the PSAR provides adequate information on control system reactivity worth, temperature coefficients, xenon stability and uncertainties in excess reactivity and control rod worths, and contains an extensive discussion of applicable experimental work, i.e., the critical facilities and the Peach Bottom HTGR. The PSAR also refers to the Fort St. Vrain reactor startup and operation as an additional source of experimental data to check our calculational methods.

The condensed discussion of computer codes and their validity in the PSAR will be supplemented with a topical report entitled "Nuclear Design Methods and Experimental Data in Use at Gulf General Atomic" which should be submitted to the Regulatory Staff this summer. The report integrates the experimental data, the calculational results and the calculational methods used to analyze the experiments to establish the validity of our codes and models. We plan to extend and update the report as Fort St. Vrain startup tests, initial operation and evaluation are completed.

We are also planning to submit a topical report on core power distributions, covering steady state, transient and load following operation, design and off-design conditions, and limiting conditions under various incidents. This is a more appropriate place to provide the information than the PSAR. The report should be issued during the latter part of this year.

2. The primary objective for the HTGR core thermal and coolant flow design is to provide assurance that the fuel particle coating retains its integrity and fission product retention capability during normal operation and operating transients. The analytical methods and the computer codes utilized to perform the design analysis are based on models which have been developed using accepted engineering correlations and conservative material properties and supported by data from experiments.

The validity of the design methods are being further established through several experimental programs currently in progress and actual performance data from Fort St. Vrain testing and initial operation.

The inlet flow to each region of the core is controlled on the basis of the measured outlet temperature from each region. The required tolerances within the core, thermal effects and irradiation induced graphite dimensional changes will provide flow paths in addition to regular coolant flow channels resulting in a redistribution of flow depending upon local pressure values and relative flow resistances. Detailed core and single refueling region flow models have been developed to determine the core coolant flow and pressure distribution over a range of potential local core gap distributions.

The validity of the computer code has been verified by comparison with 0.45 scale multicolumn flow tests. Additional flow tests will be performed in extensively instrumented full-size fuel blocks to further define flow distributions. Planned full-scale flow tests of an entire instrumented refueling region will provide additional verification of the design methods employed.

Thermal codes which utilize time dependent detailed core power distributions have been developed to determine the time dependent fuel and moderator temperature distributions as well as the performance of the fuel particles. Variable material properties are included.

Uncertainty analysis is performed to determine the systematic and random effects to be applied to the core hot spot analysis. The systematic effects include such items as power tilts, coolant temperature measurement error and region temperature mismatch. The random effects include manufacturing parameters, conservative estimates of material property uncertainties and power distribution uncertainties.

Several unique characteristics of the HTGR design provide additional assurance that adequate safety margin exists in the core thermal and flow design. The inert nonreacting single phase coolant provides predictable, nearly constant, heat transfer characteristics. The large heat capacity of the graphite moderator provides a thermally stable system which is not subject to rapid temperature changes. Known fuel failure mechanisms are time, temperature, and burnup dependent, and are well outside of the maximum predicted operating conditions for the HTGR core.

C
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3. Extensive in-core and coolant circuit instrumentation requirements for assuring that core neutronics, thermal characteristics and fluid dynamics are within acceptable limits have been specified in the PSAR. A test program is planned to improve and extend the capability and the performance characteristics of instruments that are presently available.

IV. Accident Analysis

1. Performance of Coated Fuel Particles

Fuel particle coating performance can be limited by two factors; (1) high stresses generated during irradiation, and (2) thermochemical effects. Coated particle failure due to mechanical stresses is limited to very low values (1% or less) during normal operation and normal transients by appropriate coated particle design and specification. The design and specification of coated particles are in turn established by the results of extensive irradiation tests which are interpreted with the use of a coated particle stress model and statistical analyses.

The amoeba effect is the dominant thermochemical effect: it takes the form of unidirectional migration of the fuel kernel into the coatings under the influence of a temperature gradient. Extensive theoretical and experimental studies have accurately defined the mechanism of the amoeba effect and the function dependence of the rate of migration on time, temperature, and temperature gradient. The results of irradiation tests and postirradiation annealing experiments indicate that there is no significant effect of irradiation on the rate of amoeba migration. The amoeba effect is the controlling factor in core thermal design over a wide range of temperatures.

At high temperatures (greater than about 1600°C) coating failure due to radiation induced stresses begins to occur in particles that have experienced high burnup. As a result of the wide distribution of internal gas pressures in a population of coated particles, this type of coating failure occurs over wide range of temperatures above 1600°C, with many particles remaining intact to temperatures over 2000°C.

An extensive program of in-pile and out-of-pile testing is continuing in order to further improve the statistical confidence in coated particle performance.

The details of the program are reported in the PSAR. A topical report on this subject is also planned for the fall of 1973.

2. Fission Products Within the Primary Coolant Circuit

The inventory of gaseous fission products in the primary coolant is primarily based upon their fractional release from failed fuel particles in rods. Extensive short and long term irradiation tests have been and are being conducted to measure the fractional gaseous release. In these tests, in-pile fractional release measurements taken during irradiation are correlated to the end-of-life failure fraction determined in post-irradiation examination experiments. These two basic properties are the primary experimental input to calculating gaseous fission product inventories. Natural decay and the primary coolant purification system removal also limit the fission product inventories. The coolant radioactivity inventory is calculated from solutions of simple differential rate equations.

Fission product metals have an additional source term in that some can diffuse through intact BISO fuel particles. Long term annealing experiments have been conducted to determine metallic fission product diffusion coefficients in the fuel particle. The metals also have additional barriers to overcome before release. Once out of the fuel particles the structural material of the fuel rod and graphite blocks limit the amount released. The release of metals is calculated using the FIPER code. Fission product concentration profiles predicted by the FIPER code have been tested successfully against profiles measured in Peach Bottom fuel elements and from the fuel body used in the GAIL IV loop irradiation.

Fission product coolant and surface concentration distributions are dependent on the mass transport characteristics between the coolant and surfaces of the reactor circuit. A computer program which yields the plateout activity distribution has recently been developed to numerically solve the coupled gas phase and surface concentration of deposited fission products. This code has been qualitatively verified from Peach Bottom and Dragon primary circuit measurements.

The PSAR provides additional details on how fission product inventories are determined and the calculational results.

3. Temperature and Fission Product Distributions

The issues raised here in large part have been covered by specific responses to III-2 and IV-2 above. Information on the additional area of accident conditions is provided at length in the PSAR. Several topical reports are planned to provide extended details not appropriate for the PSAR.

4. Air Ingress and Consequences

The air ingress of chemical impurities of any significance besides water in the primary coolant is atmospheric oxygen. The possibility occurs after a blowdown accident. Detailed evaluation has been made of the potential for air inflow and of the consequences. We have concluded that air ingress presents minor safety issues. The only significant result from it is the oxidation of a small fraction of the core graphite. This subject is adequately discussed in the PSAR.

5. Water Ingress and Consequences

The potential exists in the HTGR plant design for the occurrence of steam leaks into the primary coolant system. Accidental steam inleakage can result in varying degrees of reaction with the core graphite. However, due to the existence of the PCRV and the engineered safeguards (including highly redundant moisture monitors, steam dump system and reactor shutdown system) as well as feedwater flow limiting orifices, there is no single accident event, or foreseeable series of events, which could result in danger to the containment structure or in significant release of radioactivity to the environment.

Extensive studies of steam inleakage have been made. The results of these studies are reported in the PSAR. A detailed analysis of the systems provided to limit steam inleakage and of the consequences of various postulated accident conditions is covered in a published topical report, GAMD-9804. An updated version of the report incorporating additional analyses is being prepared. The documentation of the OXIDE computer code which is used to calculate the consequences of steam ingress will be completed this year.

6. Detection of Coolant Chemical Impurities

The primary coolant is regularly monitored for chemical impurities including carbon monoxide, carbon dioxide, nitrogen, oxygen, hydrogen and methane by means of gas chromatography.

Samples for the chromatograph are periodically taken from the primary coolant and helium purification systems. This ensures that the impurity levels as provided in the Technical Specifications of the license are not exceeded and that the helium purification system is performing as expected. In addition, the primary coolant is continuously monitored for the presence of carbon monoxide by means of an infrared analyzer. Levels of CO higher than normal are alarmed at sufficiently low levels to enable the operator to take corrective action. There is no requirement for automatic protective action initiated by the measuring instrumentation. If the limits of Technical Specifications are exceeded the plant operator is required to take appropriate actions consistent with the conditions of the license. The monitoring instruments employed are regularly calibrated and maintained. Recorded readings from the monitoring instrumentation form an integral part of a plant's operating record.

In the case of moisture impurity in the primary coolant, automatic protective action above certain specified levels is provided. Each main and auxiliary coolant loop is instrumented with three independent moisture detection systems. Protective actions are automatically initiated. These moisture detection systems trip the reactor, and isolate and dump the leaking steam generator to minimize the reaction of steam on graphite and to terminate further steam leakage. The detection instruments and the protective action circuits are tested at regular intervals.

The detection instruments commercially available are currently being subjected to a comprehensive qualification test program. The information provided in the PSAR will be supplemented by the scheduled R&D report as the test proceeds.

7. Containment Transients Following Design Basis Accidents

The design basis accident for the HTGR containment is the depressurization of helium from the PCR. Since helium, a single phase gas, is the only fluid involved, the analysis and prediction of successful core cooling and containment pressures following the accident is straight forward and based on established thermodynamic principles. The break of a steam line in the containment results in containment pressures lower than that of the depressurization accident and poses no threat to either the containment or the PCR.

Gulf General Atomic analyzes the accidents described above in designing and building HTGRs. The computer code applicable to these calculations is named CONTEMPT-G. The CONTEMPT-G code uses many of the computer routines of the CONTEMPT code which was developed in Idaho at the National Reactor Test Station for water reactors. The original program has been modified to treat helium and to handle superheated steam in the HTGR plant. The resulting code, CONTEMPT-G, is then used with routines and functions for gas-cooled reactors rather than the water injection routines customary for water reactors. Gulf has previously made a copy of the code available for use by members of AEC Regulatory Staff.

8. Detection and Control of Combustible Gases

In view of the specific responses provided under IV-5, IV-6 and IV-7, the information related thereto and the topical report, GAMD-9804, provided the Regulatory Staff, we consider that full and adequate information on the subject of combustible gases in the containment has been provided. We have concluded that the subject does not constitute a safety issue.

7. Operating Experience

1. Performance Data for Components in Operating HTGRs

Reports on Fort St. Vrain plant operations will be submitted in accordance with the provisions of the Technical Specifications. As operating data becomes available from FSV, that data relevant to providing further verification or assurance of the design bases for large HTGR components or systems will also be submitted to the Regulatory Staff in formal reports.

TABLE 1

COVERAGE OF LIST OF SAFETY ISSUES

ISSUE	PSAR	REPORTS TOPICAL*	R&D
I. Reactor Pressure Vessel Integrity			
1. PCRV Design	X	15	
2. Safety Margins	X		
3. Continuing Integrity	X	19	
4. PCRV Pressure Relief System	X	3	
II. Core Structure Integrity			
1. Safety Margins, Design, Continued Integrity	X	17	
2. Response to Seismic & Blowdown Loads	X		X
III. Core Design			
1. Safety Margins, Design: Reactivity	X	1,8	
2. Safety Margins, Design: Thermal & Hydraulic	X	2,9,10, 13,16	
3. Continued Assurance Within Acceptable Limits	X		X
IV. Accident Analysis			
1. Fuel Particles Performance, Normal & Accidents	X	5	
2. Circuit Fission Product Concentration & Deposits	X	14,20	
3. Temperature & Fission Product Distribution	X	11,12,18	
4. Air Ingress & Consequences	X		
5. Water Ingress & Consequences	X	4,6	
6. Chemical Impurities Detection & Protection	X		X
7. Containment Transients Following DBAs	X	7	
8. Detection & Control of Combustible Gases			
V. Operating Experience			

*Numbers refer to reports listed in Table 2.

HTGR TOPICAL REPORTS SCHEDULE

<u>Subject</u>	<u>Month Ending Date</u>
1. Nuclear Design Methods & Experimental Data in Use at Gulf General Atomic	6-73
2. Review of Graphite Thermal Conductivity	6-73
3. Reliability of the PCRV Pressure Relief System	7-73
4. OXIDE Code	7-73
5. Fuel Particle Behavior & Performance During Normal & Transient Condition	9-73
6. Water Ingress & Consequences	9-73
7. CONTEMPT-G Code	9-73
8. Core Power Distribution	10-73
9. Core Thermal Safety Limits	10-73
10. Thermal Properties of Core Materials	10-73
11. SORSE Code	11-73
12. FIPER Code	12-73
13. Core Thermal Analysis	1-74
14. Determination of Fission Product Release from Core	2-74
15. PCRV Design & Verification	3-74
16. Methods of Thermal & Stress Analyses in Use for Core Design	3-74
17. Materials & Integrity of Core Structures	6-74
18. Core Cavity Flow & Pressure Distributions	6-74
19. Continuing Assurance of Pressure Vessel Integrity	6-74
20. Physical Data Employed in Fission Product Transport Calculations	6-74

Appendix B

PEACH BOTTOM HTGR END-OF-LIFE PROGRAM

INTRODUCTION

This section presents a program for end-of-life experiments and post-operational examinations on the Peach Bottom HTGR which is currently expected to shut down permanently in September 1974.

This plan has been approved by the Philadelphia Electric Company subject to review of individual experiments by the applicable safety committee. Immediate action is needed if the activities proposed are to be fitted into the Philadelphia Electric Company's decommissioning schedule.

Due to the magnitude of the overall effort contemplated and the imminence of the decommissioning activities, a separate plan to cover the necessary safety research has been prepared. This plan is presented in this section. All work described is included in appropriate validation tasks of the main program plan. The tasks are identified in Table B.1 together with a cost breakdown. Figure B.1 contains the proposed schedule for the program. Table B.2 summarizes the work proposed.

B.1 BUDGET AND SCHEDULE

The budget and schedule proposed for the Peach Bottom end-of-life program are shown separated from the program activities in Table B.1 and Fig. B.1.

There is little flexibility in this schedule. The start is determined by the date at which Core 2 reaches end-of-life (EOL). Delay means loss of valuable information associated with ^{89}Sr , which will be one of the few fission-product nuclides present that can only have originated from Core 2.

Since ^{89}Sr will be sought in all reactor components, delay of even the removal of the steam generator will be detrimental, though not as important as a general delay in the postmortem program. Major program milestones, also shown in Fig. B.1, are listed below:

Table B.1. Cost and schedule of proposed Peach Bottom HTGR end-of-life program

Task	Planning guide task group	Contractor	Estimated costs (\$1000) for fiscal year				Total
			1975	1976	1977	1978	
Planning		GAC/Suntac	70				70
Execution							
I. Pre-shutdown phase (H ₂ injection)		GAC	20				20
II. Post-shutdown phase							
Analysis of coolant samplers, dust and γ survey	2.6	ORNL	80				80
Fuel-failure location expt.		GAC	200				200
Tritium mass balance	2.7	GAC	20				20
III. Fuel element (3) and reflector element (3) shipping		ORNL (sub)	25				25
IV. Core component examination							
Fuel element	2.5, 2.7	ORNL	200	150			350
Reflector element	2.5, 2.7	ORNL		150			150
Control rod(s)	2.7, 6.1	GAC		100	50		150
V. Primary circuit component removal and shipping to ORNL		Suntac	125	900			1025
VI. Primary circuit component examination							
Metallurgical analysis	3.2, 6.3	ORNL	85	300	240		625
	3.2, 6.3	GAC		100	120		220
Radiochemical analysis	2.6	ORNL	20	260	90		370
Steam generator tube tritium permeability	2.7	GAC		60			60
Dust and fission product re-entrainment	2.6	GAC		160	20		180
VII. Reactor performance evaluation	2.0, 3.2, 6.3	ORNL, GAC			60	120	180
Total			845	2180	580	120	3785

Fig. B.1. Schedule for Peach Bottom end-of-life program.

Table B.2. Summary of Peach Bottom end-of-life program

	Test	Measurement	Objective
I	Pre-shutdown phase Hydrogen injection	Partial pressures or concentrations of H, T, and CH ₄ in primary coolant; H, T in secondary coolant as functions of time for constant injection rate.	Verification of functional form of equations describing tritium permeation through boiler. Determination of average boiler permeability to hydrogen isotopes. Measurement of CH ₄ production rate as function of P _{H₂} . Test of tritium-distribution code. Determination of core capacity for adsorbed hydrogen.
II	Post-shutdown phase Gas sampler removal Survey of gamma emitters Failed-fuel location technique demonstration	Amounts of fission products in coolant leaving and entering core. Distribution of activity in primary circuit. Fission-gas release from fuel element in which local fissioning is caused by Cf-252 source.	Detection of change in rates of fission-product release from core with time Observation of changes in fission-product distribution in primary circuit. Demonstration of method of detecting element from which fission-product release is abnormally high by use of a traveling neutron source and fission-product gas detector.
IV	Postirradiation examinations of core components Fuel elements	Fuel and burnable-poison integrity by metallography. Fuel inventory of fission products and heavy metals. Dimensional changes Fission-product distribution throughout fuel-element graphite. Fission-product concentration in a compact matrix. Deposition of impurities on the fuel element.	Verification of fission-product transport models. Determination of time of cesium and strontium breakthrough in relation to irradiation time using various fission transport models for extrapolation as appropriate. Determination of fission-product transport in fuel bodies and other fission-product transport parameters. Prediction of coolant circuit cesium inventory. Determination of efficiency of core as a sink for iron and other circulating impurities.

Table B.2 (continued)

Test	Measurement	Objective
	Sufficient examination to determine overall mass balance for Sr, Cs, I, ^3H . Distribution of tritium in element.	Verification of prediction of tritium behavior in a graphite system. Verification of fuel inventory calculations. Comparison of fuel behavior with predicted performance.
Side-reflector element (movable reflector)	Dimensions and shape.	Verification of design predictions.
	Distributions of fission products, circulating impurities, and tritium.	Determination of value of reflector as a sink for circulating activities.
Control rod	Dimensions and shape. Metallographic determination of compact oxidation or damage and of metal-part reaction with coolant. Tritium loss from $^3\text{H}/^7\text{Li}$ ratio.	Confirmation of acceptability of design. Establishment of whether tritium formed in control rod is lost to the coolant.
VI Primary circuit component examination		
Fission products	Distribution of fission-product metals and I-129 in primary circuit, in particular Sr-89, Cs-137. Distribution of strontium and cesium between surface deposits and substrates. Distribution of gasborne solids and corrosion products and activation products.	Establishment of reliable inventories and distribution profiles for iodine, cesium, strontium, and other fission products in the primary coolant circuit. Comparison of predicted releases and distributions with those found.
	Chemical nature and morphology of dusts and deposits in primary circuit parts.	Definition of the importance of dust as a fission-product carrier.
	Ease of removal, both chemically and aerodynamically, of surface deposits, including out-of-pile blowdown terms.	Improvement of depressurization accident analysis.
	Permeability of used steam generator tubing to hydrogen.	Estimation of effects of oxide films on tritium permeation of boiler tubing.

Table B.2 (continued)

Test	Measurement	Objective
Metallurgical studies in the primary coolant circuit (reactor vessel excluded)	Metallographic examination of pieces of primary circuit for signs of interaction with the coolant. Determination of changes in mechanical properties of all structural materials as a result of use in the reactor. Measurements on pieces of helium ducting and insulation, hot valve gate, steam generator tubes, tube sheet and tube supports, circulator impellor and shroud.	Confirmation of materials suitability for HTGR service. Note: Materials used in the Peach Bottom steam generator are similar to those intended for use in large HTGRs.

Peach Bottom end-of-life program milestones

- (1) Acceptance of program by Philadelphia Electric Co.
- (2) Complete cost estimates.
- (3) Award contract for component removal.
- (4) Approve contractor's detailed proposal.
- (5) Remove helium samplers. Begin coast down to lower power.
- (6) Final shutdown.
- (7) Complete H₂ injection experiments.
- (8) Report on coolant impurity behavior.
- (9) Report on fission-product behavior in primary circuit.
- (10) Report on ³H behavior in Peach Bottom.
- (11) Report on feasibility of failed-fuel location system.
- (12) Report on distribution of fission products in core components.
- (13) Report on control rod performance.
- (14) Finalize engineering studies for circuit component removal.
- (15) Begin circuit component removal.
- (16) Ready to receive coolant circuit parts.
- (17) Report on metallurgical studies.
- (18) Report on fission-product behavior on primary circuit surfaces.
- (19) Report on permeability of steam generator tubes to ³H.
- (20) Report on dust re-entrainment.
- (21) Evaluate reactor performance — comparison of expectation and reality.

B.2 PROGRAM JUSTIFICATION AND SCOPE

The decommissioning of the Peach Bottom HTGR offers a unique opportunity to obtain otherwise unavailable information about the performance of an HTGR core and related components. Before shutdown, the program will develop new information on (1) the release of fission products from a 900-day-old core, (2) the deposition rates of iodine, cesium, and other condensable fission products, and (3) the behavior of tritium in an HTGR. Following the final shutdown, complete inventories of all condensable fission products in the primary circuit, including the steam generator, will be made. Components of the core will be sampled to determine the state of the fuel and the distributions of the reference set of fission products (isotopes of H, Te, I, Cs, and Sr). Metal components of the primary loop will be examined to determine performance, the presence of incipient defects, interaction with the coolant, mechanical damage, and changes in mechanical and chemical properties.

The information to be obtained is unique in the following respects: (1) it relates to an HTGR that has run for 7 years, whose core contains predominantly one type of element; (2) the irradiation time of the Core 2 elements is longer than that of other U.S. HTGR elements; (3) almost all parts of the primary circuit will be accessible; and (4) all nuclides of interest can be measured. It will facilitate safety and design analyses. Specifically, attempts will be made to verify the three fission-product distribution models — FIPER, PAD, and TRITGO. Examination of ducting, a steam generator, a control rod, and a circulator will generate confidence in the suitability of relevant materials for HTGR service and the performance of key components. There will not be another opportunity to make such an examination for several decades.

B.3 THE PEACH BOTTOM HTGR

The Peach Bottom Atomic Power Station, the HTGR prototype plant, is owned and operated by Philadelphia Electric Company and is located on the Susquehanna River about midway between Philadelphia and Baltimore. The plant, with a power of 40 MWe, is the world's first nuclear power station to produce commercial electric power at a modern steam conditions of 1450 psi and 1000°F temperature.

The plant consists of a containment structure housing the reactor, a steel pressure vessel, primary helium ducting and circulators, steam generators, and auxiliary systems; a building for the turbine-generator, condensing system, and feedwater systems; an auxiliary services building; offices and shop; and the main transformer. As

shown in the accompanying isometric view of the primary coolant system (Fig. B.2), the helium coolant at a pressure of about 350 psi flows through the reactor core where it is heated to 1300°F. From the reactor the helium is directed to the steam generators. After passing through the steam generators, the coolant is returned to the reactor by helium circulators. Two identical loops are used, each including a single steam generator and circulator.

The core of the Peach Bottom HTGR is composed of 804 individual fuel elements vertically oriented in a close-packed array. The core array is cylindrical, with an effective diameter of 9 ft and an active height of 7.5 ft. The helium coolant gas passes upward through the tricuspid-shaped passages formed by the elements.

The Peach Bottom fuel element, shown in Fig. B.3, has an overall length of 12 ft and an outside diameter of 3.5 in. and consists of an upper reflector section, a fuel-bearing middle section, a bottom reflector section, and an internal fission-product trap containing activated carbon. A low-permeability sleeve extending from the upper reflector to the bottom reflector contains the fuel compacts and the internal fission-product trap. The fuel compacts consist of fuel particles in a graphite matrix. The fuel particles are between 200 and 600 μm in diameter and consist of uranium-thorium carbide particles coated with pyrolytic carbon.

Helium purge gas enters each fuel element through a porous plug in the upper-reflector piece and then flows downward and around the fuel compacts, sweeping volatile fission products out of the grooved space between the fuel compacts and the graphite sleeve. The purge gas then flows through the internal trap, where some of the fission products are absorbed by the activated carbon. Volatile fission products leaving each internal trap enter a purge line leading to the external fission-product trap. The primary coolant system consists of the reactor, two steam generators, and two helium compressors arranged to form two parallel loops for circulating the helium coolant.

Commercial operation of the Peach Bottom Power Station began in June 1967. In October 1969, the reactor was shut down after 450 equivalent full-power days. The reactor attained full-power operation on July 14, 1970 with its second core. Up to December 1973, more than 1,000,000 MWhr of electrical energy had been generated at the Peach Bottom Atomic Power Station with a total release of only about 600 Ci of gaseous and 0.03 Ci of liquid radioactivity to the environment.

Fig. B.2. Peach Bottom HTGR plant layout.

Fig. B.3. Diagram of fuel element for Peach Bottom HTGR.

The excellent operating record of this plant has been an important step in the demonstration of HTGR operating and performance characteristics. Although the physical characteristics of the reactor vessel, steam generators, and certain other components of the system are different from those employed in large HTGR systems, many significant features are very similar.

B.4 PRE-SHUTDOWN TESTS

B.4.1 Injection of hydrogen into the primary circuit

This test involves increasing the hydrogen level in the helium coolant and observing the effect on tritium transport and levels of gaseous impurities in the coolant.

The objectives of the test are: (1) to establish the relationship between hydrogen and methane concentrations in the helium coolant, (2) to determine the effect of change in hydrogen concentration on tritium transport through the steam generator tubes, and (3) to obtain data on the capacity of the core for hydrogen. The data would augment existing data for use in fixing impurity-level technical specifications for the large plant and in predicting tritium transport in the large plant.

The hydrogen level in the helium coolant would be increased in steps to around 100 vpm by operating with reduced flow or no flow to the copper oxide bed. An alternative procedure would be to inject hydrogen from an external source. The latter procedure would allow the exact injection rate to be known. The methane level would not be allowed to exceed the technical specification of 2 vpm. (Hydrogen levels have already been as high as 130 vpm during Core 2 operation.) Gaseous impurity levels in the helium coolant and tritium levels in the helium coolant and secondary coolant water would be monitored during the test.

Existing analytical equipment is believed to be adequate. The duration of the test would be 10 days.

The result will be analyzed using the TRITGO and GOP codes that describe tritium distribution and coolant composition in HTGRs.

B.5 POST-SHUTDOWN TESTS

B.5.1 Fission-product release and distribution measurements

Immediately after the last shutdown, before reduced power operation begins, the two coolant samplers will be removed from Loop 1 for radiochemical analysis. Following the final shutdown a survey of gamma-emitting nuclides in the primary circuit and analyses of dust in the centrifugal dust collector will be made.

Analysis of the samplers will yield (1) concentrations of iodine, cesium, and strontium in the gas up and downstream of the steam generator, (2) rates of deposition of iodine, cesium, and strontium on the steam generator, (3) average rates of release of these elements and their individual isotopes from the core, (4) information on the role of dust and organic iodine species in fission product transport, and (5) evidence of cesium and strontium breaking through fuel sleeves.

B.5.2 Failed-fuel location system demonstration

A roving neutron source may be used for failed-fuel-element detection in large HTGRs. In a large HTGR, the method involves axially traversing a ^{252}Cf source (10^9 neutrons/sec) through a control-rod hole of the region to be tested for failed fuel. The source induces fissioning in the fuel, and fission gas released, proportional to the amount of failed fuel, is sampled by means of a sniffer probe positioned at the outlet of the region.

Demonstration of the technical feasibility of this method in the Peach Bottom HTGR has been proposed. Ideally, the program would involve fabricating two special test components. One component would consist of a ^{252}Cf source contained within a standard Peach Bottom Core 2 sleeve and upper reflector assembly for ease of handling and for core compatibility. The other element would be a test "failed" fuel element. It would be of standard fuel test element design but would contain a driver fuel. However, at specified axial locations, some of this driver fuel would be replaced by special fuel rods fabricated with different amounts of bare fuel particles. The element would probably contain several regions of different "failure" fractions.

At the beginning of the Peach Bottom EOL program, the failed fuel element would be inserted on an individual purge-sampled position. The source element would be systematically substituted for Core 2 fuel elements at different distances from the failed fuel element. At each position, the fission gas released by induced fissioning

would be measured in the individually purged stream. The procedure would serve to establish source efficiency and sensitivity in a purged system.

Finally, the source element would be traversed axially alongside several other purge-sampled fuel test elements scheduled for post-irradiation examination. The observed release would subsequently be correlated with fuel failure fractions determined during the examination.

B.5.3 Tritium mass balance

Tritium in all accessible plant components will be determined following final shutdown, so that, when core analyses are complete, a plant balance may be made.

B.6 CORE COMPONENT EXAMINATION

In addition to the various special elements that are scheduled for post-irradiation examination at ORNL or GAC, the following Peach Bottom core components will be removed for detailed examination to improve understanding and test predictions of performance:

- (1) several high-exposure Core 2 elements,
- (2) several hexagonal reflector elements,
- (3) a control rod.

The reflector elements will be removed as soon as possible after shutdown. The control rod can only be removed after the remaining fuel. Examinations of similar items irradiated for shorter times have been performed previously to fulfill technical specification requirements and to obtain information on changes in fission-product distribution with time. The results obtained would be directly comparable with those obtained previously and would be invaluable in testing time extrapolations of element performance, particularly with respect to cesium and strontium releases.

Axial and radial distributions of fission products on and within the graphite will be determined for the fuel elements. Similar measurements will be made on the hexagonal reflector elements so that proper account may be taken of fission-product deposition on unfueled core components in future fission-product distribution calculations. The state of the fuel will be recorded, and elements

will be searched for mechanical defects. Emphasis in the control rod examination will be on establishing mechanical reliability and the extent of any losses of tritium that may have occurred.

The results will be analyzed using the FIPER, PAD, and TRITGO codes. Successful prediction of core performance would enhance confidence in core-design and fission-product control methods.

B.7 PRIMARY CIRCUIT COMPONENT EXAMINATION

Brief descriptions of the radiochemical and metallurgical examinations in Sects. B.7.1 and B.7.2 are followed by a chronological description of the primary circuit dismantling.

B.7.1 Fission product behavior

The specific objectives of studies of fission products on primary circuit components are to:

- (1) Examine the distributions of fission products, with emphasis on ^{90}Sr and ^{129}I which have not been measurable before, and obtain a more precise measurement of the contents of the steam generator. This will involve cutting out samples of ducting and steam generator tubing.
- (2) Examine primary circuit parts to determine types and distributions of solid deposits.
- (3) Determine the chemical and physical characteristics of deposits.
- (4) Perform tests on samples from selected components including steam generator parts to determine how readily surface deposits may be re-entrained in the gas stream.
- (5) Identify fission products on surface deposits and their distributions between substrate, surface films, and base material on samples cut from metallic components in the main coolant circuit.
- (6) Determine effect of circuit conditions on components using mechanical and other tests as required.
- (7) Perform hydrogen and tritium permeability studies on selected steam generator tubes.
- (8) Determine amount and distribution of activation products.

These measurements complement those made previously during operation.

The information gained will be used (1) to verify the predicted whole core release, (2) to improve analyses of the consequences of depressurization accidents, (3) to establish the relative capacities of various surfaces for fission products, and (4) to determine whether there are any chemical interactions between fission products and circuit materials.

B.7.2 Metallurgical examinations

The objective of this part of the program is to establish the effect on HTGR structural materials common to Peach Bottom and large HTGRs of extended exposure to primary and secondary coolant environments.

The components to be examined are:

- (1) ducting,
- (2) metallic thermal barrier materials,
- (3) valve internals,
- (4) circulator components,
- (5) steam generator.

The metallurgical examinations to be performed on these components will comprise detailed documentation of surface appearance, identification of surface products, metallographic examination, and determination of mechanical properties. The mechanical property evaluations will include (where possible) tensile, creep rupture, fatigue, and impact data.

Observed changes will be extrapolated to the lifetime of a large plant using structural design methods developed in HTGR research and development programs and elsewhere.

B.7.3 Component removal schedule and objectives of individual component examinations

Concentric duct

After the regular ducting sections are removed, the T-section of concentric duct including the hot valve at steam generator in the same loop would be removed. Figure indicates the

proposed locations for cutting free this ducting section and other areas described below. This is the only section of hot duct that is readily accessible, the other sections between the reactor and the hot valve being closely surrounded by thick concrete. Removal of this section provides access to the steam generator, and radiation monitors can be inserted several feet past the hot valve in the direction of the core. The section of concentric duct would be monitored internally at the site and then removed from the secondary containment for shipment to ORNL. Examination at ORNL and subsequently at GAC would include analyses for distributions of fission products, evaluations of corrosion products on metallic surfaces, observations of erosion, etc., in the area of impinging gases in the T-section and examination for crevice corrosion and other phenomena in the Solami insulation.

Hot valve internals

The hot valve is also removed with the concentric duct T-section. After removal, the top and bottom heads of the valve will be unbolted and put aside, and the remaining valve internals including the thermowells removed. All these parts are readily accessible. Evaluation of the hot valve would include determination of plateout profiles and deposits of both the ducting and the thermowells, investigations for wear and galling on moving parts, evaluations of surface corrosion products on the ducting and gate sections, and examinations for crevice corrosion, etc., in the 304 stainless steel insulation.

Cold ducting

In addition to the cold ducting knee removed with the concentric ducting, several additional samples will be removed. These include: a portion of the cold return duct from the steam generator to include the first and second knees (see Fig. B.2) and about 80 circular sections at least several inches in diameter trepanned from 20 locations in both loops. During removal of these sections, gamma and beta monitors will be inserted into piping to obtain activity profiles to compare directly with the end-of-life external gamma survey. The samples will then be sent to ORNL and GAC for further investigations of plateout profiles and distribution.

Helium circulator and components

The impeller, bearings, and drive assembly of the helium compressors in the main loop are readily accessible; they can be unbolted from

their housing in the main circuit after lifting the removable concrete plates and wall sections. One impeller plus its bearings should be removed and shipped to ORNL where plateout examination will be performed. Pieces of the turning vanes associated with the circulator will also be removed for metallurgical examination. The impeller will be inspected for wear, erosion, surface corrosion products, and ease of decontamination. The bearings will be examined for wear, oil leakage, and resultant blowby on the impeller shaft.

Steam generator tubesheet and tube bundle

The most difficult item for removal, yet the one that will yield the most information relative to design and performance of large HTGRs, is the steam generator tubesheet and tube bundle. Detailed evaluation has revealed that it should be possible to remove (after constructing a "clean room" on the refueling floor) the entire tube bundle, shroud, tube sheet, and channel head as one unit by sectioning the steam generator shell at the top, freeing the shroud from the concentric duct and shell, and lifting the bundle out into what appears to be sufficient space above (see Fig. B.4).

During removal, an internal gamma and beta survey would be conducted by dropping monitors inside the shell.

Examination of the tubesheet and bundle constitutes perhaps the most important item in the Peach Bottom postmortem tests. Initial examinations at ORNL would consist of determination of plateout profiles on metallic surfaces for all important condensable fission products as functions of temperature and position in the steam generator. Samples of representative weld joints and sections of tubing from the economizer, evaporator, and superheater sections would then be cut out for mechanical, metallographic, and other tests. The latter samples would be chosen from low, medium, and high radioactive parts of each U-tube and in areas of interaction with restraining plates, etc. The impingement baffle and other gas baffles would be examined for deposits and samples cut out for metallurgical examination. The tubesheet could be examined and incidence of stress corrosion cracking would be investigated.

Mechanical tests on metallic components, particularly the Incoloy 800 tubing which is also common to Fort St. Vrain and large HTGR steam generators, are proposed both at ORNL and GAC. These tests would include tensile, creep, fatigue, and stress relaxation tests. Metallography for degree of oxidation or carburization would be conducted, and some corrosion tests might also be done. Hydrogen and tritium permeability tests on sections of tubing and blowdown tests on tubing sections are intended.

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Fig. B.4. Steam generator cross section.