July 9, 1981

DRAFT

LONG RANGE RESEARCH PLAN

FY 83-87

SUPPLEMENT TO NUREG 0740

FOR

GAS-COOLED REACTOR SAFETY RESEARCH

DRAFT

TABLE OF CONTENTS

TABLE OF CONTENT (continued)

LONG RANGE RESEARCH PLAN SUPPLEMENT
GAS-COOLED REACTOR SAFETY RESEARCH

1. Introduction

An alternative but as yet unmatured gas-cooled power reactor industry proposes to produce gas-cooled reactors as an alternative to uranium-plutonium fueled light-water reactors in the U.S. In the event that the reactors are to be deployed in the U.S. (and NRC is now given to understand that application for a new plant is a possibility) it will be necessary for the NRC to be prepared to perform its licensing functions.

The following long-range plans represent a reasonable estimate of NRC research efforts anticipated for support of NRC regulation of gas-cooled reactors. The plans provided are representative of current thinking and research directions, yet conjectural, considering that during the first two years a comprehensive data needs re-analysis and research re-planning effort would take place so as to focus on concerns from detailed review.

1.1 Background

The development of the current U.S. concept in gas-cooled reactors, the High Temperature Gas-Cooled Reactor (HTGR) follows nearly 40 years of gas-cooled reactor operating experience accumulated in Great Britain, France, West Germany, and the USA. The U.S. version differs in a number of respects over the earlier foreign reactors.

The 40 Mw(e) Peach Bottom HTGR with its high on-line availability quite successfully produced electricity for Philadelphia Electric for 7-1/2 years beginning in 1961 and has provided valuable practical operating data on the concept. That U.S. licensed plant has been followed by the licensed 330 Mw(e) Fort St. Vrain HTGR plant which began initial operations in 1975, and is currently providing additional valuable data from operations. These plants were to have been followed by the Summit and Fulton full sized HTGR plants for which the NRC had nearly completed initial licensing reviews at the time the projects were withdrawn.

The above history, together with NRC reviews requested by the DOE for HTGR proposals under the nonproliferation program, the series of Licensing Topical Reports (LTR) being submitted to the NRC by the principal U.S. HTGR vendor, and the repeated requests by GAO and the U.S. Congress for NRC to conduct preapplication reviews and safety reseach have kept the NRC somewhat current with progress of the potential HTGR industry. NRC research personnel have been keeping current on the latest anticipated HTGR concept for possible near term application as described in GCRA High Temperature Gas-Cooled Reactor Application Study,* December 1980 from which a number of the following dates are derived. The schedule which is assumed for this plan is:

1981 ASSUMED MILESTONES FOR HTGR COMMERCIALIZATION

NRC initial HTGR safety research program planned	1974
NRC research program initiated	1975
Fort St. Vrain demonstration plant operational	1978
Initial safety codes operational	1979
Lead standard plant (LSP) docketed (assumed)	1982
PSAR Effort Initiated	1983
Requirements for large-scale proof test determined	1984
PSAR Submitted	1985
Construction Permit granted, LSP construction initiated	1987
Safety code models tested	1989-90
FSAR Submitted	1990-91
LSP operating license issued	1993
Large-scale proof tests completed (as required)	1993

After the cancellation of the large HTGR plant orders by electric utility companies, the NRC's HTGR safety research program was reduced in scope and funding level.

*Gas-cooled Reactor Associates (GCRA) publication

In recent years, however, the advanced reactor safety research program has been authorized repeatedly by Congress to provide the NRC with tools to license advanced reactors effectively at the time of their commercial introduction, rather than experience the sort of information gaps which characterized the early water reactor licensing phases. A 10- to 12-year program is anticipated, and hence the current plan carries the program for HTGRs well past its midpoint.

If, as assumed, the HTGR concept is again introduced by a utility in the near future, major elements of the HTGR safety research program, including most major proof tests ultimately judged necessary, should be completed by the time the lead plant receives an operating license. Most major research is and will be carried out by DOE and industry; only that safety research that only NRC can do is to be carried out by NRC.

1.2 Overall Objectives

The objectives of NRC HTGR Safety Research are to provide an independent data base and methodology for timely NRC licensing and standards operations to accomodate the anticipated re-entry of U.S. industry into the commercial HTGR power generation field and the current needs to discharge the NRC's responsibility for the operating Ft. St. Vrain HTGR. This plan is issued as a supplement to the USNRC's Office of Nuclear Regulatory Research "Long Range Research Plan FY1983-1987," NUREG 0740, March 1981.

- o Evaluate the status and applicability of the Accident Initiation and Progression Analysis (AIPA) study produced in the mid-70s as related to the currently proposed HTGR concept and perform whatever additional probabilistic risk assessment should now be required to carefully identify accident potentials and needs for safety research.

- o Develop a basic understanding of the processes modeled by key accident computer codes and the implications with respect to accident consequences.

- o Verify through separate effects and integrated out-of-reactor tests (where DOE/industry testing does not satisfy NRC needs) the range of adequacy of models used in the key accident computer codes. (example: CHAP, NONSAP-C)

- o Develop detailed designs for whatever proof tests, not provided by DOE/industry, are needed for large-scale code validation.

- o Bring the codes to a state of readiness for use by licensing bodies.

- o Determine the effects of design choices and operating limits on plant safety and thus review systems to improve reactor safety.

- o Determine the adequacy of graphites as structural materials, including the effects of oxidation.

- o Appraise the performance of improved and specialized instruments for inservice inspection of components and structures within the PCRV.

- o Determine the adequacy of metallic materials related to the integrity of the primary coolant system.

- o Determine the appropriate safety criteria for liner and thermal barrier integrity and liner cooling including redundancy considerations.

2.0 Gas-Cooled Reactor Safety Research

The goal of the HTGR safety research planning is to be prepared for licensing the next HTGR plant without significant unresolved issues affecting the process, to assist NRR in developing safety criteria for HTGR's, to provide whatever guides and standards as may be appropriate on a timely basis, and to extend the scope of NRC rules to include the HTGR.

Although the NRC is aware of the DOE/industry proposed "HTGR-SC/C" cogeneration unit concept, the long-range plan considers mainly generic safety issues at this time, with some anticipation of the resolution by DOE and industry of fuel cycle options and detailed design choices. It is assumed that the next, lead plant will not be of the very high temperature operation design and thus operating temperatures up to 700° to 750°C will be considered for the next several years, leaving the 900°C range as required for higher process heat designs to a later time frame. Design-specific issues will be identified in the next two fiscal years, beginning with the preapplication review process and the use of probabilistic risk assessment techniques, in order to provide the necessary information for licensing decisions in the projected time frame. As these design-specific issues are identified (e.g., inservice inspection requirements for thermal barrier, prestressed concrete reactor vessel (PCRV) closure design, containment requirements, etc.) the research technical objectives will be focused more sharply.

Assuming the HTGR Safety Research program continues at NRC the progress is expected to follow the general approach outlined here. The effect of continuing the program on the minimum maintenance level basis, however, as it has been for the last several years tends to distort the rates of progress of various tasks which cannot all be pursued in parallel. A more complete program along the following lines is scoped out briefly in the following sections.

Specific needs of the NRC HTGR safety research program are as follows:

- o Developing licensing review bases appropriate to HTGRs considering general design and siting criteria, assessment of basic standards, etc.

- o Define design basis for testings required for HTGR mechanical and electrical equipment qualifications and fire protection.

- o Bring accident delineation studies to the point where the risks of low-probability accidents are better established.

These needs are generally responsive to the initiatives expressed by the U.S. Congress (which are dealt with much more specifically below) as given in the NRC Authorizations Committee Report No. 97-22 dated April 10, 1981. The activities also recognize the extensive research undertaken by DOE and the industry, and coordination with DOE and industry researchers is continually exercised to maximize the utility of information available to NRC. Further, many research needs are more properly met by DOE or industry, and the NRC seeks to minimize sponsorship of extensive programs.

The program also recognizes the five-year plan stated efforts by the Electric Power Research Institute (EPRI) (p. 198 of the 1981-85 Research Plan) wherein an evaluation of alternative reactor concepts including the HTGR is to be reported (Key Event No. 26) in mid-1982.

Over the years the Advisory Committee on Reactor Safeguards (ACRS) has made comments on NRC's advanced reactor safety research program plans. These comments are recognized in the preparation of this Long Range Research Plan Supplement.

This plan has also included those items requested by NRR in the June 23, 1981 memo from NRR (H. R. Denton) to RES (R. B. Minogue): Long-Range Plan for Gas-Cooled Reactor Safety Programs. That memo states the view that "NRC-sponsored HTGR research to support licensing needs should have the following broad objectives:

"1. Support the development and assessment of design criteria acceptable for licensing;

"2. Assure that safety and accident analyses and experiments performed by applicants are consistent with regulatory requirements. This would include review and evaluation of computer codes for licensing acceptability; and

"3. Support and organize the development of a base of physical data, computer codes, and design and engineering information, so that the technical bases for licensing HTGRs is clear. This could result in a NUREG document enumerating, for example, helium properties, graphite properties, fuel properties, inherent safety features, designed safety features, codes and standards, applicable computer codes, low probability accidents with analyses and mitigation devices, and risk assessment analyses. Such information should allow a comparison of HTGR features and corresponding LWR features, so that persons experienced in LWR safety can readily compare LWR and HTGR features in reviewing generic issues common to both reactor types."

Section 2.0 is organized into seven categories as follows:

Development of Licensing Review Bases
Analysis
Materials Interactions
Radiological Release and Transport
Structural Integrity
Equipment Qualification and Fire Protection
Test Facilities

A number of the safety research items of concern cut across several of the above categories. For example, the concern for sufficient cooling of the PCRV during a severe core overheating transient translates to concerns for thermal barrier adequacy and structural integrity, liner cooling system functional adequacy and structural integrity, liner-PCRV penetration closure design structural integrity metal/concrete interface effects, and liner cooling redundancy criteria development.

The section on Development of Licensing Review Bases describes plans in RES to assist in pre-application review and development of licensing criteria, some of which would normally be carried out by NRR if their resources were not currently required to be totally focused on LWR licensing.

The Analysis area of the HTGR safety research program is central to program accomplishment in that it serves to coordinate and synthesize the phenomenological research conducted in other areas and provides the means of making the developed information available to the NRC Licensing staff in a useful fashion. Probabalistic risk assessment techniques are employed in this area, as well as deterministic techniques, to bring into focus needed research in other areas.

The Materials Interaction area includes interaction of coolant and its impurities with primary system materials, the interaction of core materials with each other at high temperatures and the interactions of primary system effluents with secondary containment atmosphere and structures. The Radiological Release and Transport area investigates the integrity of fuel, the release of fission products from the fuel and through the reactor materials, and the plateout and lift-off of fission products throughout the system and into and within containment. The Structural Integrity area encompasses properties of HTGR-specific reactor materials, failure mechanisms for primary system components, and structural integrity of the reactor, the PCRV, and the containment.

The Equipment Qualification and Fire Protection area includes research efforts to define criteria for electrical and mechanical equipment qualifications for operability under accident conditions and for mitigation of fire hazard conditions and results.

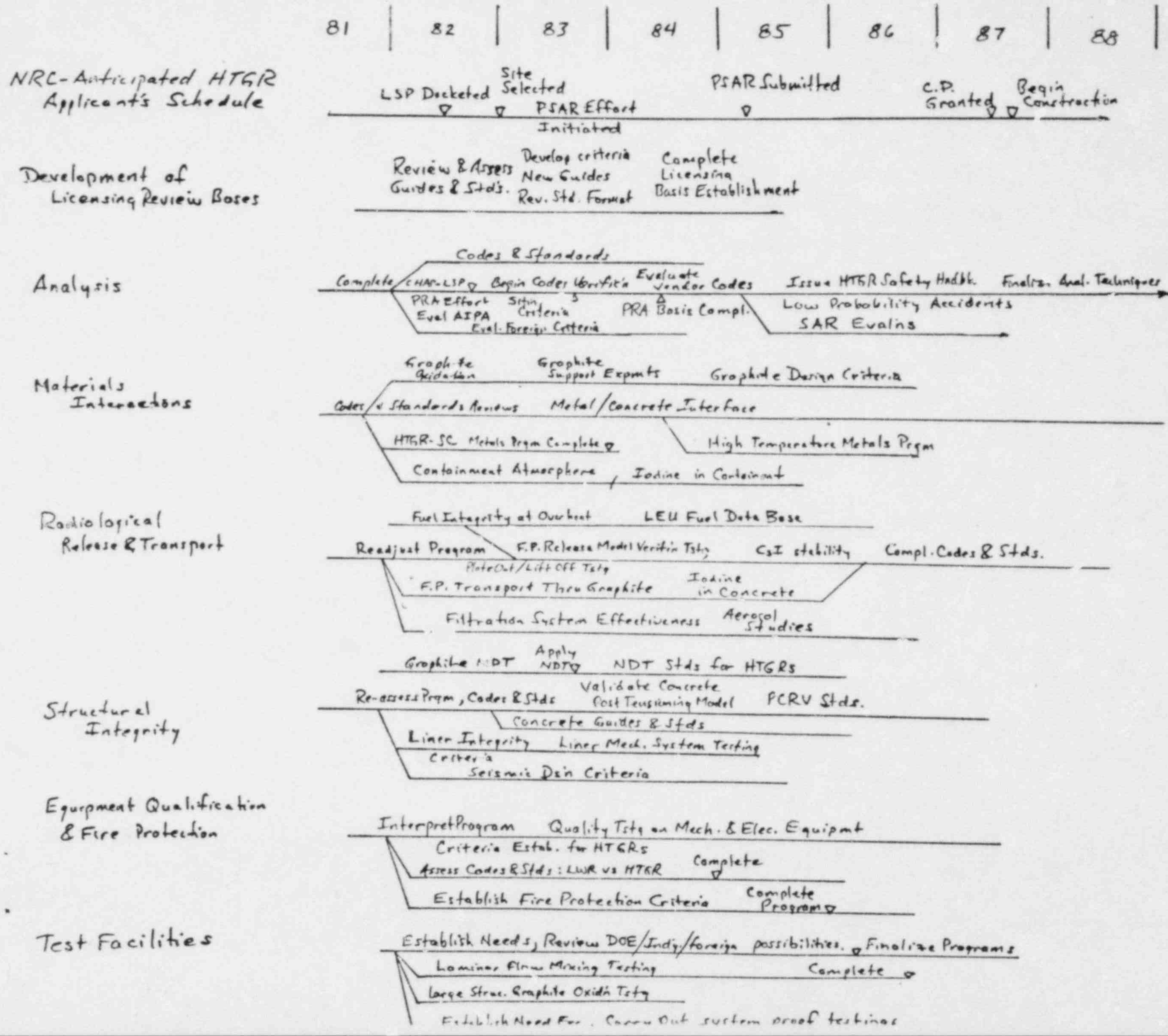
The Test Facilities area, which involves the acquisition and use of any experimental equipment or test rig costing more than $0.5 million, may have the most dramatic increase in activity; it starts from essentially zero. It is also the area in which the greatest potential for major increases in scope and cost are found.

A chart showing the development of the research efforts as compared with the HTGR commercialization time table given above is shown in Figure 1. This chart will be significantly modified, as discussed further below, when the lead plant is identified and when more detailed research planning has taken place which will recognize progress to date in all the research fields compared to the reevaluated needs of NRR.

Being Developed

Figure 1

HTGR SAFETY RESEARCH PLAN



2.1 Development of Licensing Review Bases

This section provides a basis to assist NRR with development of preapplication review plans and development of RES input to licensing criteria such as:

- Regulatory Guides
- Codes and Standards
- Design and Siting Criteria
- Recommended changes to Standard Format and Content of Safety Analysis Reports (SARs)

2.1.1 Regulatory Objective

The objective of this program area is to provide a focus on the development of basic licensing criteria for HTGRs at an as early as possible stage to assure efficient use of U.S.N.R.C. resources involved with the evaluation and licensing of the next HTGR plant.

2.1.2 Technical Capabilities Required

This effort requires the review of existing licensing policy and its adaptation or translation as necessary to HTGR designs and situations. It further requires the capability to assess the need for, or severity of, proposed impositions of safety requirements on HTGR designs and proposed sitings.

2.1.3 Status of Capabilities

The Office of Research (RES), including the recent addition of the personnel and resources from the previous Office of Standards Development, and its contractors have been involved with licensing criteria and standards development and have been engaged at a low level in researching HTGR-specific licensing issues for some time. Thus, RES is prepared to address this effort for NRC.

In July of 1973 a draft HTGR Edition of the Standard Format and Contract of SARs for Nuclear Power Plants was issued. The General Atomic Company has submitted a number of Licensing Topic Reports (LTRs) over the last several years with the intent of alerting the NRC to potential issues and possible analyses or evaluations for HTGR-specific situations. Included with the group of reports are, for example:

GA-A15697	Evaluation of Proposed German Safety Criteria for High Temperature Gas-Cooled Reactors (May 1980)
GA-A16077	Licensing Topical Report: Applicability of Division 2 Regulatory Guides to High Temperature Gas-Cooled Reactors (December 1980)

The NRC has licensed the Ft. St. Vrain HTGR to proceed with its power escalation to the 100% level. There have been fairly thorough licensing reviews of the previously considered Summit and Fulton HTGR projects, and the partial review of a standard plant HTGR (GASSAR). A joint NRR-RES paper entitled, "HTGR Postulated Accidents and Safety Research Needs in the United States" prepared in November 1978, and documentation NRR and RES prepared in connection with the NASAP project (Non-Proliferation Alternative Systems Assessment Program) in January 1980 further illustrate the status of HTGR issue understandings.

As part of the NASAP study, NRC submitted to DOE a list of 29 questions and comments on eight "safety issues" concerning the safety and licensing documentation for the proposed commercial steam cycle HTGR design. The detailed questions for each issue are presented in the report High Temperature Gas-Cooled Reactors, Preliminary Safety and Environmental Information Document, Volume IV, NASAP, U.S. Department of Energy, January 1980, DOE/NE-0003/4.

Efforts already ongoing in this licensing criteria development area include NRR technical assistance programs recently initiated at ORNL and LANL in anticipation of the renewal of the HTGR licensing process. In April of 1981, NRR requested the technical assistance from ORNL in reviewing the applicability of the TMI-2 Action Plan (NUREG-0660) requirements to Fort St. Vrain and HTGRs. The requirements set forth in NUREG-0660 and NUREG-0737, "Clarification of TMI Action Plan Requirements" were developed for LWRs, with the realization that some would be applicable only in part, or not at all, to HTGRs. ORNL will assist in determining which of the TMI items are appropriate to HTGRs. In May of 1981, NRR requested technical assistance from LANL in reviewing Fort St. Vrain related items and their applicability to HTGRs. The following five topics have developed as a result of Fort St. Vrain operating experience:

1. Plateout Probe Data
2. Inservice Inspection
3. Component Hot Spots
4. Graphite Structural Analysis
5. Review of Rules and Regulations

The results of these probrams, particularly Item 5 of the LANL program, will be used in the development of HTGR licensing criteria.

2.1.4 Research Program Objectives

The objectives of the research efforts are to define a program in coordination with NRR to develop appropriate guides and standards and determine design and siting criteria as needed for NRC to successfully carry out its licensing functions for the next HTGR plant.

2.1.5 Research Program Plan

Beginning with the 1973 draft of the HTGR Edition of the Standard Format for SARs and the existing codes and standards, guides and criteria these items will be reviewed and assessed for modification appropriate to currently understood HTGR technology. As an example, the reviews of Reg. Guides 1.7 (for Combustible Gas Concentrations) and 1.120 (Fire Protection) are appropriate to this effort and are further referred to in Sections 2.3 and 2.6 of this plan. Consideration will be given to recent developments in foreign HTGR licensing insofar as it may provide useful perspective for U.S. licensing criteria (e.g., "Evaluation of Proposed German Safety Criteria for High Temperature Gas-Cooled Reactors," A. Barsell, GA-A15697, May 1980).

Plans will be laid for cooperation with industry in standards development such as for ASTM standards and ASME code Sections III and XI as related to HTGR technology.

In addition to the general effort categories described above certain issues need resolution for clear delineation of NRC design and siting criteria and will be focused on as follows:

- All currently identified safety issues will be reviewed for their impact on established NRC licensing ground-rules (e.g., Regulatory Guides, General Design Criteria). For example, such issues as core seismic criteria, graphite criteria and performance models, etc.

- Applicability of NRC proposed siting policy task force recommendations (NUREG-06125) and the proposed new siting rule.

- The development of containment requirements considering fuel form (coating type), a spectrum of depressurization accidents, and the potential for low probability accidents, including the generation and control of combustible gases.

- Criteria for inservice inspection of structural graphites, and of components and structures within and including the primary system boundary (Investigation, development and potential utilization of improved inspection techniques and instrumentation will be considered.) Should the inspection detail required be equivalent to that for LWRs (e.g., full inspection of the thermal barrier/liner and steam generator tubes?

- Regulatory criteria for PCRV liner cooling capacity and redundancy

- Potential necessity to require emergency core cooling by natural convection as a mitigating feature for the unrestricted core heatup accident.

- What constitutes the appropriate site suitability source term for HTGRs?

- In the long range should criteria now be applied which clarify or minimize eventual issues in decommissioning (e.g., plans or criteria for disintegration and disposal of a PCRV)?

All of the above will be coordinated with efforts ongoing under Section 2.2, below, where probabilistic risk assessment is also employed to focus on safety issues of concern.

Results:

FY82 Review and assessment will begin of guides and standards and the standard Format of SAR's. Criteria development needs will be determined. Safety issues of most criteria concern to NRC plans and licensing capability/research development will be addressed.

FY83 Appropriate guides and standards work as identified in FY82 will be ongoing. Remaining issues for determination of NRC approach will be resolved.

FY84 and Beyond Initial programs on licensing review and methodology development will be completed and remaining efforts will be scheduled for completion within the anticipated time frame required for HTGR licensing evaluations.

2.2 Analysis

The analysis area serves to coordinate and synthesize the phenomenological and component research conducted in the other areas and provides a means of organizing the results and making the developed information available to the licensing staff in a useful manner. The interactions of accident transient consequences research and the impact on criteria, codes, and standards are developed in this section.

Activities in the analysis area also focus on the development, testing, and use of computer codes that describe:

- System transients under normal and accident conditions--to the point of a significant change in geometry.
- Sustained loss of forced circulation.
- Primary system depressurization when coupled with another process, such as moisture ingress or loss of circulation.

One of the functions of the analysis area that is closely associated with the plant transient response evaluation is the delineation of all potentially significant accident scenarios to identify processes and phenomena requiring further investigation. Particular attention has been paid to this responsibility already and will continue through the program. Use of probabilistic risk assessment (PRA) techniques will be employed to better focus residual safety research needs; new analytical methods will be required to extend current PRA techniques to HTGRs.

2.2.1 Regulatory Objective

Adequate computer codes for evaluating the safety of nuclear power plants are needed within the NRC. The TMI-2 Lessons Learned Task Force identified the need for analysis techniques to evaluate specific potential accident transients and to assess consequences of operatc errors. It is recommended that RES perform analyses of selected transients using the best available computer codes, assess the adequacy of these codes for NRC use, develop or improve codes when necessary, and compare the results with results from methods used by the industry. The Task Force also recommends verification of the analytical methods by comparison with actual reactor operational data and with appropriate test results.

Probabalistic risk assessment is planned to identify important accident sequences and their consequences and time available for emergency action and to identify and help resolve important regulatory issues such as whether alternative siting criteria are appropriate for HTGRs.

2.2.2 Technical Capabilities Required

Development, assessment and verification of analysis techniques for evaluation of HTGR plant transients and for structural response of components requires a thorough going understanding of the plant systems and their interactions, access to and ability to utilize the various analysis techniques so far developed, the capability to program new techniques, and the ability to obtain and use appropriate operational data for verification. Appropriate operational or test data must be carefully selected in order to (1) determine particular code calculational accuracy "bandwidths" in predicting test case results and (2) in determining in which of the various HTGR situations the codes are accurately useful.

Additionally, it is necessary to establish the degree of analytical accuracy which must be acceptable to the NRC to determine whether further code developments, in each particular area, are warranted.

Sufficient capability to carry out useful probabilistic risk assessments is also required.

2.2.3 Status of Capabilities

Analytical model development has already produced a large number of codes useful in plant transient analysis. A selection of these codes is listed as follows.

CHAP	whole plant system analysis
CORTAP	dynamic simulation of HTGR core
LARC	fission product release
NONSAP-C	three dim concrete trans. behavior
ORECA	emergency cooling analysis of HTGR core
ORTAP	dynamic simulation of HTGR transients
SUVIUS	behavior of fission gasses in HTGR coolant

There are many other current HTGR analysis codes available for safety and licensing assessments; the Brookhaven National Laboratory (BNL) code library now lists more than 50 of them. For most of the codes, efforts continue on their updating and on their validation.

In the probabilistic risk analysis area, a several year study was undertaken by General Atomic (GA) for the DOE in the mid-70's entitled the Accident Initiation and Progression Analysis (AIPA). This study was useful in studying specific accident scenarios and identifying safety issues for the most worthwhile safety research. Although the study was useful in providing guidance for research, it is now somewhat obsolete because of improvements in PRA methodology and the current need to apply PRA to what is now considered the most likely next HTGR design to be submitted for licensing.

2.2.4 Research Program Objectives

The objectives of this analysis program are to identify and develop or verify the data and methods necessary to allow the NRC to assess the level of protection to the public health and safety from operation of a gas-cooled reactor facility. This encompasses development, verification and application of systems analysis and component analysis methods and the application probabilistic risk assessment techniques to HTGR safety evaluations.

2.2.5 Research Program Plan

2.2.5.1 Systems Analysis

The system transient analysis efforts make use of the CHAP code which is under development at the Los Alamos Scientific Laboratory and ORTAP, CORTAP and ORECA which are under development at Oak Ridge National Laboratory. The low level of support the program has endured through most of its life has required postponing the development of the more detailed, sophisticated, and realistic models for such phenomena as helium gas dynamics, which will be required for a realistic prediction of transient thermal conditions external to the core. During the plan the emphasis in CHAP development will be on the refinement of models to permit more realistic predictions and on the verification testing of component models.

Each of the currently available codes will be reevaluated for applicability to the NRC's needs with the objective of eliminating any unnecessary duplication and utilizing the better portions of developed codes as appropriate.

The CHAP code will also be used in conjunction with other codes such as ORTAP and ORECA to identify and develop additional computational capabilities required for accident consequence evaluation appropriate to a probabilistic analysis of HTGR safety.

The analysis of a sustained loss of forced circulation event beyond the point of fuel-coating failure and the fuel carbide melting point(s) is expected to require a special version of CHAP that incorporates the significant phenomena being investigated in the materials interactions area. (Depending on the resolution of criteria for natural convection discussed in Section 2.1, this analysis may be done for a natural convection emergency cooling system.) Development of that core model will be required at an early date and will be based on existing experimental fuel failure data. Subsequent revision is anticipated as additional and improved experimental data becomes available. The use of auxiliary codes to evaluate the deformation of out-of-reactor components will also be required. It is anticipated that, when such deformations or displacements become large enough to influence flow distributions, an interaction with a modified CHAP representation will be necessary.

Because of potentially high temperatures of the helium coolant in accident situations, it is necessary to establish time and temperature relationships for all critical components of the primary system as well as for the fuel during emergency core cooling conditions. Continued investigations of convective flow mixing and natural convection phenomena, including hot streaks, stratification and plumes effects, are needed for design dependent cases. Means for benchmarking analytical techniques will be established. Improved understanding of conditions relating to the transition between laminar and turbulent flows must be developed.

Human factors engineering studies will be initiated which may include task and system analysis dealing with control room design, training and staffing requirements, procedures development, and establishment of simulator needs.

When accident sequence analyses, PRA, and code evaluations and development are well underway, studies will begin on potentials for low probability accidents with the study of consequences possibly more severe than the design basis accidents. probability accidents include core drop, control rod ejection, simultaneous moisture ingress with reactor depressurization, rapid depressurization of the PCRV, and unrestricted core heatup. Research supporting the study of these accidents will largely be design specific but some work will be identified early. Initial efforts probabilistic methodology to assess which low probability accidents, if any, should be considered in the design of mitigation systems. Studies of protective system instrumentation will be considered in this area.

Ongoing and planned LMFBR and LWR safety research will be surveyed for applicable work and plans made for its utilization in this progrm.

Review, and coordination as useful, will be made with safety research and evaluations performed by other countries, especially in Germany and Japan, and in particular, to more carefully identify information of use to U.S. programs and to identify information gaps which may be investigated either in the U.S. or abroad.

2.2.5.2 Component Analysis

A computer code system for analyzing the behavior of the containment system during depressurization events coupled with another process that produces combustible gases will be developed. A preliminary version is scheduled to be available when the research program is restarted from its minimum level. It will be used extensively in the development of an experimental test program that will probably identify inadequacies in the initial modeling. Following the development of improved models and their incorporation into the analysis system, an analysis program in support of an experimental verification program will be conducted. To the extent feasible, NRC will utilize verification data as may be obtained from DOE experiments.

A detailed assessment will be made of thermal barrier and liner cooling system requirements. This will encompass the heat loads under "normal accident" and degraded accident conditions, the required performance of the materials, the materials' interfaces and penetrations, and the functional requirements of the system including redundancies. Results of the assessment will be utilized in planning in other sections of the overall program as appropriate.

The analysis of component structural response involves a variety of both conventional and unusual considerations. In the essentially conventional analysis of metallic and ceramic components (steam generators, control drives and guides, PCRV insulation cover plates and core-support graphites) the principal issues are associated with the characterization of materials in the temperature and chemical environment of the HTGR and with the development of useful damage and failure criteria.

The unusual features of the HTGR that affect the structural analysis requirements include th prismatic element core, whose seismic response must be judged to be acceptable, and the PCRV, whose design margins and failure modes are not will established. Both of these topics have received significant attention. The shortage of funds, however, has severely limited the experimental phase of the effort. During the plan, the analysis effort will support an expanded experimental program, use the output of that program to improve the analysis as required, and support the planning for verification tests.

Independent reviews will be carried out for NRC evaluation of safety analysis codes developed and used by the reactor vendor such as:

RATSAM	Thermodynamic system response
OXIDE	Air/Water graphite oxidation
PADLOC	Plate-out of fission products
SORS	Fission product release from core

Specific attention will be given to the evaluation of existing natural circulation codes for the analysis of adequacy of cooling and coolant mixing under such conditions within the PCRV. Verification testing for the code(s) is anticipated to confirm the usefulness to NRC.

2.2.5.3 Probabalistic Risk Analysis

Early efforts to prioritize the safety issues and identify research needs utilized PRA methods in the AIPA study. That study and some German work in the PRA area will be reviewed and applied to the currently anticipated lead HTGR plant. New information from this survey of important accident sequences and safety issues will be factored into the licensing evaluations and the planning for detailed risk analysis. The information will also be used to guide the development of the entire research program plan.

2.2.5.4 HTGR Safety Handbook

An HTGR Safety Handbook will be developed which will draw on the great deal of fundamental information being generated by the Fort St. Vrain project important to the support of long-term Fort St. Vrain operations and of generic application to advanced HTGRs. The handbook contents will be selected from operational data and from supporting research studies data and analytical techniques for documentation in an organized and concise manner. The development of the structure of the handbook will be aided by an HTGR edition of Reg. Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants.

2.2.5.5 Results

FY82 Initiate evaluations of existing PRA and AIPA work to define further program data. PRA comparison between LWR and HTGR safety to establish and evaluate HTGR levels of protection. Complete CHAP version for Lead HTGR Plant. Layout evaluation program for research and safety developments in other countries. Initiate system and component code accident sequence analysis (SASA) as required to support PRA. Initiate development of an HTGR safety handbook. Initiate codes and standards work anticipating licensing needs for early application reviews.

FY83 Perform major PRA effort for lead plant siting evaluation (considering in particular the November 1980 vendor report: GA-A16084* in response to NUREG 0625). Complete CHAP verification efforts using full power operating data from FSV. Establish simulator needs. Restart evaluation of vendor safety analysis codes.

*A. Borsell, et al., "Reactor Siting Risk Comparisons Related to Recommendations of NUREG-0625," General Atomic Company, November 1980.

FY84 Complete the updated more thorough PRA effort defining issues for licensing review and for eventual closing out of HTGR reactor safety research. Provide a risk-basis for recommending whether alternative siting criteria are appropriate for HTGRs. Develop analysis techniques as required to satisfy the isolated needs. Outline verification needs. Complete evaluation of vendor safety analyses.

FY85 Complete improvements to systems and component codes reflecting the results of the PRA. Perform verification tests on the analysis techniques. Study low probability accident potentials and consequences.

FY86 Finalize the HTGR safety evaluation codes and techniques. Perform evaluations of PSAR.

FY87 Complete PSAR evaluations.

2.3 Materials Interactions

This program area deals with the investigation of:

- The interaction of primary coolant impurities such as moisture and oxygen with primary-system materials.
- The interaction of reactor and core materials with each other at very high temperatures.
- The interaction of primary-system effluents with the containment atmosphere and structure.

Coolant impurity interactions involve two major areas: the oxidation of graphites and the corrosion or other modification of metals. The investigation of graphite oxidation--whether by oxygen, water, carbon dioxide, or some other agent--involves both small-scale (sample) studies directed toward understanding the microscopic mechanisms and large-scale studies that deliberately incorporate the effects of sample geometry.

The investigation of corrosion or other modifications of metals generally involves controlled atmosphere testing of metals long term structural properties on small-scale samples.

2.3.1 Regulatory Objective

The objective of this program area is to assure development and availability of an adequate data base for NRC evaluations of the effects of normal and abnormal HTGR environments over lifetime on the structural enveloping materials used in the HTGR reactor systems. Development of criteria and standards as appropriate is a corollary objective.

2.3.2 Technical Capabilities Required

For licensing evaluation and analysis of the HTGR, NRC requires basic data pertaining to the physical and chemical characteristics of the special materials of construction of HTGR reactors. Reliable information must be obtained from testing that closely simulates reactor conditions under normal operation, transients and postulated accident conditions.

Specifically, data are needed on the thermal and pure and impure coolant chemistry effects on core and structural graphites and on vessel cavity, thermal barrier and liner materials and primary system component metals.

2.3.3 Status of Capabilities

Excellent progress is being achieved in the pursuit of the required capabilities in this area. Through the years since the HTGR safety research program began, the abilities at the laboratories have grown in defining and carrying out testing programs specifically suited to obtaining data for NRC and for exploring behavior of the materials. In parallel with and supplementary to the broad extensive work for DOE by GA and Oak Ridge National Laboratory (ORNL), the Brookhaven Lab has developed in-depth evaluations of critical parameters.

A review of relevant material and physical property data from the literature has been conducted in an effort to identify the most important areas for safety-related research.

The investigation of the effect of coolant impurities on the properties of structural materials has produced important confirmation and extension of available data. The graphite oxidation studies, for example, have already shown that for catalyzed oxidation by moisture, such as takes place in "PGX" graphite, the reaction products increase the reaction rate rather than inhibiting it as hydrogen does in purer graphites.

Basic understandings have been developed in graphite microstructure, impurity effects on oxidation, resulting fracture properties, effects of stress on properties, and various strength loss mechanisms.

There had been very little data on the mechanical behavior of metallic materials exposed to high temperature helium coolants as found in a steam cycle HTGR. This was even more true for the materials exposed to the impurities in the coolant, including H_2, H_2O, CO, CO_2, CH_4 formed when moisture interacts at high temperatures with metal and graphite surfaces and fission products such as I_2, Te and Cs. Long term creep and fatique data have been developed for the HTGR conditions of interest and have greatly reduced the uncertainties for high cycle fatique in Co-1 Mo Steel, Type 304SS, Incaloy 800H and Hastelloy X.

In regard to containment atmosphere effects, studies of the interaction of coolant impurities with the secondary containment environment during depressurization events have provided at least partial quantification of flammability concerns and resulting additional loads on the containment structure.

2.3.4 Research Program Objectives

The objectives of this program are to provide research support to NRC in the following areas:

- o The corrosion and mass-transport interactions of metals and graphite with air, steam, and other helium impurities.
- o Rapid graphite oxidation and potential combustion hazards caused by the accidental ingress of air and/or steam and the potential formation of combustible gas mixtures in the containment after a depressurization accident.
- o The mechanical properties of metals, graphites, PCRV, and other materials in a prototypic and abnormal helium environment.

- The characteristic effects at metal/concrete interfaces.

- The chemical and physical behavior of core materials under conditions of sustained loss of forced circulation.

2.3.5 Research Program Plan

The research plans are designed to meet the above objectives within a time frame adequate for needed licensing evaluations. The plans also incorporate specific anticipated needs from the licensing staff and address issues further identified by the 97th Congress in its Report 97-22 on NRC Authorizations for FY82-83. To the extent practical, cooperation will be resumed with foreign countries to enhance the NRC research product. Early attention will be given to the assessment and improvement of codes and standards for NRC use.

2.3.5.1 Graphite

Further graphite material characterizations will be carried out, concentrating on thermal oxidation of graphite, graphite oxidation profiles, mass transport in graphite, and mechanisms for strength losses in graphite. Consideration will be given to the potential for effects on core graphite due to possible concrete decomposition products. The current program has been restricted to small-scale studies, and the extent of those has been drastically limited by budgets. During the revived program the small-scale program will be continued and modestly expanded. A major effort will be undertaken in large sample testing where the effects of geometry, flow rate, pressure, stress, and temperature will be examined. Oxidant levels simulating the ones experienced in Ft. St. Vrain will be applied to the large samples over long-term exposures.

Studies of high temperature mass transport of gases in HTGR graphites will address the determination of effective diffusion coefficients of ternary mixtures of gasses (e.g., H_2, CO, H_2O) in several HTGR candidate graphites.

Development or modification of codes and standards will be undertaken as appropriate for graphites when sufficient data becomes available. Plans will be made early to determine these data needs.

2.3.5.2 Metals

Investigation of the effect of impure helium environments on primary-system metals has been a significant component of the HTGR safety research program since its inception. It is planned to complete most of these long-term studies of fatigue, creep, and creep/fatigue interaction in the high-temperature range appropriate to HTGR application but to extend some of the studies to include the effect of fission-product interaction with the metals.

The primary focus is on creep and fatigue of Incaloy 800H and Hastelloy X. Experiments on oxidation and carburization kinetics will be restarted in the HIL loop to continue previously budget-cut-reduced development of models of gas-metal reactions. Considerations will be given to wear and adhesion properties.

Particular consideration will be given to the adequacy of the thermal barrier and liner materials, thermal barrier and liner attachment characteristics and interface effects with concrete. Consideration will also be given to development of any needed criteria for nil-ductility temperature (NDT) effects on the materials, considering the appropriate epithermal neutron flux. This work is coordinated with the Section 2.5, Structural Integrity work.

2.3.5.3 Metals/Concrete Compatibility

Concerns of particular significance to the HTGR will be evaluated and needs for any program work to resolve issues will be developed. Of concern are the liner/concrete and anchor interfaces and other primary system components, as they may indicate some safety significance over the long term.

2.3.5.4 Containment Atmosphere Effects

Early accident-delineation studies identified two aspects of the loss-of-forced-circulation accident, about which very little was known and it was possible at least to postulate scenarios with substantially more severe consequences than the scenarios proposed by the vendor. These were the behavior of the fuel above its melting point and the potential for imposing substantial additional loads on the containment as a result of the production of combustible gases. Analytical and experimental studies of both areas were started and then cutback because of budget constraints; they are expected to require substantial expansion during the plan. Applicability or modifications will be considered for Regulatory Guide 1.7, Control of Combustible Gas Concentration in Containment Following a LOCA.

Studies will also be made of the chemical compound formations and the retention of released iodine and other key fission products in the PCRV and in containment during core heating.

2.3.5.5 Results

FY82 The graphite and metals programs will be rescoped by comparing accomplishments to date with earlier versions, objectives and reidentifying remaining objectives and necessary timing. The focus of thermal oxidation and mass transport on graphites in the existing reactor will be augmented to include the candidate materials for the new lead reactor. Containment atmosphere studies will be replanned. Needs for codes and standards work will be developed.

FY83 A large portion of the existing metals program will be complete and determination will have been made on priorities for any additional creep/ fatigue testing. Graphite property characterizations will continue as defined from previous year requirements. Modeling of long-term core support structures confirmatory experiments will be initiated. High temperature nickel alloy materials properties will be assessed for adequacies in applications to reactor internals components, thermal barriers and ducts. Metal/concrete interface effects will be evaluated. Containment atmosphere mixing and impurity/ flammability studies will be reviewed.

FY84 Specific results to be obtained in this and remaining years on an anticipated heavy program will be defined in greater depth in FY82. Efforts focused toward requirements to fill out data needs for codes and standards for core support structures will require major efforts.

FY85 Studies this year and beyond will include the forgoing efforts plus extension of materials characterizations for high temperature applications beyond that required for the lead HTGR-SC toward the higher temperatures for the future reformer HTGR objectives.

FY86 Efforts will continue as discussed above.

FY87 Termination of many of the long term programs will be accompanied by extensive data analysis for application to the final licensing approvals required for a construction permit.

2.4 Radiological Release and Transport

This area of the HTGR safety research program investigates:

- The integrity of the first fission-product containment barrier (the individual fuel particle coatings).
- The release of fission products from fuel particles with both intact and failed coatings.
- The transport of fission products through graphite and into the primary coolant.
- The adsorption and desorption of fission products from component surfaces and into and within the containment.
- The significance of graphite aerosols as a transport vehicle for fission products.
- The effectiveness of primary and secondary cleanup and filtration systems.

2.4.1 Regulatory Objective

The general objective of the radiological release and transport safety research is to accumulate a sufficient data base and associated analysis techniques to allow accurate licensing determinations of the factors contributing to the on-site and off-site radiation doses anticipated for the various postulated accident scenarios for the HTGR.

2.4.2 Technical Capabilities Required

Design bases and safety limits have evolved for HTGR fuel that are not analogous to those used for LWRs. Fuel failure mechanisms and their relationship to design bases and safety limits involve failure phenomena which must be examined statistically over wide ranges of time and temperature. For a thorough understanding and ability to define the offsite source term for postulated HTGR accident scenarios a comprehensive set of validated data and analyses techniques are required covering fuel particle integrity and behavior, release and transport of fission products from fuel and through graphite, the mechanisms for plate-out and lift-off of fission products from various reactor and within containment surfaces, the effectiveness of various transport mechanisms including HTGR-specific aerosols, and the efficacy of filter systems. In each case sufficient background data are required to enable confident assessment of projected licensee conditions, and, sufficiently accurate analysis models and techniques to enable confident evaluations by the NRC.

2.4.3 Status of Capabilities

A significant body of data on materials interactions during unrestrained heatup accidents has been obtained. The observation that oxide fuels are converted to the dicarbide with the release of carbon monoxide in the vicinity of the carbide melting point and the observation that the dicarbide is quite mobile in graphite at temperatures above its melting point are expected to have a significant effect on the evaluation of events involving sustained loss of forced cooling.

A thermochromatographic apparatus has been constructed at Brookhaven to study the chemical state and distribution of simulated iodine-, cesium-, and strontium-bearing fission products in HTGR primary-system environments. A series of experiments has been ongoing with the facility. Capabilities have also been developed to carry out significant research in the following areas:

- Coated fuel particle failure and release rates
- Iodine permeability and retentivity in concrete
- Measurement of fission product diffusion in HTGR media
- Diffusion of thorium and uranium
- Aerosol formation mechanisms in HTGRs

Fission product migration at high temperatures

Sublimination of graphite

Transport of Nonvolatile Fission Products through grapite

Chemical analysis of fission products from thermally failed fuel

A Materials Test Loop (MTL) has been built, tested, and put into operation at Brookhaven to provide various normal and abnormal HTGR environments for materials samples. The Helium Impurities Loop (HIL) was renovated and is being used to study gas-phase reactions with various surface materials.

A furnace system has been designed and assembled at Los Alamos to measure failure rates and the time-dependent release of Kr-85 from irradiated fuel particles. Preliminary experiments were run on single particles of both the BISO and the TRISO types before the budget cutback interrupted the experiments.

The LARC-1 and LARC-2 computer codes for calculating the time-dependent release of fission products from an HTGR core during the LOFC accident, the SUVIUS computer code for determining circulating activity in the primary coolant loop has been completed, and the LEAF code for calculating the time-dependent release of fission products from a reactor containment building with a containment cleanup system has been completed.

Technology exists for in-pile phenomenological testing in the ACRR (Sandia test reactor) for obtaining data if needed on HTGR fission product release mechanisms and rates for severely overheated fuels.

Significant and comprehensive data sets have been produced by DOE and the industry in the areas of fission product release and transport and such data are carefully considered by NRC safety researchers in conducting the complimentary NRC programs. Cognizance is also given to international developments although budgetary restraints have precluded significant cooperation and data gathering in recent years.

Generic aerosol behavior codes such as HAARM and QUICK, as well as applicable technology produced in the LMFBR and LWR programs, are available for evaluation of aerosol activities in the HTGR.

2.4.4 Research Program Objectives

Current objectives of the research program are to assure the adequate development of independent capability for NRC assessment of radiological release and transport in HTGR applications through a reanalysis of the current extant capabilities compared with newly defined regulatory needs. These needs are redefined in light of the TMI-2 accident impact on NRC and in consideration of the operating experience to date of the Ft. St. Vrain reactor. Development of remaining needed data and methodology should be sufficient for licensing appraisal of the postulated source terms.

2.4.5 Research Program Plans

Research program plans are formulated to provide the necessary codes and standards, data and methodology for the above objectives within the time frame required for licensing evaluations for the construction permit and operating license. The plans incorporate specific areas to meet needs anticipated from the licensing staff and are consistent with guidelines outlined in Authorization Report 97-22 of U.S. Congress. Generally, data exist for normal operating ranges, these programs focus on high temperature overheating situations.

Careful consideration is being given to existing and ongoing DOE/industry research and to obtaining useful data through international cooperation.

2.4.5.1 Fuel Integrity and Fission Product Release

Consideration is being given to in-pile testing of fuel particle retention and release of fission products. Specific tests will be designed for verification of the data base and the performance of the HTGR fuel, and particularly the fuel particle coating integrity at high (accident induced) temperatures, potentially to be run in a facility such as Sandia's ACRR.

Confirmation of the vendor fuel-failure and fission-product-release models has not been feasible at previous funding levels. The revived program involves the fabrication, irradiation, and subsequent testing of capsules containing reference fuel samples under both steady-state and transient conditions. A series of four instrumented capsules is expected to provide sufficient information unless some significant and unexpected observation is made. This information will also be used in verifying some of the models in the LARC computer code. Additional data will be provided for NRC's release model (NUREG-0111). Additional plans will be prepared for the NRC evaluation of and data base preparation of LEU fuel anticipated for HTGR application. This will be followed by fission product transport studies with LEU fuel to reassess the data bases for releases, transport, plate-out and lift-off.

2.4.5.2 Fission Product Transport Through Reactor Materials

The investigation of fission-product transport in graphite must be performed with well-characterized graphite and with a near-prototypic combination of fission-product species. In-reactor studies are not believed to be necessary. Some significant information pertinent to normal operating temperatures should become available from post-irradiation examination (PIE) of Fort St. Vrain test fuel assemblies. Studies to date have been limited to a few species.

Evaluations will be made of the more recently developed vendor models and codes for analysis of fission product transport.

Initial experiments have been performed with cesium iodide tagged with Cs-137 and I-131. The vaporization and deposition are being studied to characterize the behavior of CsI under HTGR conditions. (Recent microprobe studies of fission-product distribution in irradiated GCR fuel particles indicate the possible formation of CsI in the porous pyrocarbon layers. In addition, the stability of CsI in contaminated HTGR coolant streams (helium containing H_2O, Co, Co_2, etc.) and its interaction with various metallic and ceramic surfaces in HTGRs need further definition.) We will consider additional reactions of cesium iodide (with graphite, metals, Kaowool, etc.) with respect to their pertinence to various accident sequences.

2.4.5.3 Fission Product Plate-Out and Lift-Off

An understanding of fission-product adsorption and desorption on primary- and secondary-system surfaces as a function of temperature, surface condition, etc., is essential to a realistic prediction of accident consequences. These phenomena are incorporated into a liftoff and plateout models in the fission-product code SUVIUS, and good data are needed for model development and verification. The scope of the current program in this area is very limited. This program will be expanded to include additional surface materials. Permeability and retentivity studies of iodine in stressed concrete had been studied some before funding cuts and will be required. Design Basis Accident No. 2, "Rapid Depressurization" will be reviewed for how fission product lift-off is enhanced and how the consequences affect building habitability and off-site doses. Isothermal studies of the adsorption and desorption of cesium, iodine, and strontium on steels, Incoloy 800, and silica in helium and in helium mixed with water vapor will also be resumed.

Careful coordination will be maintained with the studies under DOE initiatives.

2.4.5.4 Aerosols and Filtration Systems

Initial research is planned to identify possible sources of aerosol formation in HTGR accidents, with emphasis on the understanding of the aerosol formation mechanisms (vaporization of nuclear materials at high temperatures followed by condensation, soot formation accompanying graphite oxidation at rapid rates, and liftoff from metallic surfaces).

Experiments will be initiated to investigate the interaction of volatile fission products with airborne particulates and to study the rate and extent of fission-product adsorption on aerosol particles as a function of aerosol material and concentration.

It is projected that the investigation of the high-temperature materials interactions associated with the sustained loss-of-forced circulation accident will quantify the preliminary indications of graphite aerosol formation and their effectiveness in serving as condensation centers for condensable fission products. Under these conditions, an investigation of aerosol properties to characterize its agglomeration characteristics will be required. Adaption of the aerosol transport analysis methods developed under the LMFBR program such as HAARM and QUICK, mentioned earlier would then be appropriate.

Effectiveness of primary and secondary cleanup and filtration systems under abnormal conditions will be evaluated to determine any necessity for further NRC research and standards development peculiar to HTGR's in this area.

2.4.5.5 Results

FY82 Review the status of all radiological release and transport efforts as defined in this section and as compared to remaining data and methodology needs for licensing (including those for codes and standards) and adjust detailed plans for later fiscal years as necessary. Continue ongoing fission product and uranium/thorium transport and plate out data and model development.

FY83 Restart fuel particle integrity and fission product release studies to obtain verification data for fission product release models, including plans for in-reactor testing. Further investigate the stability of CsI in contaminated coolant streams and the interaction with metallic and ceramic surfaces. Restart program on permeability of iodine in stressed concrete. Redefine program on effectiveness of cleanup and filtration systems. Continue transport studies in graphite and coolant.

FY84 As appropriate, layout and begin the program on the data base preparations for LEU fuel including releases, transport, plant-out, lift-off. Carry out the in-reactor testing of fuel particle fission product retention and release. Continue program on iodine retentivity in stressed concrete. Study the effectiveness of graphite aerosols as condensation centers for condensable fission products. Carry out filter and cleanup system abnormal operation effectiveness studies.

FY85 Continue all programs in the category with objective of completing as many as possible, fulfilling required data needs, by FY87. Reassess actual remaining needs and plan the completion of the release and transport work and the integration of results into appropriate guides and standards within the time frame required for licensing activities.

FY86 For this and remaining fiscal years continue the programs as required by FY85 analysis through to completion.

2.5 Structural Integrity

The structural integrity area includes the investigation of:

- The failure mechanism related properties of HTGR-specific reactor materials
- Failure mechanisms for the metallic and ceramic components of the primary system.
- Development of criteria for inservice inspection and testing of structural graphites and other HTGR-specific components and structures.
- Structural integrity of the reactor, the steam generators, the PCRV and its barriers and liner systems, and the containment under seismic and accident induced conditions.

2.5.1 Regulatory Objective

The regulatory objectives for the structural integrity area of gas reactors safety research are that the research define or outline the needed material properties, failure mechanisms, inspection methods and structural analysis techniques peculiar to HTGR's sufficient for adequate licensing reviews for the next HTGR plant to be proposed.

2.5.2 Technical Capabilities Required

A comprehensive data base is required by the NRC for the properties and failure mechanisms of the various HTGR structural materials to enable accurate assessment of vendor claims for reactor structural integrity recognizing that HTGR materials are exposed to substantially higher temperatures and temperature differentials than those in water-cooled reactors and are subject to different corrosion conditions. Similarly the NRC requires accurate and sufficiently conservative techniques for evaluating the integrity of the structures and the postulated failure modes under abnormal and seismic events.

2.5.3 Status of Capabilities

Programs at the laboratories performing HTGR safety research for NRC have provided a great deal of useful materials property data and have developed useful techniques for evaluating strength and component integrity as a means of checking vendor claims of safety. Moreover the laboratories have developed the capability to carry out the significant remaining research efforts which will be required in the program.

Safety code provisions have been developed for pressure boundary integrity but have not yet been reviewed by the NRC for acceptance.

NONSAP-C, a finite-element code for the structural analysis of reinforced concrete structures under static, dynamic, and long-term loads has been prepared at LANL for confirmation of the seismic safety analysis techniques being used. Experimental and analytic seismic modeling techniques have also been developed under the HTGR safety research program at LANL.

Techniques are available at LANL to evaluate fatigue failures of graphite under cyclic loading, fractures caused by small crack growth, effects of secondary stresses on graphite strength, and other considerations in graphite behavior.

A nondestructive examination technique is under development of Pacific Northwest Laboratory to monitor strength charges due to oxidation of core support graphite.

2.5.4 Research Program Objectives

It is the intent of the research programs in this area to identify and provide the necessary material property data and structural analysis techniques for development of necessary standards and guides and for the development of credible evaluation techniques for NRC licensing reviews.

2.5.5 Research Program Plan

The graphites, metals, and insulator materials whose interaction with primary-system impurities has been studied under the materials interaction area are considered here from the standpoint of the failure criteria and failure mechanisms involved and the effect of the interactions on those mechanisms and criteria. This is a continuing effort that has received too little attention to date. The efforts in this area will be expanded.

Research involving primary system integrity must consider not only working helium conditions but also the potentially high helium temperatures that could result under emergency core cooling conditions. Previous research results indicate that in addition to temperature, impurities in the helium also have significant effects on material integrity.

Topics to be considered in primary system integrity research include PCRV structure, liner and thermal barrier, penetration closure design, and the liner cooling systems. Design and selection criteria must be developed using the ASME Code (Section III, Division I) if metal closures are to be considered. Code requirements and limitations will be considered as they apply to high temperature metals that may be used within, or which serve as a portion of, the primary system boundary.

The seismic design experience and practice for light water reactors is only generally applicable to the HTGR core and core supports because of the many differences in core and core support arrangements and in the structural design criteria for graphite. Substantial progress was made on this topic at the time of the Summit/Fulton/GASSAR reviews. The seismic capability of the Ft. St. Vrain plant will not be taken as typical of HTGR capability because of the originally assumed low seismic activity area it was designed for. Evaluations will be made of what further design specific research is required to confirm the applicability of the existing seismic design computer codes and techniques.

The analysis of the core response to seismic excitation is needed to establish the conditions required to induce failure in prototypic fuel elements. Consideration will be given to continuation of small-scale or seismic tests to improve and test the core seismic analysis methods.

Material property and strength data for different grades of graphite particularly nuclear-grade graphites under HTGR operating conditions need to be developed. Testing efforts are to be planned to measure irradiation--induced creep and dimensional changes in isotropic graphite.

Development of techniques to measure strength changes in core support graphite will be completed for NRC and will be applied by DOE to the monitoring of core support graphite strength in the Ft. St. Vrain Reactor.

American Society for Testing Materials (ASTM) code cases do not cover design specifications for nuclear graphite components. Plans are to be made for NRC efforts to assure the timely development of the appropriate ASTM data. Particular attention is needed for graphite core supports (floor and posts) and thermal stresses.

More exact or more satisfactory theories for graphite behavior under multiaxial states of stress in graphite based on measured or given allowable primary stresses will be developed--an allowable stress criteria will be developed.

For concrete, it is planned to design and conduct a series of tests to confirm our ability to evaluate the effect of local anomalies in the concrete structures such as the PCRV and the containment. Also tests are planned for the verification of the NONSAP-C methodology for treatment of posttensioning of concrete structures.

Concrete degradation under abnormally high temperatures and/or in the presence of impurities such as fission products is a significant concern.

A review of safety codes for design of components and structures including review of the bases for their provisions will be performed to establish bases for their acceptance by NRC for generic application. A test program will be developed for obtaining additional data for concrete under triaxial stress as required for verifying the three-dimensional concrete constitutive models. Tests will also be designed to establish head capacity, off gassing, and spalling rates. More detailed assessment will be made of concrete constitutative laws (including effects of moisture, temperature, stress) and the results will be incorporated in the concrete behavior models and applied to the analysis of PCRVs and containments. Standards, where appropriate, will reflect the findings.

Criteria for the inspection of structural graphites, and of components and structures within and including the primary system boundary require confirmation and, as needed, the development of additional detail. Investigation, development and potential utilization of improved inspection techniques and instrumentation will be studied.

Further studies will be made to verify assumptions of containment failure modes and leak rates.

Special emphasis will be given to the development of tasks to focus on thermal barrier and liner cooling systems and PCRV/liner penetration integrity and failure modes and consequences. Data will be developed as necessary for appropriate standards.

Results

<u>FY82</u> The intermediate and longer-term goals for graphite, concrete and other HTGR materials data, codes and standards, and safety technology development will be reassessed to assure timely completions for NRC licensing needs. Software will be completed for core support graphite in - strength monitoring. Liner integrity and PCRV/liner penetration studies will be initiated.

FY83 Tests for validating NONSAP-C posttensioning models for concrete structures will be completed. Graphite material design specifications and standards work will be defined. Concrete degradation characterization efforts will be underway with specific goals identified. Application of graphite NDT methods will be confirmed. PCRV and containment failure modes programs will be underway. Additional studies needed to improve codes and standards, as identified in FY82, will be undertaken.

FY84 Efforts will be continued on concrete, graphite, liner, and other structural materials as developed in the earlier program redefinitions. Proposed changes to codes and standards will be developed.

FY85 Efforts continued further for this and remaining years as necessary for licensing goals for FY87 through FY93.

2.6 Equipment Qualifications and Fire Protection

Activities in the Equipment Qualifications and Fire Protection area include:

- o Electrical equipment survivability assessments under severe conditions.
- o Mechanical equipment functionability assessments under severe conditions.
- o Fire suppression and protection effectiveness under accident conditions.

2.6.1 Regulatory Objective

The objective is to provide a licensing basis for evaluation of reactor plant equipment for assurance that safety system functions will be maintained under design basis accidents and fire conditions.

2.6.2 Technical Capabilities Required

For licensing evaluation and analysis of HTGR equipment, NRC requires the ability to test and evaluate the reliability of electrical, mechanical, and safety equipment which would be needed under accident or severe transient conditions. Reliable information must be obtained from testing that closely simulates the normal and abnormal special HTGR conditions.

2.6.3 Status of Capabilities

Work performed at Sandia and underwriter laboratories has developed considerable capability and useful techniques for LWR equipment qualification and fire protection assessment. Much of the results of this work will be applicable to HTGRs and many of the testing techniques will also be applicable to any necessary HTGR equipment testing program.

An NRR equipment qualification program plan is being developed for LWRs with program completion targeted for FY84.

2.6.4 Research Program Objectives

The objectives of the research programs are to review the safety-related equipment of the HTGRs for reliability requirements and to identify any needs for testing technique or data development safety research peculiar to HTGR application so as to assure the adequacy of available evaluation means for NRC licensing efforts, and development of appropriate design criteria, codes, and standards.

2.6.5 Research Program Plan

Initial efforts in the research program will encompass review of HTGR plant designs and procedures to identify critical equipment and fire hazard situations and to determine where there may exist qualification and fire protection needs peculiar to HTGR reactor plants.

U.S. and foreign gas-cooled reactor equipment qualification experiences will be reviewed for applicability to the present situation. The applicability of LWR technology in these areas will be carefully assessed.

Definitions peculiar to HTGRs will be developed for design basis conditions for qualification requirements and for fire protection and suppression requirements. This will include thorough review of postulated accident environments to identify anticipated ranges for radiation, temperatures, seismic loadings, hydrogen and other gasses, etc., during the course of the accidents.

Existing or planned LWR codes, standards, and regulatory guides will be reviewed for applicability or needed changes for the HTGR conditions.

Fire protection safety research efforts will coordinate with data obtained and research planned under Section 2.3.5.3, Containment Atmosphere Effects (Materials Interactions). Research efforts will be directed to assure coordination with DOE/industry work to define reasonable and effective fire suppression alternatives.

Results:

FY82 Conduct literature searches and discussions with U.S. and foreign resources (e.g.: Peach Bottom, Ft. St. Vrain, Magnox, AGR) to establish baselines for expectations in qualifications and fire protection methods and criteria for NRC. Determine design bases events for equipment qualifications and design basis fire hazards and design basis hazards and conditions for which protection must be provided (e.g., suppression of graphite fires). Establish objectives for development of necessary codes, standards, and regulations. Begin determination of the nature of postulated accident environments.

FY83 Depending on plans established in FY82, the following efforts are anticipated. Research will focus on synergisms unique to the HTGR environments of helium, radiation, temperature and pressure, and on aging phenomena unique to the HTGR containment environment. Further determinations will be made of LWR technology applicable to HTGRs. Physically determine the spectrum of fire situations identified in FY82 including fire suppression alternatives, fire containment and mitigation, and plant effluent filtration or scrubbing.

FY84 Continuation and expansion if needed of efforts begun earlier. Complete guides and codes and standards work required for FY85 NRC licensing evaluations.

FY85 Complete the qualifications and fire protection programs.

2.7 Test Facilities

This program area includes the design, procurement, assembly, and shakedown of all major dedicated test facilities required. It also includes the design, fabrication, and operation of the test programs utilizing these facilities. However, it is expected that large test facility experiments will be carried out by DOE/industry to the maximum possible extent.

Although the distinction between the individual phenomenological research areas and the test facilities area has been arbitrarily defined in terms of hardware cost, it is based on the observation that the level of review and control that is appropriate to both the acquisition and the use of experimental facilities increases with their cost and complexity.

Investigations that have been initiated under the severely restricted phenomenological research areas during the initial program and are expected to require extension to large tests under more nearly prototypic conditions include the following:

- o Fission-product adsorption and desorption
- o Structural graphite oxidation
- o Laminar flow mixing
- o Containment mixing and stratification
- o Core seismic response

2.7.1 Regulatory Objective

The regulatory objective for the test facility section of gas cooled reactor safety research is that the facilities and planned testing are adequate in relation to other experimental program results to provide the data base required by NRC which may otherwise be unavailable.

2.7.2 Technical Capabilities Required

The fission-product investigations will require a large high-pressure system in which high flow rates of helium can be maintained and rapid depressurization can be simulated.

The graphite oxidation studies will need a high-pressure and high-temperature system that can accommodate approximately 1-foot samples in controlled-impurity flowing helium environments for long periods of time.

The laminar-flow mixing studies are expected to require an essentially full-scale simulation of a significant fraction of an inlet and/or outlet plenum that will operate over an intermediate pressure range in the HTGR temperature regime with controlled moderate helium flow rates. Confirmation of both flow distributions and heat-transfer rates is required.

The investigation of containment mixing and stratification is expected to require a series of large-scale tests to confirm the extrapolation from smaller phenomenological tests. These should involve an existing DOE facility, but major adaptation to supply the appropriate source of high-temperature helium and measure the composition and temperature as a function of position and time will require a substantial investment.

2.7.3 Status of Capabilities

The laboratories currently involved in HTGR safety research appear well suited to provide all parts of these capabilities when the time arrives for the initiation of the individual parts of the large test plans. No decisions, however, have been made at this time as to which facilities would be employed or at which location.

2.7.4 Research Program Objectives

The objectives in utilizing the large facilities for the HTGR experimental work is to assure the scaling-law effects are properly accounted for in the development of the data base and the verification of analytic capabilities to be used in licensing assessments. The facilities will be used to develop and confirm NRC benchmark data and evaluation models which cannot reliably be obtained from other sources. Further, the research objective is to develop and operate only theose facilities where it can be shown that use of existing facilities would not be possible or suitable to NRC objectives.

2.7.5 Research Program Plan

The core seismic response tests will make use of an existing shaker facility. It is anticipated that a major investment will be required in the fabrication of adequately instrumented fuel assembly models. This program will be designed as a verification of the confirmatory analysis procedure developed in the analysis and structural integrity areas.

All of these test programs, which are firmly projected, involve out-of-reactor facilities. Except as noted above, it is expected that the NRC will be required to fund them. There are, however, facilities in Europe that might be suitable for some programs. These possibilities will have to be evaluated carefully before decisions are made.

Results:

FY82 An indepth evaluation of test facility needs will be outlined this fiscal year consistent with above guidelines and the anticipated NRC reactor safety research needs.

FY83 The evaluation of test facility needs will be completed, considering all potential domestic and foreign resources, and the large facility test program will be outlined and initiated.

FY84 Any required testing programs will be conducted in this and future FYs as designated with emphasis on completion as early as possible and in particular in time for the lead standard plant (LSP) operating license issuance (assumed in FY93).

→ RETURN TO:
M. FRAZIER 1130-SS

PDR

DISTRIBUTION w/o enclosure

RBMinogue	Subj.
DFRoss	Circ
OEBassett	Chron
CNKelber	Branch R/F
WBeach	RFoulds
LHSullivan	RFoulds R/F
M. Silberberg	
S. Fabic	
R. Curtis	
F. Arsenault	
G. Arlotto	
K. Goller	
R. Bernero	
TEMurley, NRR	
GWKnighton, NRR	

PDR