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Advanced Reactor Research Plan 12/27/01

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I INTRODUCTION

In an information paper titled "Future Licensing and Inspection Readiness Assessment [FLIRA]," SECY-01-0188, dated September 17, 2001, the staff made a commitment to the Commission to develop an advanced reactor research plan to support the development of the NRC infrastructure necessary to ensure efficient and effective licensing reviews of future reactors. This document fulfills that commitment. The focus of the research plan is on those research activities associated with the reviews of two high temperature gas-cooled reactors (HTGRs) — Pebble Bed Modular Reactor and the GT-MHR, and two advanced light water-cooled reactors (ALWRs) — AP-1000 and IRIS, as well as items of a generic nature such as a new framework for future plant licensing. This plan represents an integrated look at research needs, including what others are doing in the international arena and where there are opportunities for future cooperation. Although this plan is directed toward research in support of NRC needs, it is recognized that not all of the work has to be done by NRC and that such information can be obtained through cooperation with others (both domestically and internationally) as well as through the work sone by developers. Accordingly, the costs involved in this or preliminary draft do not currently reflect the fact that some of this information can be obtained from others. Additionally, the focus of the plan is on the research needed over the next 5 years (FY 2002-2006).

From October 10 through 12, 2001, a workshop on high temperature gas-cooled reactor (HTGR) safety and research issues, was held at the Nuclear Regulatory Commission (NRC) headquarters in Rockville, MD. The highlights of the workshop were included in a report dated, December 27, 2001. The focus of this workshop was on identifying key HTGR safety issues and the need for future research, including independent tools and data that the NRC would need to develop to support licensing reviews of new HTGR designs. The workshop discussions and information developed on important HTGR safety issues, research needs, and priorities were useful to the staff in identifying major safety areas where confirmatory and anticipatory research is needed. These insights served as an important input to development of this research plan to guide NRC's future advanced reactor research program. The workshop discussions identified several opportunities for international cooperative research which will be followed upon and the NRC will continue to draw upon the existing domestic and international experience.

In developing this research plan, the staff's objective was to identify the tools and data that will be needed to support an efficient and effective licensing process, without regard to budget constraints. This is consistent with the approach taken in the FLIRA report. The priorities described in the report can then be used as an input in the development of specific research plans and resource needs.

A comprehensive HTGR research program is crucial for the NRC to establish an information basis for conducting an effective licensing process. A sound research program that includes both anticipatory and confirmatory research would:

- improve the staff's knowledge and quality, and effectiveness and efficiency of reviews.
- explore margins in safety
- provide capability for independent confirmation of applicant claims.

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II. SCOPE

The staff has developed this research plan to address reserach needs in key safety areas to support the pre-application and licensing reviews of four designs, using the industry projections as outlined in an Nuclear Energy Institute (NEI) letter to the Commission, dated, August 10, 2001. This research plan focuses on the four near-term deployment plants, which include two HTGRs — PBMR and the GT-MHR, and two ALWRs — the Westinghouse AP-1000 and IRIS designs. The PBMR pre-application review is ongoing and is expected to continue through Spring 2003. Exelon, the PBMR applicant is expected to soon follow with an application for a Combined Operating License (COL) under 10 CFR Part 52. General Atomics, the GT-MHR applicant, held a preliminary meeting with the staff in December 2001, expressing an intent to request a pre-application review in the near future. The AP-1000 pre-application review is underway and is expected to last through the early part of 2002. Westinghouse also has plans to request staff's review of the IRIS design as some testing possibly beginning in FY 2003. The time frame covered by this research plan is 5 years (FY 2001 through 2006). It is recognized that some of the research programs can last longer than this time period, but given the projected schedule for the above plants, much of the work should be done in this time period to support the regulatory process.

The scope of this research plan includes both confirmatory and anticipatory research. It addresses the need for generating pertinent experimental data as well as developing analytical tools for the NRC to be able to independently and objectively review the licensee submittals. It is envisioned that some of the thermal-hydraulic and severe accident codes that have been traditionally used for the LWRs, can be used for the HTGRs, with some modifications. For some purposes, it may be necessary to develop new codes. Also included in the research plan, is the need to develop a generic regulatory framework that is technology-neutral, to ensure an efficient licensing review process. Throughout this report in the staff's assessment of research needs, due consideration is given to risk implications.

III. DISCUSSION

This advanced reactor research plan includes the anticipatory and confirmatory research needed to effectively support the future advanced reactor licensing efforts. The plan is organized as follows: There are two generic sections. Section III.A deals with development of a generic, technology-neutral regulatory framework, and III.B discusses the human factors aspects, including human reliability considerations, for both reactor classes . The rest of the material in the Discussion section is divided into two parts: Section III.C contains details for the HTGRs (PBMR and GT-MHR) and Section III.D contains the details for the ALWRs (AP-100 and IRIS). In both these sections, the following issues are addressed for the two reactor classes in various sub-sections: (1) thermal-fluid dynamics; (2) severe accidents, including source term; (3) fuel performance and qualification; (4) nuclear analysis (including reactor neutronics, decay heat generation, radionuclide inventories, radiation Shielding, criticality safety); (5) high-temperature materials; (6) instrumentation and controls (I&C); and (7) probabilistic risk assessment (PRA). In the HTGR-related discussions, the generic statements pertain to both the PBMR and the GT-MHR. Any design–specific issues are individually mentioned, as appropriate. The discussions on fuel performance and qualification (Section III.C.3) and on nuclear analysis (Section III.C.4) are also generic to the HTGRs, with specific reference to either the pebbles or the fuel compacts, as relevant. Similarly, Section IV.D.3 and IV.D.4 address the ALWR counterparts.

In each of the subsections, for both reactor classes, the following items are addressed: (a) description issue(s); (b) risk perspective; (c) related NRC research; (d) related international research; (e) NRC research objectives and plans; (f) resources and schedules; and (g) priorities. The staff has used past experience in pre-application reviews of the MHGTR and the AP-600 designs in arriving at the cost and schedule estimates. The schedule estimates were made in consideration of the industry projections for pre-applications and applications contained in NEI's August 2001 letter.

III-A REGULATORY FRAMEWORK AND INFRASTRUCTURE DEVELOPMENT

a. Description of Issues

A regulatory framework and infrastructure is needed that can be applied to license and regulate advanced reactors. This framework and infrastructure is needed because while the NRC has over 40 years of licensing and regulating nuclear power plants, this experience (e.g., regulations, regulatory guidance, policies and practices) has been focused on current light water-cooled reactors (LWRs) and has limited applicability to advanced reactors. There will be design and operational issues associated with the advanced reactors that are distinctly different and with no relevance to the LWRs. However, NRC LWR experience can contribute and provide insights or "lessons learned." The most important insight is the development of a documented framework and infrastructure which would ensure that the licensing process (e.g., development of regulation governing the design and operation) is performed in a structured and systematic manner. This approach will ensure uniformity, consistency and defensibility in the development of the regulations for advanced reactors, particularly when addressing their unique design and operational issues. Further, the framework and infrastructure for current LWRs has evolved over 5 decades, and the bulk of this evolution occurred without the benefit of insights from probabilistic risk assessments ((PRAs) and severe accident research. It is anticipated that PRA will play a greater role in the licensing and regulation of advanced reactors and as such, the framework and infrastructure need to appropriately integrate PRA results and insights.

The proposed tasks would develop an approach (and ultimately a framework and infrastructure) that would be applicable to any advanced reactor concept. The approach would take full advantage of lessons learned from prior regulatory experience and assure an effective utilization of both deterministic and probabilistic methods in licensing and regulating advanced reactors. The approach would be applied to the PBMR, GT-MHR, AP-1000, and IRIS designs.

b. Risk Perspectives

It is expected that future licensees will rely on PRA and PRA insights as an integral part of their license applications. In addition, and more importantly, it is further expected that the regulations licensing these advanced reactors will be risk-informed. Therefore, both deterministic and probabilistic results and insights will be used in the development of the regulations governing these reactors. Consequently, a structured approach for a regulatory framework and infrastructure for advanced reactors that provides the guidance of how to use PRA results and insights will help ensure the safety of these reactors by focusing the regulations on where the risk is most likely.

c. NRC Research Objectives and Plans

Plan. NRC research efforts required to systematically develop a suitable framework and infrastructure for advanced reactor licensing and regulation will be carefully planned. As currently envisioned, the plan will include the major tasks discussed below.

Approach. An approach will be developed that is to be used to subsequently develop a framework and infrastructure. This approach will identify the scope and level of detail of the framework and infrastructure along with the boundary conditions, ground rules, assumptions, etc. that will be used in their development. Experience gained in NRC's Option 3 efforts to risk-inform regulatory requirements for current LWRs provides a starting point for the development of an appropriate regulatory framework for advanced reactors. The approach will include both qualitative and quantitative aspects as depicted in Figure 1. An important qualitative aspect of the approach is a hierarchal structure that supports regulatory goals including the goal of protecting public health and safety and the strategic performance goals of the NRC's Strategic Plan. It is anticipated that defense-in-depth will remain a guiding reactor safety strategy. In considering the entire plant life cycle and areas such as radiation protection, physical security, and safeguards, other strategies such as remote siting, walk-away safe, regulation to safety goals, risk minimization, and ALARA may also be applied. An important quantitative aspect of the approach is the development of useful risk guidelines for advanced reactors from the Safety Goal Policy Statement. Safety Goal issues that arise in developing the quantitative guidelines will have to be resolved.

Framework. A globally applicable framework will be developed for advanced reactors that includes PBMR, GT-MHR, AP-1000, and IRIS. The purpose of the framework is to develop a process (i.e., guidelines) that will be used to formulate a global set of regulations for advanced reactors. It is anticipated, however, that certain aspects of the framework may be specific to the type of reactor being considered. For example, for advanced gas-cooled reactors, the concept of core damage does not lead to useful risk metrics, so quantitative guidelines would probably differ from those proposed for advanced light water reactors. A key product of the framework will also include guidance regarding appropriate uses of strategies and tactics to compensate for uncertainties inherent in both deterministic and probabilistic safety analyses. Generally, these strategies and tactics will include the use of defense-in-depth and safety margin.

Infrastructure. The infrastructure required to support licensing and regulation of advanced reactors also needs to be developed. Personnel, training, facilities, computer codes, data bases, and other required capabilities need to be identified in advance considering projected uses of deterministic and probabilistic methods over the entire plant life cycle. It is important that a systematic approach to infrastructure development be taken in order to avoid unnecessary duplication of effort and assure the most effective utilization of resources in the long run.

Globally Applicable Regulatory Requirements. A set of regulatory requirements that are globally applicable to any reactor type will be developed. The framework will be used to identify and formulate what regulations are needed. The infrastructure will identify the tools needed to formulate these regulations. In addition, the body of existing regulations will be examined to identify globally applicable requirements. In some cases, LWR-specific requirements will be generalized to apply to other reactor types. In other cases, global requirements that are clearly met by existing LWRs may have to be explicitly stated.

Reactor-Specific Regulations/Regulatory Guides. As currently envisioned, as much reliance as possible would be placed on the use of regulatory guides rather than reactor-specific regulations to supplement global regulatory requirements. The framework will be applied to delineate needed reactor-specific regulations and regulatory guides. The development of reactor-specific regulatory requirements and guidance will be demonstrated for each of the four advanced reactor designs. These reactor-specific regulatory guides will not provide the detailed guidance for implementation of specific technical requirement, but will provide the guidelines for expanding the global (reactor neutral) regulations to account for reactor-specific considerations.

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Oversight/Peer Review. Considering the scope of the proposed effort and its potential impact on advanced reactor licensing and regulation, appropriate oversight and peer review is deemed essential. Arrangements for such reviews will be initiated during the planning task.

d. Related NRC Research

To Be Determined

e. Related International Cooperation

The South African government has issued a Basic Licensing Guide for the Pebble Bed Modular Reactor (Document LG-1037, Rev 0).

f. Resources and Schedule

Resources and schedule for the proposed tasks are summarized below. The overall duration of the effort is 4 years to support having the framework in place in time to support design certification application for the PBMR, GT-MHR and IRIS. The approach, framework, and infrastructure tasks, which are closely related, come first. Draft products for these tasks will be developed in the first year. In the next 2 years, a draft set of global regulations and reactor-specific regulatory requirements and guidelines are developed. The approach, framework, and infrastructure will be finalized early in the fourth year based on "lessons learned" from the draft regulations. With a final approach, framework and infrastructure, the global and reactor-specific regulations and guidelines will be finalized in the fourth year. Oversight/peer review meetings are scheduled at key milestones throughout the process to integrate as the work is performed rather than wait until the end. This direct interface is more efficient and effective. As noted by the schedule, this work is performed in an "iterative" manner, such that as knowledge is gained, it is continually fed into the process.

(g) Priority

This work is considered high priority since its outcome will have a large influence on the types of data, analyses and methods that applicants and the NRC will need for future plant licensing.

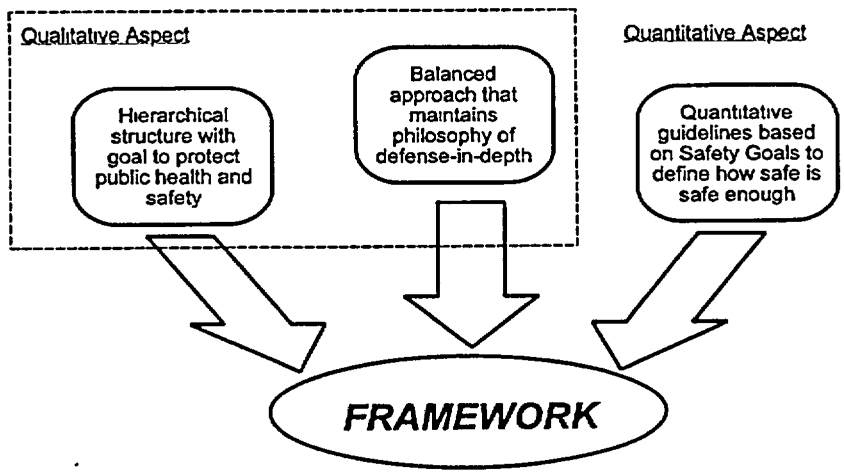

Figure 1 - Aspects of the Framework

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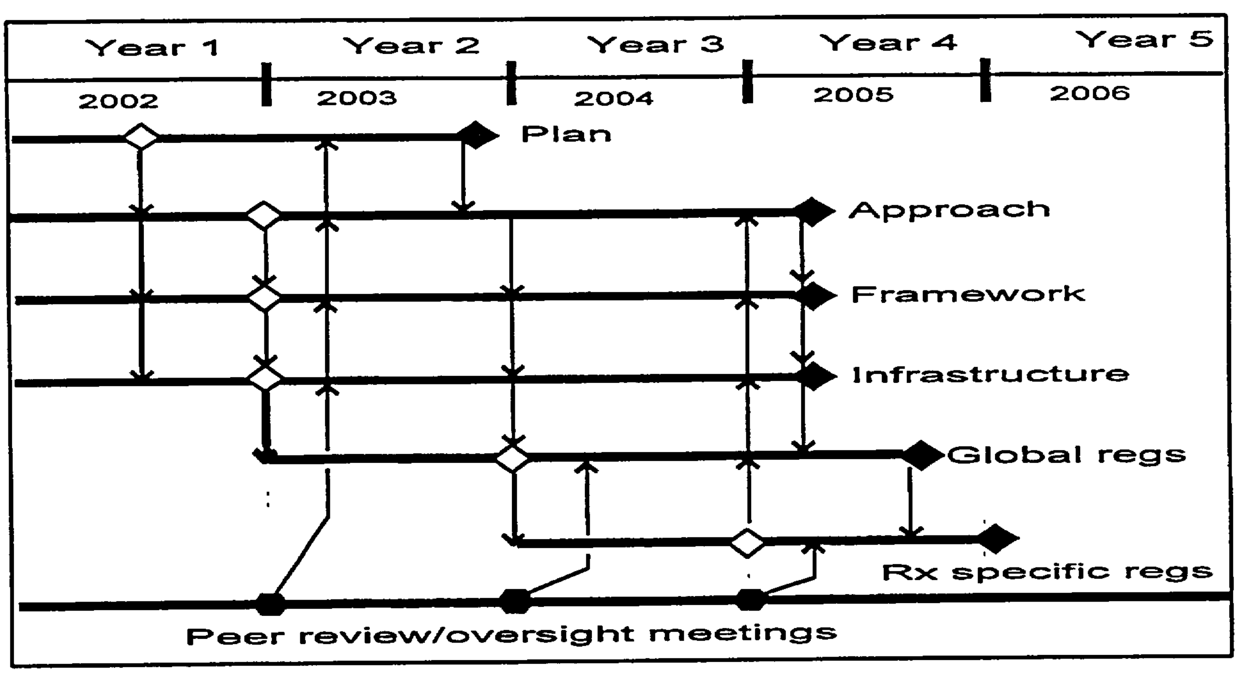
Resources

Task	Resources	
	Staff months	Dollars*
8. Develop plan	6	$150
9. Develop approach	9	$225
10. Develop Framework	18	$450
11. Develop Infrastructure	9	$225
12. Develop global set of regulations	24	$600
13. Develop considerations for reactor specific regulations/regulatory guides	48	$1200
Subtotal	114	2,850
Oversight/Peer Review**	36	$900
Project Management	15	$375
TOTAL	**165**	**$4,125k**

*Dollars are estimated assuming 1.0 staff month of an FTE costs $25k.
** 12 member team, 2 wks prep, 1 wk mtg, 1 wk report

Schedule



III. B HUMAN FACTORS/HUMAN RELIABILITY CONSIDERATIONS

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a. Description of Issue(s)

Nuclear power plant (NPP) personnel play a vital role in the productive, efficient, and safe generation of electric power. Operators monitor and control plant systems and components to ensure their proper functioning. Test and maintenance personnel help ensure that plant equipment is functioning properly and restore components when malfunctions occur.

In accordance with 10 CFR 50 and 52, the staff of the Nuclear Regulatory Commission reviews the human factors engineering (HFE) programs of applicants for construction permits, operating licenses, standard design certifications, combined licenses, and for licence amendments. The purpose of these reviews is to help ensure safety by verifying that accepted HFE practices and guidelines are incorporated into the applicant's HFE program. The review methodology (NUREG-0711, "Human factors Engineering Program Review Model" and SRP Chapters 13 and 18) provide a basis for performing reviews. The reviews address 12 elements of an HFE program: HFE Program Management, Operating Experience Review; Functional Requirements Analysis and Allocation, Task Analysis, Staffing, Human Reliability Analysis, Human-System Interface Design, Procedure Development, Training Program Development, Human Factors Verification and Validation, Design Implementation, and Human Performance Monitoring. See Figure III. B. 1.

The HFE aspects of the plant should be developed, designed, and evaluated on the basis of a structured systems analysis using accepted HFE principles. Therefore, the review method reflects a top-down approach for conducting an NRC safety evaluation so that the significance of individual topics may be seen in relationship to the high-level goal of plant safety. Top-down refers to an approach starting at the "top" with the plant's high-level mission goals and breaking them down into the functions necessary to achieve the goals. Functions are allocated to human and system resources and are split into tasks. Operator's tasks are analyzed for specifying the alarms, information, and controls that will be required to allow the operator to accomplish assigned functions. Tasks are arranged into meaningful jobs assigned to individual operators and the human-system interface (HSI) is designed to best support job performance. The detailed design (of the HSI, procedures, and training) is the "bottom" of the top-down process. The HFE safety evaluation is broad-based and includes normal- and emergency-operations, tests, and maintenance.

b. Risk Perspective

It is widely recognized that human actions that depart from or fail to achieve what should be done, can be important contributors to the risk associated with the operation of nuclear power plants. This recognition is based on actual operating experience and the results of PRA studies. The findings in NUREG/CR-6753, "Review of Findings for Human Performance Contributions to Risk in Operating Event," demonstrate that human performance has a significant impact on the risk from nuclear power generation. Specifically, operating experience indicates that latent failures resulting from deficient human performance in maintenance, testing, or work processes can impact equipment failure probabilities. Studies of PRA results found human error to be a

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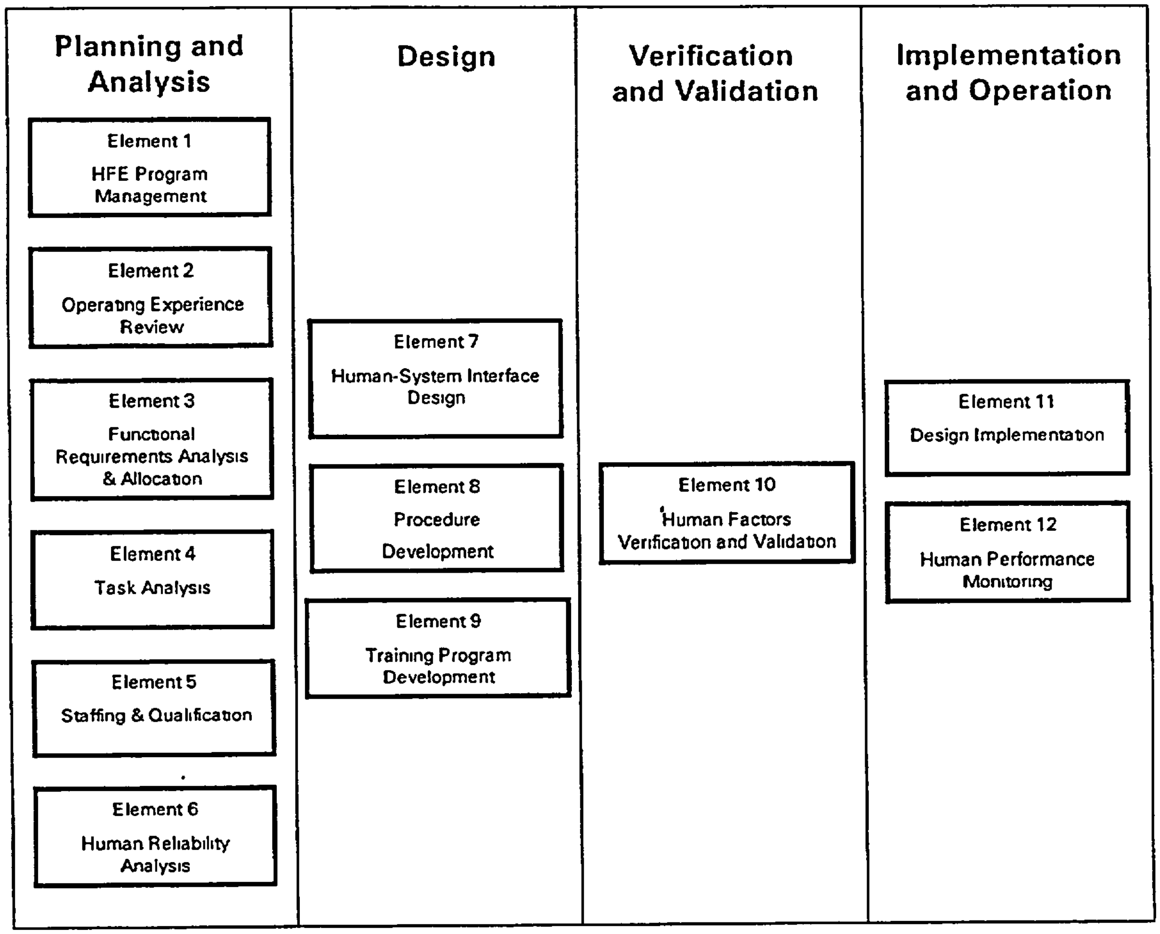

Figure III. B. 1. Human Factors Engineering Program Review Model

significant contributor to core damage frequency (CDF), that by improving human performance licensees can substantially reduce their overall CDF, that a significant human contribution to risk is in failure to respond appropriately to accidents, and that human performance is important to the mitigation and recovery from failures.

Further, there are significant uncertainties in the results of human reliability assessment (HRA) methods. Key sources of uncertainty include the adequacy of the data used to support HRA, and the adequacy of the current understanding of human behavior under accident conditions; for example, the treatment of post-initiator dynamic plant behavior (i.e., the evolution of plant conditions over time). The coupling of this behavior with the operators' training and procedures was a significant aspect of the Three Mile Island Unit 2 (TMI-2) accident in 1979. Other scenario-specific complicating factors, include multiple equipment failures and faulty instrumentation readings, which have been significant contributors in actual operational events yet are, at best, treated as operator workload issues. Another issue raised with widely used HRA methods involves the role of work processes as contextual factor for multiple human failure events (HFE) within an accident sequence and across multiple sequences. A related issue, is the adequacy of current HRA methods for dealing with "latent errors," i.e., pre-initiator HFEs and to determine the relative contribution of specific human performance issues, (e.g., knowledge, training, procedures, fatigue, communications) to CDF. Even less is known about the PRA/HRA for advanced reactors. To do in-depth analyses for advanced reactors could require new methods and sources of data (See section 3.C.7.and 3.D.7.). Until PRA/HRA models can be accurately developed for these new designs to define and prioritize human factors issues, conventional human factors methods may need to be applied.

c. Related NRC Research

The NRC has performed human factors research since the early 1980's. Much of that work was a result of the accident at TMI-2 where the role of human performance in nuclear operations was found to be a major contributor. That research resulted in numerous regulatory tools covering human-system interface; procedures; training, qualifications and licensing; staffing; and fitness-for-duty. Recently NUREG-0711, Rev. 1 was published to serve as a basis for a planned revision to the Standard Review Plan (SRP) Chapter 18 for the review of advanced reactors. This program review model includes the 12 elements shown in Figure III. B. 1. Within each of these elements there is review guidance available that was developed over the last 20 years for conventional LWRs experience. Some of it is relatively up-to-date with current technology, e.g., NUREG-0700, Rev. 2, "Human- System Interface Design Review Guideline," and could be applied to advanced reactors. However, there are other elements, e.g. staffing and qualifications and procedures, that may not apply to advanced reactors. The revision to NUREG-0700 incorporates the results of studies of advanced alarm systems (NUREG-6684 and NUREG-6691), display navigation (NUREG/CR-6690), and issues specific to hybrid control stations (NUREG/CR-6633 - 6637 and NUREG/CR-6749). Though this work was done specific to LWRs, the findings can probably be generalized to control stations for advanced reactors with some tailoring. As newer technology is developed an introduced further guidance may need to be developed.

d. Related International Cooperation

The NRC has cooperated with 20 other countries in the OECD Halden Reactor Project, in part, because of its capabilities, facilities and experience in performing human factors research on nuclear power control room simulators. This work includes recently completed research on automation (HWR-659 and 660), function allocation (HWR-639), large screen displays (HWR-662), computerized procedures (HWR-644), hybrid control rooms (HWR-661), navigation (HWR-656), human error (HWR-625 and 634) and staffing (NUREG/IA-0137). Their facilities include a reconfigurable, computer based control room that can be driven by a pressurized water reactor (PWR), a boiling water reactor (BWR) or a VVER system simulator. They also have access to licensed reactor operators who can serve as subjects in experiments. Further, they are developing a capability in Virtual Reality techniques that can simulate virtual control stations. They do not have a simulation of any of the advanced plants (e.g. PBMR), but they have the capability to develop a simulator when sufficient system and thermodynamic information is available. Their current reconfigurable control room could be used to represent an advanced reactor control station.

e. NRC Research Objectives and Plans

1. Review of Existing Requirements.

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The initial human factors effort should be a systematic review of existing licensing criteria to determine its applicability to each of the proposed advanced reactors. This would include rules, Regulatory Guides, NUREGs, the SRP and consensus standards for topics such as staffing, procedures, training, human-system interface, fitness-for-duty. As part of this effort it would be necessary to understand the proposed concept of operations, control station concepts, control room environment, expected working conditions, etc. This would be based on the top-down approach described above.

2. Automation and Concept of Operations

Advanced reactor control rooms are anticipated to be highly automated. The nature and level of automation are important aspects for the operator because it affects their situation awareness and workload. Operators will be facing a new concept of operations. Will the design be based on the concept of human centered automation? Will they deal with the automation and potential failure of automation? How will they be expected to operate multiple units? What will their role be in maintenance and online refueling? What other roles might the operator have? What role will the operator have in configuration management? What limits will be placed on their activities during periods of work underload? What information will the operators need and how should it be presented? Should procedures be automated or should intervention be required? What will be the consequences of bypassing or overriding automated systems? Who will make operational decisions during emergencies and what must their qualifications be?

3. Function and Task Analysis

Since the HFE Program Review Model is dependent on function and task analysis tools and techniques to perform and review such analyses during the design stage are important to the rest of the elements of the model. Such analytical approaches for evaluating HFE requirements for complex systems have been evolving over the past few decades. Human behavioral modeling techniques, such as task network modeling and discrete event simulation, have been developed and tested by the United States Army and Navy for a decade and some of these techniques have been accredited by the U.S. Department of Defense for use in HFE analyses during system design and engineering. These human behavioral modeling techniques and tools need to be developed or adapted for use in the licensing of advanced reactors.

4. Advanced Reactor Staffing

Central to the safety of any manned-system is the balance between the demands of the work and the available time of the staff. Not only does the humans' workload capacity have to be sufficient to fulfill their requirements during periods of normal operation, human capacity must also be sufficient to handle the periods of high task demands associated with other-than-normal operations. In fact, it is during these periods of off-normal activity that sufficient human capacity to understand the situation, make the appropriate diagnosis, and select the correct action is most critical. This is certainly true of the nuclear power plants in operation today and for those that will be designed for tomorrow. Any system with a human supervisory or control component must be adequately staffed to be safe. It is expected that operators will have longer to respond to unusual situations at advanced reactors than at LWRs, however it will still be necessary to determine the number of individuals needed to safely operate and maintain these new reactors. An analytical approach as described in III.B.e.2.could be used to develop and review staffing using a performance based approach, rather than developing prescriptive requirements. Such an approach would be consistent with the finding in NUREG/IA-0137.

5. Training and Qualifications

Training for LWRs is controlled under 10CFR 50.120 and accredited by the National Academy of Nuclear Training to be consistent with the Systems Approach to Training. NUREG-1220 is used by staff and inspection modules are used by staff in the event a for-cause training review is needed. The current training review methods should be evaluated and updated as necessary to account for possible changes, e.g. use of cognitive task analyses, in addition to traditional task analyses, for development of learning objectives. Further, innovative training concepts, such as embedded training and the use of virtual reality should be evaluated as possible enhancements to training. Qualifications are generally based not only on training but also education and experience. Questions that need to be considered include: From where will the operators and other staff familiar with advanced systems and digital interfaces come? Will past power plant or Navy experience be effective? How will operator licensing need to be changed? What will the requirements be for simulation? Can training and simulation be embedded into the operational setting?

6. Procedures

Currently the NRC has human factors review guidance only for paper-based emergency operating procedures and the operating plants use only paper-based procedures. Since advanced reactors will have computer based or glass cockpit control rooms the procedures are likely to be computerized. Guidance for the review of these system will have to be developed.

7. Computerized Operator Support Systems (COSS)

In addition to a high level of automation, it is anticipated there will be COSSs available to the operator that will help the operator to advise, monitor and diagnose system states. Guidance needs to be developed for the review of such systems so that the NRC can be assured of the accuracy of the information provided, its expected use and the level of dependance expected of the operator.

8. Human-system Interface

The recent revision to NUREG-0700 will be applicable to much of the human-system interface, however there are certain issues not covered in NUREG-0700 for which guidance may need to be developed. These issues were not included in NUREG-0700, Rev. 2, because there were no validated criteria available and there was not sufficient technical basis on which to develop the criteria. These issues include: guidance for high-level displays that are based on processed information, with different types of processing, e.g., functional decomposition; new display types such as flat panels and large screens.

f. Resources and Schedule

1. Review of Existing Requirements

This is an effort that has to be performed for each new type of reactor as information becomes available. The review for the initial reactor would require the greatest resources, with resource requirements decreasing for each new type. The effort would also have to be continued over time as new information becomes available.

FY	2002	2003	2004	2005	2006
FTE	0.1	0.1	0.1	0.1	0.1
$K	$150K	$100K	$50k	$50K	$50K

2. Automation and Concept of Operations

This is effort would also be performed for each new type of reactor as information becomes available. The review for the initial reactor may require the greatest resources, with resource requirements decreasing for each new type. The effort would also have to be continued over time as new information becomes available.

FY	2002	2003	2004	2005	2006
FTE	0.2	0.2	0.2	0.2	0.2
$K	$500K	$400K	$200K	$150K	$150K

3. Function and Task Analysis

This effort would be to develop and validate a tool that could be applied to each type of reactor as sufficient information becomes available. Application and verification would be performed as needed.

FY	2002	2003	2004	2005	2006
FTE	0.3	0.2	0.2	0.1	0.1
$K	$250K	$200K	$100K	$100K	$100K

4. Advanced Reactor Staffing

This topic has already been identified as an issue for the PBMR and would be an initial application of the tool developed in 3. above.

FY	2002	2003	2004	2005	2006
FTE	0.2	0.2	0.1	0.1	0.1
$K	$150K	$150K	$100K	-	-

5. Training and Qualifications

The need to qualify staff for the advanced reactors is in the future so this work would not have to begin immediately, but there should be sufficient lead time to allow for regulatory development. The tool developed in Item 3 above could be used to develop the guidance.

FY	2002	2003	2004	2005	2006
FTE	-	0.1	0.2	0.2	0.2
$K	-	-	$300K	$300K	$300K

6. Procedures

The need for guidance on procedures will not be immediate and can wait the development of the tool from Item 3. above.

FY	2002	2003	2004	2005	2006
FTE	-	0.1	0.2	0.2	0.2
$K	-	-	$200K	$200K	$100K

7. Computerized Operator Support Systems (COSS)

Guidance in this area will depend on the type of COSSs proposed but would need to lead licensing.

FY	2002	2003	2004	2005	2006

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FTE	-	-	0.2	0.2	0.2
$K	-	-	$150K	$150K	$150K

8. Human-system Interface

This area is dependent on the development of new interface technologies and concepts as the license applications progress.

FY	2002	2003	2004	2005	2006
FTE	-	-	0.1	0.2	0.2
$K	-	-	-	$300K	$300K

g Priorities

1. Review of Existing Requirements

This effort is **HIGH** priority, since it would serve as a basis for determining what other work would need to be done.

2. Automation and Concept of Operations

This work is **HIGH** priority, since it will set the stage for other activities.

3. Function and Task Analysis

This work is **HIGH** priority, since it would result in a tool that could be used in licensing as well as a basis for other work. This tool could reduce burden and increase regulatory efficiency.

4. Advanced Reactor Staffing

This work is **HIGH** priority since its is an existing user need from NRR and can serve as a test case for the tool developed in effort 3.

5. Training and Qualifications

This work is a **HIGH** priority, since having qualified personnel is essential to safe nuclear power operations and maintenance.

6. Procedures

This work is a **HIGH** priority, since the ability of operators to correctly respond to plant conditions is dictated by procedures.

7. Computerized Operator Support Systems (COSS)

This work is a **MEDIUM** priority, since the systems in question would be advisory.

8. Human-system Interface

This work is a **HIGH** priority, since it is imperative that operators have information and control systems that support their efforts without detracting from their work.

III. C. HIGH TEMPERATURE GAS-COOLED REACTORS

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The Pebble Bed Modular Reactor (PBMR) is a 110-Mwe modular HTGR that uses helium as a coolant. It is expected that multiple modules will be developed at a single site. The PBMR design is under development in the Republic of South Africa (RSA) and is being considered for licensing in the United States by Exelon Generation, USA. There are certain innovative aspects of design, technology, and operating characteristics that are unique to the PBMR; therefore, the PBMR licensing approach is expected to be different from that for the conventional and the advanced LWRs. To license a PBMR in the United States, it is imperative to identify and resolve the key design, safety, licensing, and policy issues applicable to the design before a COL application is submitted. The staff has experience from the licensing of Fort St. Vrain, the design certification of the evolutionary and passive LWRs, and the earlier review of the DOE-supported MHTGR.

General Atomic's Gas Turbine - Modular Helium Reactor (GT-MHR) design is an approximately 300-Mwt helium reactor design based on HTGR technology. International HTGR experience, particularly with Dragon in the United Kingdom, AVR and THTR in the Federal Republic of Germany, and domestic experience with Peach Bottom Unit 1 and Fort St. Vrain offer General Atomics an extensive technological and operational basis on which to capitalize. Similar to the PBMR, the GT-MHR design uses helium as the coolant and employs refractory fuel. The principle difference is that the ceramic-coated particles in the GT-MHR design are contained in fuel compacts that are inserted in graphite fuel elements instead of pebbles.

International collaboration on GT-MHR design work is being performed in the Russian Federation (RF) under a joint U.S./RF agreement, and is jointly sponsored by DOE and the RF (Minatom). In June 2001, during a meeting with the staff, General Atomics outlined a commercial program in the United States that would be initiated after the international program is complete. As part of the GT-MHR pre-application activities, General Atomics plans to interact with the staff extensively to identify any additional licensing, technical, and design issues; to obtain staff feedback and guidance; and to identify any significant policy issues that may need Commission consideration.

Currently, the PBMR pre-application review is in progress and it is expected to continues through early FY 2003. GA is expected to request the GR-MHR pre-application review in the near future.

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III.C.1 THERMAL-FLUID DYNAMICS

(a) Description of Issues

Power reactors are licensed by showing compliance with specified safety limits. Some limits are easily identified and predicted while others require complicated modeling to properly evaluate. When modeling is required applicants apply what are typically complicated mathematical representations of the system. Many of these "models" are typically combined into a computer code which represents the significant phenomena in the system under consideration. Due to their complexity these "codes" are difficult to fully understand and need detailed assessment to demonstrate that they are adequate for the proposed application. Both the codes and the code assessment needs to be independently reviewed.

In order to independently review applicant HTGR safety analysis, the staff needs an independent thermal-fluid dynamics assessment tool. The staff has completed a preliminary review of the analytical capabilities needed to model HTGR fluid flow. The findings of this review can be summarized as follows. Given the nature of HTGR transients, a code must reliably and efficiently predict transients that evolve over time scales of days, not hours as we have become accustomed to in LWR analyses. Furthermore, these transients are driven by conduction through solid structures, not convection and this capability, although currently exists in all codes, will have to be extended to three-dimensions and a spherical model will have to be added. The staff code also must be able to model all of the turbo-machinery and passive decay heat removal systems, which implies that we need to accurately model gases (helium and air) in natural circulation. These systems are important for long-term heat removal and recovery as well as determining initial steady state operating parameters and conditions. Turbo-machinery will more than likely be successfully modeled using existing pump models, but this capability will have to assessed and modified as necessary. For PBMR, we need the capability to model flow and heat transfer in a packed bed configuration. The code must also be able to model two different working fluids at once to model component cooling water systems. Finally, the capability to model graphite as a solid structural support element and reflector will have to be added. In formulating this summary the staff relied upon an article by Gary E. Wilson, et. al., entitled "Phenomena-based thermal-hydraulic modeling requirements for systems analysis of a Modular High Temperature Gas-Cooled Reactor," Nuclear Engineering and Design, 136, p. 319-333, 1992.

Two types of codes will be used to fulfill this need for HTGRs. These are the traditional systems analysis codes such as TRAC or RELAP and the computational fluid dynamics codes. The NRC currently uses the FLUENT code. The system analysis code for HTGR applications will be built upon our existing TRAC-M code. This will be the best use of agency resources as TRAC-M already possesses most of the features discussed above, and given its modular structure, new capabilities can be added with relative ease. For example, TRAC-M already can model helium as a working fluid and the necessary material properties for helium are already in the code. These will simply have to be assessed for accuracy. Where specific capabilities are not currently in TRAC (for example, modeling helium turbines), adding this capability can be readily achieved by changing one or more of the TRAC-M functional modules. The GUI will also need to be updated to allow analysts to model HTGR designs. FLUENT will be used because it gives us the ability to more reliably predict parts of the fluid system when we need to assess the capability of our system code against some assumed known reference standard or when we need to assess a particular phenomena in more detail.

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Several issues will need to be addressed by the proposed research:

1. Confirm and modify, as needed, the capability to model flow and heat transfer in packed beds. The solver in TRAC-M is based on a porous medium assumption which should be directly applicable to packed bed analyses if given appropriate inputs. Appropriate constituitive relationships will have to added. Three-dimensional conduction and a spherical conduction model will have to be added. An improved radiation model is also needed. These capabilities will have to be assessed.

2. Confirm and modify, as needed, the capability to model HTGR turbo-machinery. At a minimum we will need to change the turbine model to remove some restrictions related to LWR applications. Appropriate data will also be needed for input preparation.

3. Confirm and modify, as needed, to capability to model natural circulation of gases.

4. Add the capability to simultaneously model two different working fluids. Along with this the ability to track multiple non-condensible gas sources must be added.

5. Assess speed of the code and improve as necessary to allow for efficient simulation of transients on the order of days.

6. Add graphite as a structural material.

7. Update the GUI to work with HTGR designs.

An overall assessment against relevant data will have to be performed. This effort might identify a need to modify the code in areas not mentioned above.

(b) Risk Perspective

The LWR thermal-hydraulic codes have traditionally been used to assess the success of failure of different accident sequences. This information is factored into plant PRAs by modifying failure probabilities to account for the results of the analyses. At this time, no new role for thermal-hydraulic codes has been identified.

(c) Related NRC Research

No related NRC research has been identified at this time. An effort to modify TRAC-M to add the currently identified capabilities is being initiated at the Los Alamos National Laboratory.

(d) Related International Research

The International Atomic Energy Agency (IAEA) sponsored an international standard problem modeling the conduction cool-down of a HTGR. Specifically, this effort was directed at modeling passive heat removal systems. This effort highlighted the importance of accurate

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modeling of heat sources and difficulties with modeling these passive systems. The results of this study are documented in IAEA TECDOC-1163.

As of this time, no relevant experimental data has been identified.

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(e) NRC Research Objectives and Plans

NRC needs an independent computer code for HTGR thermal-fluid dynamics analyses that has been thoroughly assessed and peer reviewed. The initial effort will be focused on adding the necessary capability for HTGR analysis to TRAC-M. This is the first priority. The staff will use a PIRT panel to identify further development and experimental data needs. The results of the PIRT could lead the staff into further code development activities and experimental data collection. At a minimum, the PIRT will identify and rank relevant phenomena and assessment needs. The staff will assess the code according to the rankings of the PIRT. An uncertainty analysis will be performed to assess the effect of code modeling relative to an as yet undetermined figure of merit. Finally, the staff code will need to be peer reviewed.

(f) Schedules and Resources

The initial effort is being initiated, The estimated cost and schedule are as follows:

Cost: $1M
Period of Performance: 1 year

(g) Priority

The need for this work is considered high priority, since the NRC currently has no basis for independently confirming the HTGR performance..

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III.C.2 SEVERE ACCIDENT CODES, INCLUDING RADIOLOGICAL SOURCE TERM

(a) Description of Issues

The phenomena of concern in analyzing severe accident scenarios in any nuclear reactor are those phenomena involved in accidents that are beyond design basis and in which significant core degradation occurs. A beyond design basis accident is one that causes the release of fission products beyond an acceptable limit. Since fission products are bound in the reactor fuel, any event or action that can cause fuel damage such as chemical, heat, or mechanical damage to the fuel can be considered as an important issue for severe accident analysis. Release of the fission products to the environment (the composition and quantity of which is called the 'source term') may be impeded by particular features of a plant, such as coolant system piping and other equipment, containment filters, sprays, and other mitigative aspects that provide fission product retention. Therefore, aspects of the reactor design that impede or accelerate the release of fission products to the environment also need to be considered when attempting to predict accident source terms.

Since severe accidents are normally considered high consequence, low probability events, specific sequences leading to a severe accident are usually only of interest to the degree to which they determine the plant state during the postulated severe accident. Accounting for situations where the plant state is in one of these low probability states may require knowledge of phenomena outside of the normal design range of conditions. Therefore, predicting the plant state during a postulated severe accident may or may not require the use of a code that is normally used for design basis analysis. Thus any computer codes used specifically for analyzing severe accident phenomena would focus mainly on such phenomena and leave the thermal-fluid dynamics and kinetics phenomena to codes specifically designed for those purposes, except when analyzing situations outside of the valid ranges for those codes.

Computer codes used to evaluate severe accident phenomena in light water reactors generally focus on core degradation, relocation, and fission product transport within the reactor coolant system. Source term codes typically follow the fission product transport from the reactor coolant system to the environment. Specific aspects analyzed by our primary LWR severe accident code, MELCOR, includes:

(a) Core uncovering (loss of coolant), fuel heat-up, cladding oxidation, fuel degradation (loss of rod geometry).

(b) Core material melting and relocation, heat-up of reactor vessel lower head from relocated fuel materials and the thermal and mechanical loading and failure of the vessel lower head, and transfer of core materials to the reactor vessel cavity,

(c) Core-concrete attack and ensuing aerosol generation.

4) In-vessel and ex-vessel hydrogen production, transport, and combustion.

5) Fission product release (aerosol and vapor), transport, and deposition behavior of radioactive aerosols in the reactor containment building, including scrubbing in water pools, and aerosol mechanics in the containment atmosphere such as particle agglomeration and gravitational settling.

6) Impact of engineered safety features on thermal-fluid dynamics and radio-nuclide behavior.

For HTGRs excellent fission product retention properties of the fuel particles and the inherent limits to the maximum temperature that can be experienced by the reactor core lead to a very low expected fission product release. Under normal operating conditions, over time some level of fission products are expected to escape from the fuel and transport through the primary system. Mild transients in operation could increase the otherwise low release rates. Knowledge of this behavior will allow for proper shielding design and system maintenance design.

Off-normal events such as loss of coolant accidents do subject the fuel to some amount of heat-up that will lead to some increased release of fission products from the fuel elements. Additionally, ingress of air or moisture can lead to oxidation of coated fuel particles and elements and kernels which also tend to increase the release of fission products. Over the life of a plant, fission products, such as Cs, are expected to be released from the fuel and deposit or "plate-out" on surfaces in the primary coolant system. These deposits are subject to re-suspension or "lift-off" under accident conditions such as a loss of coolant accident.

Finally, in time, burn-up effects such as fission gas pressurization and attack of kernel coating layers by fission products will lead to increased failure rates under both normal operating conditions as well as off-normal. While the consequences of these effects may be low, they nonetheless must be quantified in order to demonstrate adequate margins of safety.

Specifically, the following issues need to be addressed for the HTGR fuel:

- coating failure due to increased gas pressure (from both fission gas and carbon monoxide)
- loss of the pyrolytic carbon pre-stressing at elevated temperatures
- silicon carbide corrosion from the interaction of fission products, notably palladium, with the silicon carbide layer
- thermal decomposition of silicon carbide at high temperatures
- migration of fission product, notably silver -- ^{110m}Ag -- through the silicon carbide layer
- oxidation of graphite at high temperature and induced cracking

(b) Risk Perspective

MELCOR was developed to be used to perform level 2 PRA analysis. As a level 2 PRA code MELCOR must be able to model any phenomena considered important in analyzing a accident sequence postulated by the level 1 PRA studies. The results of the MELCOR analysis are meant to be feed into the MACCS code for level 3 PRA analysis. MELCOR is therefore already an integral part of the NRCs ongoing risk informed regulation paradigm.

(c) Related NRC Research

In support of the Russian Production ADE Reactor conversion project, MELCOR (version 1.8.4) was modified to address potential accidents in the graphite moderated pressure tube design used in the Russian production of plutonium. Graphite properties had been added to the MELCOR database earlier in the support of "N"-Reactor studies, and graphite oxidation models were added under the ADE project. The implementation of these models was provisional and incomplete, and produced a variant code version that is incompatible with the current standard code. In particular, graphite properties were not propagated into the MELCOR FUEL component and the related code subroutines that address the MELCOR FUEL component. Nevertheless, this pre-existing work is expected to be useful in the extension of MELCOR capabilities to treat HTGR core designs.

The results of any experimental research being done related to the HTGR/PBMR work such as experimental work on graphite properties and fuel fission product release and fission product deposition (if planned by the NRC) should be integrated with the plan for model development and assessment for MELCOR. This part of the severe accident/source term plan will be left in a 'to-be-determined' state until such experiments have been identified.

(d) Related International Cooperation

A number of codes have been developed and used by the international community and by DOE laboratories. While these codes can not be used directly by the NRC for regulatory analysis they may be useful for bench-marking our codes. Of particular interest is the GRSAC code developed by ORNL for severe accident analysis of the HTGR reactor systems. GRSAC could be used in the verification and validation of the MELCOR code after MELCOR has been updated to handle the HTGR and PBMR design components.

Also, the Cooperative Severe Accident Research Program (CSARP) provides a general cooperative framework for sharing severe accident information with international organizations. Any information that is relevant to these ALWR designs being made available through the CSARP should be reviewed for potential application to the MELCOR code.

(e) NRC Research Objectives and Plans

Currently, to make the best use of resources, RES has focused on consolidating and modernizing our severe accident codes in to one code, MELCOR. Based on the information currently available, we can identify aspects of the new designs that will certainly need to be addressed by modifications to MELCOR. Aspects such as HTGR- specific material properties and material related models can be addressed now with the data and models available from previous international and domestic work on HTGRs. Other aspects, perhaps dealing with unusual flow properties or heat transfer issues, will need to be addressed as more is learned from new and existing system level experiments and maybe from operational experiences.

Incorporate Graphite as Fuel and Structural Material

MELCOR already has graphite properties as part of its library of materials properties, as a legacy of work done to develop now orphaned code versions for the US N-Reactor and the Russian ADE reactors. The major part of the required effort will therefore be in expanding the list of materials allowable for MELCOR components FUEL and STRUCTURES, and propagating the expanded components throughout the various code physics modules treating heat-up, oxidation, and component failure.

Implement Oxidation Models for Graphite

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Graphite oxidation models will be added to the current collection of material oxidation correlations. Literature will be reviewed to determine the most appropriate models and options to be allowed. Oxidants to be considered include oxygen, air, steam and moist air. The oxidation models will account for production of CO and CO_2, as well as hydrogen in the case of steam oxidation, where the CO may further react with oxygen. If sufficiently fed by an oxidant, and if heat losses are sufficiently low, the new models must be able to predict a self-sustaining graphite fire. Existing MELCOR combustion models already treat burning of CO.

Extend Fission Product Release Models

Current MELCOR fission product release models are based on CORSOR, CORSOR-M or Booth formulations for release from overheating UO_2 LWR fuel pellets. These models will be expanded to predict releases from TRISO fuel particles embedded in graphite compacts. Both spherical fuel pebbles and block/prismatic fuel configurations will be accommodated. Isotopes of importance include Kr, I, Cs, Ag, Sr, Eu and Ru, to name a few. Where models are deemed appropriate, the effects of air or steam oxidation as well as burn-up will be included. Burn-up effects include pressurization of the fuel kernels as well as chemical attack of the kernel pressure boundary layer. Both effects lead to increased kernel failure rates as well as reduced maximum temperature capabilities.

Many models exist for predicting fission product release from HTGR fuels for inert and oxidizing conditions, and models continue to be under development in several international HTGR efforts. These models will be reviewed for appropriateness and incorporated into MELCOR's Radionuclide Physics Package.

Extend Fission Product Plateout and Liftoff Models

MELCOR's fission product transport models provide for the deposition of fission products on pipes within the primary system by both phoretic processes for particulate materials, and by condensation, or plateout, from condensable vapors. While condensed vapors can be revaporized from surfaces, MELCOR currently lacks a model for the re-entrainment, or liftoff, of solid or dust depositions. MELCOR's modeling in this area will be reassessed in the context of HTGR applications and upgraded as needed. Models for plateout and liftoff particular to HTGR systems will be evaluated for use in MELCOR.

Also important in this area is the potential for formation of dust from the abrasion of graphite in the case of pebble beds, or from suspension from the prismatic blocks. Dust accumulation was observed in the US Peach Bottom and Fort St. Vrain HTGR systems. Such dusts will participate in the transport of fission products by agglomerating and adsorbing with them. Sources of dust must be included in the aerosol mechanics models in order to account for the associated transport processes. Finally, in the context of long-term accumulation of fission products in regions of the plant system, we will address where appropriate the need for fission product decay chain effects. This will be necessary in order to accurately treat buildup and decay of fission products whose half-lives are short compared to the time of release and accumulation. This is expected to be important for a some isotopes such as ^{110}Ag and ^{131}I.

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Extend Thermal-Fluid-Dynamics to Accommodate Long-Term Plant Simulation

Presently, while MELCOR can analyze LWR plant performance over a relatively long period of simulated elapsed time, several days for example, time periods of interest in HTGR analyses may be considerably longer. In HTGR's it is of interest to evaluate the accumulation of slowly released fission products in different parts of the reactor system in order to perform consequence analysis in the event of sudden system depressurization when deposits can be re-entrained and dispersed outside of the primary coolant system.

In order to make such long term calculations it would be more practical to extend MELCOR's fluid-dynamic solver method to allow for faster time step advancement. Currently, MELCOR's solver requires that the time step be no larger than that which would result in the displacement of one half of the fluid-dynamic material present in any distinct volume of the system. This is loosely referred to as a "material" Courant limitation, and imposes a limit on the stability of the existing hydrodynamic algorithms. However, it has not been a major concern, because time steps of this magnitude are normally necessary in order for MELCOR to predict rapidly changing pressure transients accurately. However, in the case of a long duration of steady plant operation, the requirement can be relaxed. One way is to perform several small Courant-limited time steps in order to obtain an accurate steady flow solution, and then to use these conditions to extrapolate effects such as fission product and deposition over longer periods of time. Changing pressure conditions can demand Courant-limited time step advancement when needed.

There is a potential to implement the SETS (semi-explicit two-step) numerical solver in MELCOR at this point. The SETS method would permit MELCOR to make much larger implicit (or semi-explicit) steps in time while maintaining numerical consistency (this method bypasses the Courant numerical limitation). Integration of SETS into MELCOR should be considered at this time not only because of the need introduced by the potential for HTGR/PBMR analysis, but also because MELCOR is currently beginning a phase of development with the goal of both consolidating and modernizing the MELCOR severe accident code. One aspect that should be considered in the modernization process is the potential to implement an "external component interface" (ECI) into MELCOR as was done recently for the TRAC-M code. This ECI interface would allow MELCOR to use detailed flow calculations from TRAC-M, detailed kinetics calculations from PARCS, etc...

Methods for applying MELCOR models over "macro" steps in order to make long term plant performance analyses more practical need to be developed. This will extend MELCOR current capabilities to include transient, as well as steady state long term plant performance analyses.

(f) Resources and Schedule

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Update Materials Properties Models

Expand the fuel and structural material components in MELCOR to include Graphite. Graphite/Fuel degradation and relocation modeling should be considered, as well as strength and integrity of core supporting structures. Core description considered should be general to allow description of both prismatic as well as PBMR core design. In this task consideration should be given to use of MERIS separate effect code models.

Estimated completion date:	January 2003
Level of Effort:	5 staff-months (FY02)
	2 staff-months (FY03)

Oxidation models

Expand the current oxidation models for various material in the code to include graphite oxidation model. Oxidant to be considered for this models should include oxygen, steam, and moist air. The oxidation model should account for CO and CO_2 as well as H_2 in the case of steam oxidation, where CO may further react with O_2. The model should be able to predict self-sustaining graphite fire. In addition to graphite fire, smoke and particulate formation should be considered.

Estimated completion date:	September 2002
Level of Effort:	5 staff (FY02)

Fission Product Release Models

Extend fission product release models in the code by expanding current fission release models which are based on CORSOR, CORSOR-M or Booth formulation to predict release from advanced gas reactor type fuel (e.g., spherical fuel pebbles, block/prismatic fuel configurations). Where deemed appropriate, effects of air or steam oxidation as well as burn-up should be included.

Estimated completion date:	December 2002
Level of Effort:	3 staff-months (FY02)
	2 staff-months (FY03)

Improve Numerics

Improve MELCOR's numerics to allow use of longer time steps in order to carry out reasonable execution times for slow developing slow transient problems that persist over an extended time span. This may involve changing the numeric solver for MELCOR to implement the SETS (semi-explicit-two-step) algorithm. This could be done as part of

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the MELCOR consolidation and modernization process.

Estimated completion date: March 2003
Level of Effort: 4 staff-months (FY02)
6 staff-months (FY03)

Assess Code Against Available Experimental Data and Other Codes

When model implementation in MELCOR code is completed, perform assessment of the code against available experiments. Moreover, upon NRC's approval, prepare input deck for selected advanced reactors design, and demonstrate code capabilities for selected performance scenarios. One month after completion of this task a draft report should be submitted to NRC for review, and a final report should be completed a month after NRC's review of the draft report.

Estimated completion date: November 2003
Level of Effort: 8 staff-months (FY03)
2 staff-months (FY04)

(g) Priorities

These tasks are assigned a high priority because model identification and implementation should be done as soon as possible so that assessment can be completed before any analyses need to be done. Numeric improvements and detailed code-to-code assessments would be a secondary priority.

III.C.3 HTGR FUEL PERFORMANCE AND QUALIFICATION

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a. Description of Issue(s)

Modular high-temperature gas-cooled reactors (HTGRs), such as the Pebble Bed Modular Reactor (PBMR) and the Gas Turbine Modular Helium Reactor (GT-MHR) are designed to have unique safety features and safety characteristics. Foremost among these is the all-ceramic fuel element with high integrity TRISO coated fuel particles (CFPs). The intended safety characteristic of the TRISO CFPs is to provide the initial barrier and primary containment function against the release of fission products to the environment. Accordingly, the safety analysis, the licensing basis and the confidence of the public in these plants will hinge in large part on the assured integrity and the fission product retention capability of the CFPs. These characteristics, with ample margins, will need to be clearly and convincingly demonstrated by the applicant for all licensing-basis conditions. They will also need to be effectively confirmed for the NRC to conclude, with certainty, that these plant designs provide adequate protection of public health and safety.

For modular HTGR plants, there is a range of significant fuel design, fuel manufacture, fuel quality and fuel performance issues which will require research initiatives by the respective applicant/vendor. Exploratory and confirmatory NRC research will also be required for the staff to conclude with confidence that the plants provide for adequate public health and safety. The following paragraphs provide a brief discussion of the background and the exploratory and confirmatory research issues in the area of HTGR fuel performance and qualification.

Background for HTGR Fuel Research Plan

The design of HTGR fuel with CFPs has evolved empirically over the last four decades. This evolution began with fuel elements utilizing fuel particles with a single anisotropic carbon layer. Later, fuel elements with BISO CFPs involving a layer of buffered isotropic pyrolytic carbon were developed, and, more recently, fuel elements with TRISO CFPs have been qualified. This most recent design involves CFPs with a fuel kernel, a porous buffer layer, an inner pyrolytic carbon layer, a silicon carbide layer and an outer pyrolytic carbon layer. The fundamental characteristics of ceramic CFPs for HTGRs have also been investigated over this period. Several countries initiated fuel development and qualification programs with the coated particle as the basic unit. These efforts have addressed the design, design-analysis, manufacture irradiation testing, accident performance and utilization of these fuels in HTGRs.

In the early 1960s, the UK Atomic Energy Agency (UKAEA) initiated a CFP development program. The objective of the program was to define the essentials of CFP production and to identify the important process parameters which determine CFP properties, and thus its irradiation and accident performance.

In the 1970s, in the Federal Republic of Germany (FRG), the production process for spherical fuel elements with BISO fuel was developed, established and licensed for use in the AVR and THTR. Later, in the early 1980s, a TRISO coated particle design with low enriched UO_2 was developed. This TRISO CFP design was later established as the reference fuel for the new FRG modular HTGR designs such as the HTR-Modul. The qualification program for the FRG TRISO fuel included a range of irradiation experiments in materials test reactors (MTRs) and the AVR and included aspects such as accident simulation testing. The FRG program was aimed

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at establishing the concept of a 1600°C limit for pebble fuel elements with TRISO CFPs. The concept was that TRISO CFP failures would not occur until well above1600°C, while the peak transient fuel temperature for a modular HTGR design would not exceed 1600°C during the most severe postulated accident. The FRG MTR fuel irradiation testing research on CFPs investigated such aspects as: particle performance (i.e., failure), fission product (FP) transport in the fuel kernel and FP transport in coating layers of intact particles, FP release from broken particles and the effects of chemical attack (e.g., moisture and air ingress) on particles. Fuel element (i.e., pebble) testing investigated aspects such as pebble surface wear and FP transport through the graphite matrix and included large scale demonstration tests in the AVR. "Proof" tests under simulated HTGR operating conditions were also carried out with test parameters chosen to envelope the selected HTGR's design conditions (e.g., operating temperature, burnup, fast fluence) followed by accident simulation heatup tests. Although the FRG HTGR developmental efforts were phased out during the 1990s, a significant number of unirradiated archive FRG reference fuel elements that were fabricated for use in the AVR are currently in storage at the Julich Research Center. This fuel is stated to be of the reference design and manufacture for the PBMR pebble fuel, but of higher enrichment. A number of these archive elements may be made available to NRC and other third parties for use irradiation testing programs.

In the United States, the reference fuel element design for modular HTGRs is based on TRISO CFPs encapsulated in small cylindrical fuel compacts which are inserted into holes in hexagonal prismatic graphite blocks. The fuel kernel in composed of UCO. The fuel operating conditions for the reference US design are more severe than FRG pebble fuel conditions. These TRISO CFPs and fuel compacts are stated to be the reference fuel design for the GT-MHR.

In Japan, the reference HTGR fuel involves hexagonal prismatic graphite blocks utilizing graphite fuel rods containing fuel compacts with TRISO CFPs. The CFPs utilize a UO_2 kernel with customized coating layer thicknesses to achieve optimum performance for the operating and postulated accident conditions of the HTTR. The burnup limit for the HTTR fuel is significantly lower than the FRG or US designs. This is intended to accommodate the HTTR's higher fuel operating temperatures and higher peak fuel temperatures for a postulated reactivity insertion (rod ejection) accident. The Japanese fuel qualification program for the HTTR has been completed and included a range of bounding irradiation conditions in MTRs. This fuel is currently operating in its first cycle in the HTTR, which achieved full power operation in late CY 2001.

In China, the current reference fuel design is very similar to the FRG reference pebble fuel with TRISO CFPs. The fuel manufacturing methods are also very similar to those used in Germany to manufacture their reference fuel. The fuel is designed for operation in the HTR-10 pebble bed reactor which is located at the Institute for Nuclear Energy Technology. Irradiation qualification testing of the fuel is currently underway in an MTR and its successful performance is a licensing requirement for power escalation of the HTR-10. As of late CY 2001, power escalation of the HTR-10 had not yet been authorized.

The PBMR fuel design is the same as the FRG reference fuel design. PBMR fuel is also to be manufactured using feed materials, processes and equipment which are "equivalent" to those that were used to manufacture the FRG reference fuel. The expectation on the part of the PBMR design team is that the PBMR fuel will exhibit the same quality, irradiation performance and accident performance as the FRG fuel. This expectation also extends to fuel performance

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under PBMR service conditions. Plans are currently being implemented to develop and establish the process, equipment and facilities to be used to manufacture the production fuel for the PBMR demonstration plant and initial commercial PBMR plants. It is not expected that fuel from manufacturing facility will be available for irradiation testing until the first quarter of CY 2005.

The GT-MHR fuel design (i.e., CFPs, fuel compacts, prismatic block fuel elements) is expected to be very similar to the reference US HTGR fuel design for TRISO coated particles with enriched UCO fuel. The GT-MHR TRISO fuel design is different from that was used in the Ft. St. Vrain reactor (which involved UC_2 with very different chemistry) and the PBMR which utilizes UO_2. Additionally, the Ft. St. Vrain TRISO CFP defect rate from fuel manufacture, although within design specifications, was higher than that achieved for FRG reference fuel and higher than desirable. Accordingly, in an effort to reduce CFPs defects induced during pressing of the fuel compacts the CFP design and/or manufacturing process is expected to change for the GT-MHR fuel.

Exploratory Research Issues

Virtually all of the past and ongoing worldwide irradiation testing research of HTGR fuel designs with TRISO CFPs involved accelerated irradiations in MTRs. Although there subsequently was significant large-scale operating experience with these fuels in plants such as the AVR in Germany, accident simulation tests (i.e., fuel heat-up test following irradiation) to qualify the fuel involved accelerated irradiations in MTRs. There is not a well-established and thorough understanding of the mechanics and properties (e.g., creep) of CFP behavior, failure and fission product release to conclude with certainty that fuel accident simulation tests following accelerated irradiations are conservative as compared to the rate of fuel irradiation in a power reactor. Accident simulation heatup tests either after realtime MTR fuel irradiations or after fuel irradiations in a power reactor would be required to resolve this issue.

Virtually all of the accident simulation tests for TRISO CFPs involved so called "ramp and hold" temperature increases. These typically consist of increasing fuel temperature at about 50°C/hr up to a set temperature (e.g., 1600°C, 1700°C or 1800°C) and then holding the fuel at the set temperature for several hundred hours while fission product release measurements are taken. The results of ramp-and-hold tests up to 1600°C, for qualified fuel, show that no additional CFP failures occur. However, in the Federal Republic of Germany, there was at least one test in which the temperature was controlled to closely simulate the predicted accident heat-up curve to 1600°C for a design basis reactor coolant pressure boundary failure. For this test, CFP failures were observed to occur. Additional post-irradiation accident simulation tests that closely simulate the predicted temperature curve for a design basis reactor coolant pressure boundary failure would be required to determine if the traditional ramp and hold test accident simulation approach is conservative with respect to establishing CFP failure rates for postulated accidents.

Among the most limiting events that could challenge HTGR CFP integrity are those involving large scale chemical attack such as air intrusion following a pipe large break in the reactor coolant pressure boundary (RCPB) and moisture intrusion for a postulated heat exchanger tube failure with the reactor helium pressure falling below the heat exchanger tube pressure. While there have been experiments on oxidation of unirradiated HTGR fuel in air and water at HTGR accident temperatures and measurements of HTGR fuel oxidation due to air or moisture impurities in helium during fuel experimental irradiations, there are no known experiments on

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fully irradiated HTGR fuels that simulate the effects of large air or water ingress events. Additional post-irradiation accident simulation tests that closely simulate air or water intrusion events and take the fuel to the onset of CPF failures would be needed to fully assess the adverse effects of air and water corrosion on HTGR fuels and the margins to failure for such events.

Very limited testing has been conducted on fuels with TRISO CFPs to assess the capabilities and the margins to CFP failure for reactivity events involving a large energy deposition in the fuel over a very short time interval (<< 1 second). Some limited testing was conducted in Japan for a postulated control rod ejection accident in support of the HTTR licensing and was one of the limiting licensing basis events. Although the staff has been told that the PBMR design does not have a potential for such large rapid reactivity events, this may not be the case for the GT-MHR with control rods located in the central core (fueled) region. In order to fully understand the margins to failure for reactivity events, fuel irradiation experiments involving such reactivity insertion events would need to be conducted.

Only limited worldwide testing has been conducted on previously qualified FRG or US HTGR CFP fuel for conditions that went well beyond the maximum qualification operating temperature and maximum qualification fuel burnup. In order to fully understand the margins to CFP failure and fission product release for fuel operations beyond the maximum allowed operating temperature (e.g., 1250 °C for PBMR) and design fuel burnup limits (e.g., 80 GWd/t for PBMR) fuel experiments involving irradiation conditions beyond such limits would need to be conducted.

Confirmatory Research Issues

It is assumed that PBMR and GT-MHR license applicants/vendors will conduct all fuel testing necessary to support licensing. Such fuel testing should address all significant aspects of the licensing basis and should address: a sufficient range of parameters to cover uncertainties and variations; the plant-specific service conditions of the PBMR and GT-MHR (core maximum operating temperature, fuel design burnup, fast fluence, number of fuel passes through the core, load follow), include a sufficient quantity of fuel elements and CFPs to establish a sufficient statistical database; and cover the range of potential CFP failure mechanisms and performance factors (e.g., fission product release) applicable to or potentially applicable to the licensing basis. It is also expected that such testing will use fuel from fabricated by the fuel production facility, utilizing equipment, processes and methods that are identical to those that are to be used to fabricate the production fuel for the (GT-MHR or PBMR) fuel cores. However, some test objectives supporting these fuel irradiation test plans may utilize counterpart German or US archive fuel or pre-production fuel.

The corresponding NRC confirmatory research activities will need to address the issues discussed in the following paragraphs:

The manufacture of the CFPs involves a chemical vapor deposition (CVD) process. In total billions of CFPs and hundreds of thousands of fuel pebbles or fuel compacts are loaded into a modular HTGR core. By its nature, the CVD process and fuel element manufacture involves distributions of the attributes of the CFP layers (e.g., density, anisotropy, thickness, microstructure, stoichiometry) and distribution of attributes of the fuel pebbles or fuel compacts. Therefor the quality and the performance of HTGR fuel are very statistical in nature. Consistent,

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reliable and repeatable fabrication is critical to making fuel that performs satisfactorily in the core during normal operations and licensing basis events. The applicant's predicted fuel performance with respect to fission product release and CFP failure is therefore also statistical for licensing basis conditions. Additionally, history shows that there is a learning curve associated with making fuel of this type and that experience is important to achieving good fuel performance. The statistical aspects and fuel performance achievement aspects should be confirmed by independent testing of the PBMR and GT-MHR production fuel. This testing should as a minimum be for fuel which reaches the design maximum core operating temperature (and for PBMR fuel has the maximum possible number of passes through the core) and reaches the design limit for fuel burnup. Independent testing should include accident simulation ramp and hold testing for design basis accident temperatures up to 1600°C. Beyond the design-basis accident simulation ramp and hold testing should also be conducted for 1700°C and/or 1800°C to confirm fission product releases and margins to fuel failure.

It is important that the NRC staff and contractors have expertise on the proper conduct of fuel irradiation experiments, including a thorough understanding of good testing practices as well as testing limitations and potential opportunities for oversights and omissions. Such knowledge and experience will provide the staff with a sound basis for judging the acceptability of the applicant's fuel irradiation program methods, quality assurance practices, etc.

The proposed design basis of the PBMR (and possibly the GT-MHR) is to include load-follow operation in which daily power cycling of the reactor fuel would be permitted. For a PBMR, relatively short-period (e.g., daily) load-follow power swings, superimposed on the relatively long period (e.g., 3 month) power cycles and "saw tooth" temperature changes from pebbles passing through the core multiple times, results in complex time dependent power and temperature profiles that cannot be closely replicated in a MTR. Accordingly, potential applicants are not expected to include load follow power-temperature profiles in their fuel qualification irradiation testing plans. However, there is not a sufficiently well-established and thorough understanding of the mechanics and properties (e.g., creep) of CFP behavior, failure and fission product release to conclude with certainty that fuel irradiation simulations that do not include load follow transients provide a conservative basis for CFP failure and fission product release. Accordingly, it is important that independent fuel performance models and methods be developed and utilized to conduct sensitivity studies to establish an adequate basis to assess the acceptability of load follow operations. These models and methods would also be needed to assess variations in postulated service conditions and fuel manufacturing or design parameter variations or uncertainties. The NRC confirmatory and exploratory testing would also be used to validate the fuel failure and fission product release models and methods.

It is assumed that PBMR and GT-MHR license applicants or vendors will conduct all fuel testing necessary to support licensing. Such fuel testing should address all significant aspects of the licensing basis and should address: a sufficient range of parameters to cover uncertainties and variations; include a sufficient quantity of fuel to establish a sufficient statistical database; and the range of CFP failure mechanisms and performance factors (e.g., fission product release) applicable or potentially applicable to the licensing basis. It is also expected that such testing will use fuel fabricated at the fuel production facility, utilizing the equipment, processes and methods and quality controls that are identical to those to be used for the production fuel for the (GT-MHR and PBMR) fuel cores. However, some test objectives supporting these fuel irradiation test plans may utilize German or US archive fuel or pre-production fuel.

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It is also assumed that PBMR and GT-MHR license applicants or vendors will utilize analytical models and methods to understand phenomena, to predict fuel performance as part of the licensing safety analysis, or to provide an acceptable basis for tests which do not fully account for all important licensing basis conditions (e.g., load follow). In such cases, it is assumed that such models and methods will be acceptably validated based on applicable test data.

(b) RISK PERSPECTIVE

Fuel performance is the dominant component of risk and risk aversion in modular HTGRs. With respect to fuel, the pubic health and safety case for modular HTGRs is founded on the assumption that: (1) the initial CFP defect fraction due to manufacture is exceedingly small, (2) the additional CFP failure fraction due to fuel burnup is very small, and (3) the potential for CFP failures due to a postulated design basis accident is also very small. For example, for a PBMR, it is expected that the safety analysis will assume that the initial CFP defect fraction is between 10^{-6} and 10^{-5}, the burnup induced additional fuel failure fraction will increase from between 10^{-6} and 10^{-5} at BOL to about 10^{-4} at EOL and no significant fuel failures will occur as a result of postulated accidents. Fission product releases during normal operation and postulated accidents are also expected to be assumed to be very small compared to LWRs. If true, the accident source term would be orders of magnitude below LWRs. Further, based on this presumption, applicants are expected to propose that modular HTGR plants be licensed with a radiological confinement structure rather than a traditional leak tight, pressure containing radiological containment structure. The licensing of HTGRs without a traditional containment will therefor require an extremely high level of confidence in the understanding of fuel behavior and the prediction of fuel performance for normal operation, operational transients, design basis accidents and beyond the design basis accident conditions. The research plan in the fuels areas is intended to achieve the needed confidence and understanding and reduce the uncertainties regarding CFP fuel performance.

(c) RELATED NRC RESEARCH

NRC does not currently have any ongoing research in the area of HTGR fuel performance and qualification. Additionally, related supporting NRC research will be needed. NRC research will be needed to understand and calculate limiting core conditions, including uncertainties, applicable to the fuel. These capabilities and results are expected to include research in the areas of core physics and thermal-fluid analytical codes and methods as well as accident analysis codes and methods. For example, this research is needed for independent confirmation that long-term operation at any core location does not exceed the stated design maximum temperature claimed in the facility safety analysis documents. These nuclear and thermal fluid design-analysis calculations are also needed as input to the fuel performance analysis models and methods.

(d) RELATED INTERNATIONAL RESEARCH AND COOPERATION

The international Atomic Energy Agency (IAEA) has had a number of coordinated research programs related to the technical basis and safety performance aspects of HTGR fuels utilizing CFPs. These research programs are part of the broader International Working Group on Gas Cooled Reactors. The working group and the constituent programs, including the HTGR fuels program area, have served as fora for the international exchange of technical information. Several meetings of technical specialists working in the area of HTGR fuels research and

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development have taken place, beginning in the early 1980s, and continuing during 1990s. Meeting topics have included, HTGR fuel development (1983), fission product release and transport in HTGRs (1985), behavior of HTGR fuel during accidents (1990), response of fuel elements and HTGR cores to air and water ingress (1993) and retention of FP in CFP and transport of FP (1992-1996). The proceedings from these meetings have been published and are publically available. Most recently, IAEA support for the gas reactor working group and associated coordinated research programs has substantially declined although limited periodic meetings among the international experts in different HTGR technology areas including HTGR fuels may still continue for some time.

The European Commission (EC) is currently sponsoring a €20M, 4-year research program on high temperature gas cooled reactors. Several European organizations are participating in this activity. TRISO particle fuels for both fuel compacts (General Atomics) and fuel pebbles (German archive fuel) are being irradiated at the HFR Petten MTR to a burnup of about 80 GWd/t. Following the irradiations the fuel will be heated to 1600°C to study their behaviors under accident heat-up conditions, including FP release and transport through the fuel graphite matrix.

Since 1985 the Japanese Atomic Energy Agency (JAERI) conducted an HTGR R&D program in cooperation with the DOE under a DOE-JAERI memorandum of agreement. Under this agreement joint CFP fuel experiments were conducted and information was exchanged. However, the agreement was terminated in September 1995. Also since 1995, JAERI and the Julich Research Center (KFA) have carried out exchange of information in several HTGR safety arenas including fuel performance. The JAERI-KFA agreement ran from 1996 to 2001. Currently the NRC has an agreement with JAERI covering the exchange of technical information involving safety research and includes aspects such as HTGR fuel technology. A JAERI fuel irradiation test program to qualify the CFP fuel for HTTR operation has been completed and documented. The results were reviewed by the Japanese regulatory authorities in connection with the safety review and licensing of the HTTR. The JAERI fuel testing program has now entered the operational phase in which CFP fuel performance will be assessed on a large-scale as part of HTTR power operations.

In China, the Institute for Nuclear Energy and Technology (INET) is currently conducting an HTGR fuel irradiation qualification testing program for the HTR-10. This testing is being performed on both CFPs and fuel elements that were produced for use in the HTR-10. The fuel is currently being irradiated in a materials test reactor. The fuel elements will be irradiated to burnups of 30,000, 60,000 and 100,000 MWd/t. At each of these burnups, the fuel pebbles will be subject to a temperature increase to simulate design-basis accident temperature conditions. The irradiation testing is a license condition for initial power escalation and long term power operation of the HTR-10. Once the fuel qualification testing is completed, it is expected that the INET fuel testing program will enter the operational phase in which CFP fuel performance will be assessed on a large-scale as part of HTR-10 power operations.

The Massachusetts Institute of Technology (MIT) has established a high temperature pebble bed reactor research project for student research. One area of student research is improved CFP performance modeling. CFP modeling aspects being pursued include: chemical issues (e.g., migration of fission products through coatings, chemical attack (Pd) of SiC, the calculation of temperature distributions inside pebbles, models to predict the mechanical behavior, including failure, of CFPs, finite element models of CFPs, and fracture mechanics based failure models to predict CFP failure probability.

(e) NRC RESEARCH OBJECTIVES AND PLAN

Research Objectives

The overarching objective of the NRC research in the HTGR fuel performance and qualification arena is directed toward developing a sufficient technical basis for the NRC to effectively review and resolve the significant technical and regulatory issues in the area of performance and qualification of HTGR fuels utilizing CFPs. The specific objectives are as follows:

A. NRC HTGR fuels (PBMR and GT-MHR) testing. The purpose of the testing would be to:

1. Confirm an applicant's claims of fuel performance and fission product release;

2. Explore the limits (i.e., margins) of fuel performance and fission product release for parameters which are important to the margins such as fuel operating temperature, fuel accident temperature, fuel oxidizing environment, fuel burnup, energy deposition and deposition rate in fuel due to reactivity accidents.

3. Provide a basis for judging the acceptability of an applicant's fuel irradiation test program (e.g., test methods, QA program, data analysis methods)

4. Provide data for use in developing/validating NRC analytical models and methods.

B. NRC fuel analytical model and methods development. The purpose would be to:

1. Independently evaluate HTGR fuel behavior, including CFP failure, fission product release and margins of safety,

2. Evaluate the effects of variations in irradiation service conditions, and uncertainties (i.e., sensitivity studies).

C. NRC interactions and cooperative research with other international organizations. The purpose would be to:

1. To stay current on design, development, fabrication, testing, operational experience and research with HTGR fuel utilizing TRISO CFP,

2. To leverage NRC resources in pursuit of objectives A and B above.

Research Plan

A. <u>HTGR Fuel Irradiation Testing Plan</u>

The NRC HTGR fuel irradiation testing program plan has three elements. These are testing of unirradiated German archive pebble fuel fabricated for the AVR, testing of PBMR production fuel for the PBMR demonstration plant and initial PBMR plants that may be built in the US, and

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testing of GT-MHR production fuel compacts that will be used for the initial GT-MHR plants that may be built in the U.S. Table 1 at the end of this section summarizes the NRC independent irradiation testing plan needs for German archive pebble fuel, Table 2 summarizes the NRC independent testing plan needs for PBMR production fuel, and Table 3 summarizes the NRC independent testing plan needs for GT-MHR production fuel compacts.

It is assumed that a joint test program between the NRC and the applicant/vendor is desirable for efficiency and timeliness and should be pursued. That is, the NRC test program elements which the respective applicant/vendor (i.e., PBMR or Exelon for the German archive fuel; General Atomics for the GT-MHR fuel) should plan to include in their own respective fuel performance and qualification irradiation test plans should be identified. In this regard, it is the staff's view that a number of the confirmatory and exploratory fuel irradiations and tests in the NRC test matrix should be part of the applicant's/vendor's required test program. If a joint test program can be agreed upon, NRC funds will only be used for tests that would not be part of the required applicant's test program.

In addition the in reactor irradiation and accident tests conducted by the NRC and the applicant/vendor should be carried out and documented in accordance with an accepted and approved quality assurance program.

B. HTGR Fuel Analytical Model and Methods Development

The NRC should enter into a cooperative agreement with a university or a contract with a national laboratory to develop analytical tools for assessing CFP behavior and fuel element performance, including fission product release and CFP failure. As a first step a search and review should be conducted of: (1) ongoing research aimed at developing tools for performing mechanistic analyses of HTGR fuel performance and (2) existing HTGR fuel performance analysis models and methods tools. The NRC should enter into cooperative agreement or contract agreement to further develop the fuel performance mechanistic model into a state-of-the-art thermal-mechanical-chemical CFP analysis tool for use by the staff. The resulting tool should be benchmarked against existing empirical CFP fuel performance data, other codes and the results of NRC and applicant/vendor fuel performance and qualification test data. A user guide should be developed for use of the tool analytical tool. Sensitivity calculations should be conducted to assess the effects of variations and uncertainties in significant fuel characteristics and reactor core conditions (e.g., load follow). These assessment should be conducted for both the PBMR and GT-MHR fuel designs.

C. NRC Interactions and Cooperative Research

The NRC should participate initially as an observer and later as an active participant in selected IAEA Working Group meetings in which the results of applicable CFP and HTGR fuel research and development activities are presented and discussed. The NRC should consider participating in ongoing or planned HTGR fuel irradiation programs in cases where the irradiation tests are closely applicable to NRC fuel performance and qualification testing objectives an/or model and methods development needs. The fuel testing being conducted by the EC may be an example of such an irradiation program. NRC staff should explore potentially mutually beneficial opportunities on the national and international level.

(f) RESOURCES AND SCHEDULE

The following subsections describe the estimated resource needs and schedules for implementing the respective areas of the research plan.

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HTGR Fuel Testing Plan

It is assumed that the applicant/vendor will agree that half of the testing elements in the NRC program are required elements of the applicant/vendor fuel performance and qualification test program and that the costs will therefor be borne by the applicant/vendor.

Test German Archive Pebbles:

It is assumed that the German archive AVR fuel pebbles will be available for irradiation testing by the NRC beginning in FY2003 of the five-year planning period. During this period, planning including aspects such as MTR core analysis and test capsule design can be conducted. Additionally, it is assumed that the cost of transporting and storing the archive pebbles will be jointly borne by the NRC, DOE and the vendor/applicant. Estimated NRC resources required for the testing of German archive fuel pebbles are tabulated below.

	Fiscal Year				
	2002	2003	2004	2005	2006
FTE	.20	.20	.20	.20	.10
$ K	300	1500	1500	300	100

Test PBMR Production Pebbles:

It is assumed that the PBMR production pebbles will be available for irradiation testing in the final year of the next five-year planning period. Additionally However, during this period, planning, including aspects such as MTR core analysis and test capsule design can be conducted. NRC resources for the testing of PBMR production fuel pebbles are estimated in the following table.

	Fiscal Year				
	2002	2003	2004	2005	2006
FTE	.10	.10	.10	.10	.20
$ K	0	0	0	500	2500

Test GA Fuel Compacts:

It is assumed that the GA production fuel compacts will not be available for irradiation testing during the next five-year planning period. However, during this period, detailed planning activities, addressing aspects such as MTR core analysis and test capsule design for the fuel irradiations, can be conducted toward the end of the planning period. The following table summarizes the estimated resource requirements for these test planning activities.

	Fiscal Year				
	2002	2003	2004	2005	2006
FTE	.05	.05	.05	.10	.20
$ K	0	0	0	0	500

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HTGR Fuel Analytical Model and Methods Development

Estimated resources needed for the development of HTGR fuel analytical models and methods are tabulated below.

	Fiscal Year				
	2002	2003	2004	2005	2006
FTE	.10	.10	.10	.10	.10
$	100	200	100	50	200

Interactions and Cooperative Research

It is assumed that the estimated costs of this element of the research plan, as tabulated below, will be in the categories of coordination and exchange efforts and travel.

	Fiscal Year				
	2002	2003	2004	2005	2006
FTE	.05	.05	.05	.05	.05
$ K	15	15	15	15	15

Total Resource Requirements

The following table summarizes the estimated resource requirments for all activities of the NRC research plan for HTGR fuel performance and qualification.

	Fiscal Year				
	2002	2003	2004	2005	2006
FTE	**.50**	**.50**	**.50**	**.55**	**.65**
$ K	**415**	**1715**	**1615**	**865**	**3315**

(g) PRIORITIES

Confirmatory and exploratory irradiation testing research for PBMR fuel is the highest priority based on the apparent risk importance of the issues and the value to assuring confidence in regulatory decisions. Extremely high costs and very high benefits to NRC

Exploratory irradiation testing research for GT-MHR fuel related to postulated reactivity insertion events is next highest priority based on the apparent risk importance of the issue and the value to assuring confidence in regulatory decisions.

Interactions and cooperative research with other international organizations is next highest priority based on the somewhat lower value to assuring confidence in risk significant regulatory decisions. Relatively low cost and moderate benefits to NRC

Fuel analytical model and methods development is the next highest priority based on the somewhat lower value to assuring confidence in regulatory decisions of somewhat lower risk significance. Potentially moderate cost and moderate benefits to NRC. Overall priority would be higher if a low-cost cooperative agreement could be established for this task.

Other confirmatory and exploratory irradiation testing research for GT-MHR is the lowest priority based on the somewhat duplicative aspects of these tests with the PBMR test apparent risk importance of the issues and the value to assuring confidence in regulatory decisions.

Table 1. German Archive Fuel Irradiation Tests

#	Irradiation Purpose	Burnup Increment (GWd/t)					Safety Test Δ	PIE
		0 to 25	25 to 50	50 to 75	75 to 100	100 to 125		
1	Archive Pebble						N/A	
2	Archive Pebble						N/A	
3	Design Max Fuel Temp+Ramp Hold	-----Accel-----	------Accel------	-----Accel--------	-----Accel--------	Δ	1600°C Ramp Heatup	Y
4	Design Max Fuel Temp + Acc Temp	-----Accel-----	------Accel------	-----Accel--------	-----Accel--------	Δ	1600°C Accid Simulation	Y
5	Design Max Fuel Temp+Ramp Hold	----Accel------	------Accel------	-----Accel--------	-----Accel--------	Δ	1800°C Ramp Heatup	Y
6	Design Max Fuel Temp+Real Time	-- Real-Time--	---Real-Time--	--- Real-Time---	-- Real-Time----	Δ	1800°C Ramp Heatup	Y
7	Design Max Fuel Temp+50° C	----Accel------	-----Accel-------	-----Accel--------	-----Accel--------	Δ	1800°C Ramp Heatup	Y
8	Design Max Fuel Temp+Air Ingress	----Accel------	-----Accel-------	-----Accel--------	-----Accel--------	Δ	1600°C+ Air Ingress	Y

Δ = Burnup at which the safety test is conducted.

All irradiation tests are at the upper bound on burnup with margin (i.e., ~100 MWd/t).

All irradiation tests are for the upper bound on temperature with margin and simulate a sawtooth temperature history.

Irradiations should involve a conservative fast fluence vs burnup history (fluence > max expected fluence vs BU line) for the plant.

Accel = the burnup rate is accelerated compared to the burnup rate expected in the core.

Real time = the burnup rate is about the average real time burnup rate expected for the core.

PIE = Post Irradiation Examination (e.g., leach-burn-leach, micrograph).

Air Ingress = simulates the worst case oxidation expected for the worst case air ingress event with margin.

Table 2. PBMR Production Fuel Irradiation Tests

#	Irradiation Purpose	Burnup Increment (GWd/t)					Safety Test Δ	PIE
		0 to 25	25 to 50	50 to 75	75 to 100	100 to 125		
1	Archive Pebble						N/A	
2	Archive Pebble						N/A	
3	Design Max Fuel Temp	------Accel------	-------Accel------	------Accel-------	------Accel------	Δ	1800°C Ramp Heatup	Y
4	Design Max Fuel Temp+50° C	------Accel------	-------Accel------	------Accel-------	------Accel------	Δ	1800°C Ramp Heatup	Y
5	Design Max Fuel Temp+20K BU	------Accel------	-------Accel------	-------Accel------	-------Accel-----	------Accel------Δ	1600°C Ramp Heatup	Y
6	Design Max Fuel Temp+Real Time	-- Real-Time--	-- Real-Time---	---Real-Time---	-- Real-Time--	Δ	1800°C Ramp Heatup	Y
7	Design Max Fuel Temp+Air Ingress	------Accel------	------Accel-------	------Accel-------	------Accel------	Δ	1600°C+ Air Ingress	Y
8	Design Max Fuel Temp +RIA	Δ					Reactivity Insertion	Y
9	Design Max Fuel Temp +RIA	------Accel------	-------Accel------	Δ			Reactivity Insertion	Y
10	Design Max Fuel Temp +RIA	------Accel------	-------Accel-----	------Accel-------	------Accel------	Δ	Reactivity Insertion	Y
11	Design Max Fuel Temp+Rmp Hold	------Accel------	-------Accel------	------Accel------	------Accel------	Δ	1600°C Ramp Heatup	Y
12	Design Max Fuel Temp +Acc Temp	------Accel------	-------Accel------	------Accel-------	------Accel------	Δ	1600°C Acc Simulation	y

Δ = Burnup at which the safety test is conducted.

All irradiation tests are at the upper bound on burnup with margin (i.e., ~100 MWd/t).

All irradiation tests are for the upper bound on temperature with margin and simulate a sawtooth temperature history.

All Irradiations involve a fast fluence vs burnup which is conservative (fluence above the maximum expected fluence vs BU line) for the plant.

Irradiations should involve a conservative fast fluence vs burnup history (fluence > max expected fluence vs BU line) for the plant.

Accel = the burnup is accelerated compared to the burnup rate expected in the core.

Real time = the burnup rate is about the average real time burnup rate expected for the core.

Air Ingress = simulates the worst case oxidation expected for the worst case air ingress event with margin.

PIE = Post Irradiation Examination (e.g., leach-burn-leach, micrograph).

RIA = Reactivity insertion accident TBD; energy deposition spike TBD (temperature increase over delta time); RIA time history simulation includes later core and fuel heat-up profile to simulate longer term fuel heat-up (e.g., loss of helium cooling due to loss of forced circulation following a reactivity insertion pebble compaction).

Table 3. GT-MHR Production Fuel Irradiation Tests

#	Irradiation Purpose	Burnup Increment (GWd/t)					Safety Test Δ	
		0 to 25	25 to 50	50 to 75	75 to 100	100 to 125		
1	Archive Compact						N/A	
2	Archive Compact						N/A	
3	Design Max Fuel Temp	-----Accel------	------Accel-----	-----Accel-------	-----Accel------	Δ	1800°C Ramp Heatup	
4	Design Max Fuel Temp+50° C	-----Accel------	------Accel-------	-----Accel--------	-----Accel------	Δ	1800°C Ramp Heatup	
5	Design Max Fuel Temp+20K BU	-----Accel------	------Accel-------	------Accel-------	------Accel-----	------Accel-----Δ	1600°C Ramp Heatup	
6	Design Max Fuel Temp+Real Time	-- Real-Time--	--- Real-Time---	--- Real-Time---	-- Real-Time--	Δ	1800°C Ramp Heatup	
7	Design Max Fuel Temp+Air Ingress	-----Accel------	------Accel-------	-----Accel--------	-----Accel------	Δ	1600°C+ Air Ingress	
8	Design Max Fuel Temp +RIA	Δ					Reactivity Insertion	
9	Design Max Fuel Temp +RIA	-----Accel------	------Accel-------	Δ			Reactivity Insertion	
10	Design Max Fuel Temp +RIA	-----Accel------	------Accel-------	-----Accel--------	-----Accel------	Δ	Reactivity Insertion	
11	Design Max Fuel Temp+Rmp Hold	-----Accel------	------Accel-------	-----Accel--------	-----Accel------	Δ	1600°C Ramp Heatup	
12	Design Max Fuel Temp +Acc Temp	-----Accel------	------Accel-------	-----Accel--------	-----Accel------	Δ	1600°C Acc Simulation	

Δ = Burnup at which the safety test is conducted.

All irradiation tests are at the upper bound on burnup with margin (i.e., ~100 MWd/t).

All irradiation tests are for the upper bound on temperature with margin and simulate a sawtooth temperature history.

All Irradiations involve a fast fluence vs burnup which is conservative (fluence above the maximum expected fluence vs BU line) for the plant.

Irradiations should involve a conservative fast fluence vs burnup history (fluence > max expected fluence vs BU line) for the plant.

Accel = the burnup is accelerated compared to the burnup rate expected in the core.

Real time = the burnup rate is about the average real time burnup rate expected for the core.

Air Ingress = simulates the worst case oxidation expected for the worst case air ingress event with margin.

PIE = Post Irradiation Examination (e.g., leach-burn-leach, micrograph).

RIA = Reactivity Insertion accident TBD; energy deposition spike TBD (temperature increase over delta time); RIA time history simulation includes later core and fuel heat-up profile to simulate longer term fuel heat-up (e.g., loss of helium cooling due to loss of forced circulation following a reactivity insertion pebble compaction).

(e) NRC Research Objectives and Plans

The NRC research objectives are to establish and qualify independent analysis capabilities and develop technical insights needed for assessing the adequacy of the applicant's safety analyses. For analytical issues involving reactor neutronics and decay heat generation in the PBMR and GT-MHR designs, the following research activities are planned:

1. Familiarization with Applicant's Codes and Methods: In coordination with pre-application review activities, gain familiarity with the reactor neutronics codes and decay heat algorithms and associated analysis assumptions, validation data, and uncertainty treatments that are being used for the PBMR and GT-MHR licensing-basis safety analyses. Incorporate insights and questions arising from this familiarization process into the prioritization, planning, and execution of the NRC's overall research efforts in this and related technical areas.

2. Initial Exploratory and Scoping Studies: Use available independent codes (e.g., GRSAC, MCNP/MonteBurns, SCALE/NEWT/SAS2D, WIMS/MONK, Venture2000, PEBBED), and available applicant codes where needed, to perform exploratory and scoping analyses on selected issues such as described in Section (a) of this chapter. Incorporate insights and questions arising from these exploratory and scoping studies into the prioritization, planning, and execution of the NRC's overall research efforts in this and related technical areas.

3. Preparation of Modern Cross-Section Libraries: Using the upgraded AMPX code system, supplemented by NJOY as needed, prepare and test state-of-the-art master cross section libraries for use in performing exploratory and confirmatory analyses on reactor safety and material safety issues involving conventional and advanced reactor technologies, including PBMR and GT-MHR.

4. Preparation and Testing of Spatial Kinetics Model: Develop a PARCS input models of the PBMR and GT-MHR reactors, and using appropriate lattice physics depletion analysis tools with state-of-the-art cross section libraries (see previous item), prepare the design-specific nodal data tables needed for performing spatial kinetics analyses with the PARCS code (coupled with TRAC-M thermal-hydraulics).

5. Validation for Depletion and Decay Heat Analysis: Review existing and planned validation databases (e.g., spent fuel isotopic assays and decay heat calorimetry) and perform sensitivity analyses, based on methods developed in recent years under RES sponsorship, to help assess their applicability to PBMR and GT-MHR fuels and operating parameters and to help prioritize further data needs and assess remaining validation uncertainties. Participate in cooperative programs for new experimental data as well code-to-data and code-to-code benchmarking activities.

6. Validation and Testing for Reactor Neutronics: Review existing and planned validation databases (e.g., critical experiments, worth measurements, reactor tests) and perform sensitivity analyses, based on methods developed in recent years under RES sponsorship, to help assess their applicability to PBMR and GT-MHR reactor neutronics phenomena and to help prioritize further data needs and assess remaining validation

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uncertainties. Participate in cooperative programs for acquiring new experimental data and conducting relevant code-to-data and code-to-code benchmarking activities.

(f) Resources and Schedule

The following table summarizes the labor resources and completion dates initially estimated for the six research activities described in the preceding section. The labor resources represent the combined efforts of NRC staff and contractors. A breakdown between NRC staff and contractor resources is not provided. Not included in the table are the costs of acquiring or developing new experimental data through dedicated or cooperative research.

Activity	Estimated Level of Effort	Estimated Completion
1. Familiarization with Applicant's Codes and Methods	4 staff months	September 2002
2. Initial Exploratory and Scoping Studies	8 staff months	January 2003
3. Preparation of Modern Cross-Section Libraries*	12 staff months*	November 2003*
4. Preparation and Testing of Spatial Kinetics Models	12 staff months	May 2004
5. Validation for Depletion and Decay Heat Analysis	12 staff months	December 2004
6. Validation and Testing for Reactor Neutronics	9 staff months	March 2005

* The same work also appears in the ALWR plan. The resulting master cross section libraries will be generically applicable to all reactor types.

(g) Priorities

Fundamental to reactor safety analysis is the ability to adequately predict the fission and decay heat sources that arise under credible normal and accident conditions. High priority must therefore go to the research activities that enable and support the staff's independent assessment of reactor neutronics and decay heat analysis issues. The importance of these planned activities is heightened by the fact that the NRC has had little recent experience at analyzing new reactor designs that differ significantly from current LWRs in their reactor neutronics and decay heat analysis issues.

As outlined in the preceding sections, these high-priority research activities entail developing the staff's technical insights in these areas and applying those insights toward establishing and qualifying independent analysis tools and capabilities. The development activities include the assessment of validation issues and modeling approximations in order to inform the staff's evaluation and treatment of potential biases and uncertainties in the computed nuclear heat

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sources. Also of high priority is the development of state-of-the-art master cross section libraries. The resulting master cross section libraries will play a fundamental role in all nuclear analysis activities for reactor safety and material safety and will be generically applicable to all technologies associated with conventional and advanced reactors.

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III.C.3.5.ii NUCLEAR-GRADE GRAPHITE

(a) Description of Issue

To be able to effectively review the new HTGR designs, there is a need to conduct confirmatory research to establish an information base related to the long-term performance and behavior of nuclear-grade graphite under high temperatures and radiation levels expected under normal operating and accident conditions in high temperature gas-cooled reactors. There is also a need to carefully examine the loss of structural integrity of the nuclear-grade graphite because it is one of the key issues which would impact the performance of the structural elements and the reflector (side and bottom) and also the end-of-life behavior of all the graphite elements, including the moderator balls. It is also important to understand graphite behavior under accident conditions (e.g., air ingress). Various graphite production variables, including coke source, manufacturing experience of nuclear-grade graphite, processing, quality control in uniformity of batches and samples within a batch; testing of production parameters such as density, thermal conductivity, isotropy, fracture toughness, grain size, crystallite size and uniformity are some of the important considerations. In the absence of any national or international standards, acceptance criteria need to be established for suitability of graphite in HTGR applications. The advanced gas-cooled reactor operational experience in UK is related to graphite in service in a CO_2 atmosphere as compared to the inert Helium environment employed in both the PBMR and GT-MHR designs, where graphite is also expected to be exposed to considerably higher operating temperatures. Furthermore, various performance parameters such as effect of temperature, radiation (e.g., burn-up, maximum fluence, radiation levels, cumulative life-time dose), chemical attack and oxidation in the event of an air ingress need to be examined. To be able to effectively review the new HTGR designs with reasonable confidence, NRC should consider conducting research to obtain confirmatory data to assess changes in the physical characteristics of nuclear graphite, such as, swelling and shrinkage; creep; cracking; corrosion; distortion; weight loss and porosity changes.

There are several outstanding questions and issues, that should be addressed by the research:

a. Can "new" graphite be produced to perform at the same level as the "old" graphite? What standards and acceptance criteria should be applied? What performance criteria would be used?

b. Can "old" graphite data be extrapolated to the "new" graphite? What is the validity of applying the UK AGR data that was obtained under comparatively lower operating conditions and in a CO_2 environment, to the new helium-cooled HTGRs?

c. Since "new" graphite will be produced with "old" graphite technology because that is the only available experience and information base, various physical characteristics, such as, grain size, crystallite size, isotropy, fracture toughness, and uniformity, of the "new" graphite would also need to be assessed for application in the current HTGR designs.

d. What should be the scope of a robust graphite qualification program?

e. What confirmatory data would NRC need to develop have a reasonable confidence for reviewing the acceptability of an applicant's graphite qualification program?

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f. Which existing national and international standards are applicable to the nuclear-grade graphite? What new standards should be developed as acceptance criteria for physical characteristics and operational performance of graphite in HTGR applications?

g. What acceptance criteria should be in place for graphite design, manufacturing, testing, sampling, surveillance, in-service inspection plans and techniques?

h. The U.S. is already represented at the IAEA-sponsored graphite database development efforts. What additional international collaborative efforts (e.g., those sponsored by the European Union) that the NRC should participate in for optimum benefit and leveraging cost.

(b) Risk Perspective

In HTGRs, graphite acts as a moderator and reflector as well as a major structural component that may provide channels for the coolant gas, channels for control and shutdown, and thermal and neutron shielding. Additionally, graphite components are employed as supports. Graphite also acts as a heat sink during reactor trip and transients. During reactor operation, many of the physical properties of graphite are significantly modified as a result of temperature and irradiation. There is significant internal shrinkage and stresses which may cause component failure. Additionally, when graphite is irradiated to very high radiation dose, ensuing swelling causes rapid reduction in strength, making the component lose its structural integrity. During normal operation, neutron flux and thermal gradients in the graphite components, including the reflector can cause component deformations, bowing and build-up of significant stresses. In the event of an accident, say air ingress, subsequent graphite oxidation causes further changes in its physical properties.

There may be significant contributions to the overall plant risk in terms of long-term graphite performance, especially, thermal, radiation- and chemically induced changes, such as, loss of structural integrity and consequently, its impact on core geometry. Changes in the physical characteristics of graphite, especially at the end-of-life, may also impact safety. Therefore, implications of the end-of-life issues, both for the moderator balls and the graphite structural elements including the side and the bottom reflector, need a careful assessment.

(c) Related US Experience

(c)(a) Related NRC Research

None at present. Preliminary evaluations are being conducted for planning and implementing a nuclear-grade graphite research program.

(c)(b) Related US Research and Experience

Fort St. Vrain used high purity graphite for the fuel blocks, but not as pure a graphite was used for core support, The latter had a high iron content which was oxidized by moisture resulting in serious loss of strength. However because of extensive design margins, no structural problems

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were encountered. Two fuel blocks, however, cracked as a result of stress-induced lattice crack between coolant holes and the outside of the blocks. Additionally, because of moisture ingress, the FSV licensee, in agreement with the NRC, instituted a surveillance program and at each refueling, remotely examined the core support graphite blocks to ensure that the cracking problem did not continue. It is recommended that at PBMR, in-service examination of graphite moderator balls, using a statistically valid sample size, should be conducted

A recent report by Electric Power Research Institute (EPRI), "Graphite for High Temperature Reactors," dated August 2001, examines nuclear-grade graphite for HTGR applications and compiles pertinent data.

ORNL has done extensive work that was published in ORNL/TM-13661, "Potential Damage to Gas Cooled Reactor Graphite due to Severe Accidents," April 1999. The GRSAC (Reactor Severe Accident Code) will also be relevant to future HTGR research .

(d) Related International Research and Experience

For HTR-10, China imported graphite from US. No new experimental data exist. An appraisal of in-vessel graphite is admittedly very difficult, and the best way to minimize the loss of structural integrity issues of in-vessel graphite components is to limit neutron fluence. British data are available and applicable to HTR.

In HTTR, Japan has used high purity graphite. The HTTR operates at comparatively low radiation levels. No problems thus far. No graphite problems were encountered in Germany during either AVR or THTR operation.

Germany has AVR off-normal operational data, including air and water ingress events as well as subsequent core flooding. In-service inspection at AVR involved pebble removal and carbon dust removal. Fretting of graphite blocks as a consequence of loss of structural integrity was observed. During AVR decommissioning, a huge cavity in the central reflector column was noticed. Its formation was attributed to erosion and thermal distortion.

The European Union, as part of the HTR-N project, which began in November 2000, has planned efforts to review state-of-the-art techniques for determining graphite properties to set up a database and perform oxidation tests at high temperatures on fuel matrix graphite to obtain kinetic data for advanced oxidation (THERA facility at Julich) and on advanced carbon-based materials to obtain oxidation resistence in steam and air, respectively (at INDEX facility at Julich). EU has also proposed that as part of the HTR-N1 project, which was expected to start in November 2001, structural graphite samples -- side reflector -- from the decommissioned AVR will be studied. In addition to the long-term testing of graphite components, the objectives of this program include verifications of models describing the graphite behavior under irradiation and thermal distortions and screening tests for graphite properties.

The Russian nuclear-grade graphite comes from a plant in Siberia. It is a new type of graphite. Extensive cooperation is ongoing between the republics of the former Soviet Union regarding assessment of graphite properties. Russia believes that no final HTGR design should be approved without independent experimental qualification of graphite.

UK has an extensive advanced gas-cooled reactor operating experience. The AGRs employ CO_2 as a coolant and consequently, most of the British data are in a CO_2 environment. Some of this information may not be directly applicable to the currently planned HTGRs that employ helium as a coolant. A comprehensive in-service inspection plan and surveillance program is recommended for monitoring possible graphite degradation.

IAEA

Various IAEA Coordinated Research Programs (CRPs) and publications[1], such as, TECDOC-690, TECDOC--901, TECDOC–1198, TECDOC--1154, IWGGCR--11, IWGHTR--3, deal with the subject of world-wide research and experience related to nuclear-grade graphite. Especially noteworthy are the following:

A specialists' meeting was held on the subject of graphite development for gas cooled reactors at the Japan Atomic Energy Research Institute (JAERI) in September 1991. This meeting was attended by representatives from France, Germany, Japan, the Russian Federation, the United Kingdom and the United States of America. Papers were presented in the topical areas of graphite design criteria, fracture mechanisms and component tests; graphite materials development and properties; and non-destructive examinations, inspections and surveillance of graphite materials and components. TECDOC--690 contains the details.

In 1995, a "Specialists Meeting on Graphite Moderator Lifecycle Behaviour" was held in Bath, UK. Recognizing that many experts in the field are nearing their retirement with no apparent replacement of qualified professionals in the field, the IAEA's objective in sponsoring this meeting was to establish a central archive facility for the storage on irradiated graphite. Twenty-seven papers were published where the experts representing their countries shared the ongoing graphite research and other pertinent experience. Details of international research activities are included in TECDOC--901.

With support from Japan, South Africa and the United Kingdom, the IAEA has established a database related to irradiated nuclear graphite properties[2]. The objective of this effort is to preserve the existing world-wide knowledge on the physical and thermo-mechanical properties of irradiated graphite, and to provide the validated data source to the member countries with interest in graphite-moderated reactors or development of the HTGRs, and to support continued improvement of graphite technology applications. The database is currently being developed and includes a large quantity of data on irradiated graphite properties, with further development of the database software and input of additional data in progress. On-line access will be available to the IAEA member countries. This database is expected to be operational in the year 2003.

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1 http://www.iaea.org/inis/aws/htgr/abstracts/index.html

2 http://www-amdis.iaea.org/graphite.html

Under the auspices of IAEA, the objectives of the International Working Group on Gas Cooled Reactors (IWGGCR) is to identify research needs and exchange information on advances in technology for selected topical areas of primary interest to HTR development, and to establish within these topical areas, a centralized coordination function for the conservation, storage, exchange and dissemination of HTGR-related information. The topical areas identified include irradiation testing of graphite for operation to 1000° C. The duration of this CRP is from 2000 through 2005. This IAEA program is discussed in detail TECDEOC--1198.

NEA

Various NEA conferences held in the past few years have covered the subject of nuclear-grade graphite:

From September 27-29, 1999, NEA/OECD held in Paris the first information exchange meeting on "Survey on Basic Studies in the Field of High Temperature Engineering."[3] The conference was co-sponsored by JAERI. Component behavior, including graphite performance, under normal and accident conditions were discussed. Some of the topics presented include status in the UK and the Netherlands of research relevant to irradiation of fuels and graphite for HTGRS; oxidation of carbon based materials and air ingress accidents in HTR-modules being studied at Julich; graphite selection for the PBMR reflector; study of crack growth in nuclear; the modeling of dimensional change in nuclear graphite; and irradiation effects on carbon-carbon being investigated in Japan.

On October 10-12, 2001, there was an NEA/OECD conference held on "The Second Information Exchange Meeting on Basic Studies in the Field of High Temperature Engineering," in Paris. In the afternoon of the 11th, there was a session dedicated just to "Basic Studies on Behavior of Irradiated Graphite/Carbon and Ceramic Materials including Their Composites under both Operation Storage Conditions" - 8 papers were presented - the last one on the status of the IAEA Graphite Database. Proceedings are not yet available.

International Standards

International cooperation is also crucial in establishing consensus standards, as well as for developing acceptance and performance criteria, for nuclear-grade graphite. It is important to determine which existing national and international standards are applicable to the nuclear-grade graphite, and what, if any, new standards should be developed as acceptance criteria for physical characteristics and operational performance of graphite in HTGR applications. Various American Society for Testing and Materials (ASTM) standards, such as, C781-96 Standard Practice for Testing Graphite and Boronated Graphite Components for High-Temperature Gas-Cooled Nuclear Reactors, and others, would need to be examined for applicability to the nuclear-grade graphite in the new HTGRs.

Cooperation with Other Countries

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3 http://www.nea.fr/html/science/htemp/iem1/session1.html

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In October 2001, NRC held a High-temperature Gas-Cooled Reactor Safety and Research Issues Workshop in Rockville, MD. At this 2-1/2 day workshop, representatives from Germany, UK, European Union, China, Japan, the Russian federation, Republic of South Africa, IAEA, as well as from the Department of Energy and various DOE national laboratories, and two members of the Advisory Committee on Reactor Safeguards discussed various safety and research issues. Various HTGR accident scenarios (such as air ingress, loss of forced circulation, and seismic events), which could possibly lead to release of radioactive material, were examined. Several key safety issues, which warrant further examination, including likely candidates for possible cooperative research were also identified. Long-term graphite behavior under normal operating as well as accident conditions was one of the several issues discussed at this workshop. Specifically, qualification of structural graphite, oxidation, and in-service inspection plans and techniques were discussed. Evaluation of long-term behavior of graphite, such as temperature-, radiation- and chemically-induced changes in physical characteristics, oxidation measurements, and in-service inspection methods were assigned a high priority. The past experience from UK, Germany, and more recently, from Japan and China would be extremely beneficial. Need for confirmatory research in some areas, and priorities were identified. These include investigating: (1) applicability of the "old" graphite data to the "new" graphite; (2) qualification of "new" graphite for HTGR applications; (3) physical property changes (e.g., growth; stress; corrosion/weight loss; failures; graphite dust generation, deposition and oxidation); (4) distortion of structural elements and changes in core geometry; (5) distortion of control elements and possible failure to scram -- the last two in the event of a seismic occurrence.

(e) NRC Research Objectives and Plans

NRC research should be directed towards developing the technical basis to enable the NRC to effectively review various graphite issues. Future research should answer some of the most fundamental questions: What are acceptable graphite design criteria? What standards should be applied to fabrication and structural design of nuclear-grade graphite? What is the impact on physical properties of nuclear-grade graphite (including oxidation, thermal properties, structural properties, and neutron moderating characteristics) as a function of temperature, irradiation? What in-service examinations, inspections and surveillance should be performed on graphite and how should these be done? What is the impact of radiation, temperature and chemically-induced physical characteristics and the resulting impact on safety?

To be able to achieve these objectives, research related to the following broad categories needs to be considered:

(1) Physical Characteristics of Nuclear-Grade Graphite -- manufacturing and design.

- Need to develop nuclear-grade graphite design criteria; institute parameters to control process for nuclear-grade graphite development; establish acceptance standards; develop quality control/assurance standards; establish standards and acceptance criteria for physical characteristics of nuclear-grade graphite, and inspection/surveillance requirements.

- For PBMR, given the Exelon's desire to use AGR fuel sleeve graphite for the replaceable and permanent graphite structures in the PBMR core, what information is available in the UK for the production history of fuel sleeve graphite? Particularly, is the current material (Nitetsu pitch coke) substantially different from the earlier material (VFT coke) with respect to physical properties and property variability?

- Given that the UK approach to probabilistic assessment of graphite performance is to be adopted by PBMR, (i) what are the most important factors to be in the graphite design; and (ii) what graphite materials property data is needed to define property distributions for the purpose of these analysis?

(2) Graphite Qualification Program – irradiation and oxidation behavior of graphite – to determine impact of temperature-, radiation- and chemically-induced changes on physical characteristics, especially, loss of structural integrity and neutron moderating characteristics, and the resulting impact on HTGR safety?

Applicability and adequacy of the existing graphite irradiation data to the PBMR?

Impact of carbon/graphite dust arising from attrition/abrasion of the fuel pebbles be treated separately in air ingress accident studies -- should a separate oxidation kinetic data set be established for the dust/deposit arising from the fuel pebbles?

Necessary test program for graphite materials.

(3) Surveillance, In-service Inspection and Surveillance Plans and Techniques - to achieve

Representative samples of statistically valid size, with acceptable sampling frequency. Reasonable confidence in the surveillance and in-service examination during operation to ensure that the graphite performance is as predicted. This would include that the examination techniques are adequate and the samples are true representatives. Furthermore, acceptance criteria must be clearly defined.

(4) International Consensus Standards for Nuclear-Grade Graphite

Determine applicability of the ASTM graphite standards. NRC may also invite the international community, industry organizations and professional societies to participate in establishing international consensus standards, as well as acceptance and performance criteria for nuclear-grade graphite.

(f) <u>Resources and Schedule</u>

a. **Physical Characteristics of Nuclear-grade graphite**:

Characterization of the key physical properties of full size blocks of PBMR reflector graphite (based on AGR fuel sleeve graphite), establishing in-block, and batch-to-batch variability:

Estimated cost: $1000k. Period of performance: 24 months.

b. **Graphite Qualification Program:**

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(a) Conduct a review of available high dose irradiation data for nuclear grade graphite, including data from ORNL taken under the DOE NP-MHGTR program that has not been published:

Estimated cost: $120k.
Period of performance: 6 months.

(b) Determine air oxidation kinetics data required for core performance and safety modeling for: (i) PBMR reflector grade graphite, (ii) fuel pebble matrix graphite, (iii) graphite/fuel pebble dust.

Estimated cost $200k.
Period of performance: 12 months.

(c) Conduct high dose graphite materials test reactor experiments on PBMR graphite and GT-MHR graphite. Two HFIR target capsules at each of three temperatures (total of six capsules). Two graphite irradiation creep experiments (HFIR RB position).

Estimated cost $3,500k-4,000k (excluding neutron costs)
(Perhaps DOE would pay for some of the tests.)
Period of performance: 48 months.

c. Determining Acceptability of the Existing Standards or Development of New Consensus Standards for Graphite Design and Fabrication for HTGR applications

Design and fabrication standards are needed for nuclear-grade graphite. Also needed are acceptance and performance criteria for graphite performance in the HTGR applications. NRC should consider taking lead in developing consensus standards by inviting international community, industry organizations and professional societies.

Estimated Cost:	TBD
Duration:	TBD

(g) Priorities:

Confirmatory research related to nuclear grade graphite is a high priority item.

III.C. High Temperature Gas-Cooled Reactors

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6 Instrumentation and Control

6a Description of Issues

The new generation of advance reactor concepts, both for high temperature gas cooled reactors and for advance light water reactors will be the first opportunity for vendors to build new control rooms in this country. The advances that have been made in the development of many of the current generation of operating reactors in other parts of the world will be used in the design and construction of any new plant constructions in the US. The new plant will have fully integrated digital control rooms, at least as modern as the N4 reactors in France or the ABWRs. In addition the desire for much smaller control room staffs for economic reasons will also push the plants in the direction of more automation as has been seen in the natural gas fire power plants. The use of multiple modular plants will also require significantly more complex control of both the primary instrumentation and control systems and all of the support systems including the switch yard.

These plants will be designed for autonomous operation with a minimum of supervision by plant operators for long periods of time. This will include automated startups and shutdown, and changes of operating modes. The operating crews will be very small compared with current generation nuclear power plants, as few as three operators for ten modules. This will require that not only normal operations but off normal operations and recovery be much more highly automated. To make modular reactor concepts effective the plant must work like a single larger plant. The will require a level of automation and coordination that is heretofore unheard of in the nuclear power industry. How plant control and safety systems will deal with reorganize itself to deal with partial failures of interconnected particularly at the switch yard and the control room will need to be investigated.

Because of the longer fuel cycles and much longer time between maintenance outages the plants will require much more extensive use of on-line monitoring and diagnostics, and predictive maintenance. Instrumentation will be needed to support this increased automated surveillance and the issue of how these systems will integrate with the control systems will need to be investigated. Additionally, because some of the systems that will be used as part of this new generation of high temperature gas cooled reactors will be operating in temperatures and other conditions that they have not previously been used at it is expected that several new kinds of sensors will be developed to support these reactors. These new sensors will need to be investigated. There may be temperature, pressure, flow and neutron detectors use in this kind of reactor that will require changes in the way we do design and safety calculations (drift, calibration, response time, etc)

It has been seen in the application of highly automated control rooms in other industries that the availability of computer resource to use of modern control theory controllers to increase plant availability, and decreased workload on operators will drive control and diagnostic systems in this direction. It is likely that these new high temperature gas cooled reactors will use some of these advance modern control methods. These could include simple feed forward controllers, non-linear controllers, neural-fuzzy controllers or even more exotic methods. Likely control algorithms that may be used in high temperature gas cooled reactors will need to be reviewed.

6b Risk Perspective

Instrumentation and control systems have potentially very high risk importance to plant safety, primarily because a failure can prevent operators or automated systems from performing their intended safety function. In most risk analysis performed to date have not identified instrumentation and control systems (including RPS and ESFAS) as risk important primarily due to the multiple redundances in these systems and there general lack of common mode failure. However, a failure in these systems can prevent the safe shutdown of the plant or permit other unattended actions to occur. For this reason these systems tend to be very important in assuring maintaining a set level of risk (high Risk Achievement Worth).

For temperature gas cooled reactors the slow response time of the reactor itself and the relative benign response to ATWS transient the risk important accidents that will include instrumentation and control failures will transient initiated accidents by causing unexpected actions. These could include the inadvertent venting of the coolant, loading or discharging of the fuel events (for PBMR), reactivity events, or load rejection events.

In all risk studies to date few instrumentation and control systems have been modeled in any detail. To adequately understand how the much more complicated digital instrumentation and control systems will be used and to support the risk informed licensing that is proposed for high temperature gas cooled reactors this area of risk modeling will need to be expanded. This will also be needed to support the interface with control room human interface research. Because of the lack of adequate models and data to support risk analysis the uncertainties in this area are relatively high and can only be reduced by significant new research in this area.

6c Related NRC Research

The NRC Research Plan for Digital Instrumentation and Control (SECY-01-155) outlines current and future research into several areas of emerging instrumentation and control technology and applications that will be used in the high temperature gas cooled reactors. These included smart transmitters, wireless communications, advanced predictive maintenance and on-line monitoring methods and enhanced cyber security issues. The NRC has recently started new research programs in the areas of wireless communications and on-line monitoring. This research will support the development of review guidance for NRR for these new and improve technologies, and will also support there introduction into advance reactors.

6d Related Research and International Cooperation

The national and international research community has been involved with research and development of advance control and monitoring systems for nuclear power plants for many years. The international community, particularly in Europe, Japan and Korea have developed integrated advance control rooms and done much more research in the areas of automation of plant operations and advance plant monitoring and diagnosis that we have done in the US. There will be significant opportunities for international cooperation in this area.

GA is doing detailed control systems design studies using plant simulators to help optimize control system design. PMBR corporation is also looking into advance control systems. These vender research and development is being carried out both by the venders and by joint efforts with other organization such as the universities and national labs, including ORNL and INEEL. There may be an opportunity to piggy back on to some of these research programs, particularly in the areas of advanced control algorithms and control of multiple plant modules.

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One of the major areas of research outlined on the Department of Energy (DOE) Long-term Nuclear Technology Research and Development Plan is the Instrumentation and Control area. Several of the research topics proposed in this plan are of particular interest to high temperature gas cooled reactors including advance instrumentation, such as robust communications and wireless sensors, smart instrumentation and condition monitoring. Also of interest were research into distributed computing, condition monitoring, advanced control algorithms, on-line monitoring.

As part of the implementation of this long term research plan DOE has developed NERI six programs in this area. These include research in the areas of automatic generation of control architectures, self diagnostic monitoring systems, smart sensors and advance instrumentation to support high temperature gas cooled reactors. It will be important for the NRC to use the results of this work in its

6e NRC Research Objectives and Plans

NRC will need to conduct research into the following areas in support of advance reactor analysis for high temperature gas cooled reactors :

1) Review of current practices and lesson learned from ABWR and N4 control system development

This is an effort that has to be performed for each type of reactor for which sufficient information is available. The review for reactors that have both operational experience and design lesson learned will be the first priority. The effort would also have to be continued over time as new information becomes available.

2) New risk models for instrumentation and control systems in advance reactors

This is effort will complement the current work that is on-going at the University of Maryland and the University on Virginia, but would develop risk models for advance reactor instrumentation and control systems for review of the possible safety issues of the systems and for integration into advance reactor risk models.

3) Analysis of the requirements and potential issues involved with high temperature gas cooled reactor instruments

This effort would review the requirements for and the development of new instruments to support design, construction and operation of high temperature gas cooled reactors. These will include new neutron detectors, particularly for PBMR, temperature sensors, etc. This effort will also support the review of needed prototype plant instruments.

4)Development of models of Autonomous control

This effort will include the development of information and models to review and examine advance autonomous control methods that will be used in advance reactors. The effort will review both current methods used in other areas, such as natural gas power plants and methods that have been proposed by the vendors.

5)Analysis control systems used to integrate the control of multiple module plants

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The amount and way systems will be integrated in advanced reactors using multiple modules will be investigated. At what points control and safety systems are integrated and the amount of automated actions will be reviewed

6) Analysis of on-line monitoring systems and methods and advanced diagnostic methods needed to support high temperature gas reactors.

This effort will review both current methods and investigate the required development of instruments and techniques to support this the current availability and maintenance schedules.

7) Review of advance control algorithms for application to advanced reactors.

The effort will develop information on the current methods likely to be used in advanced reactors and investigate the potential issues with these algorithms when used in a reactor setting.

These research program will be supplemented by the ongoing research program in digital instrumentation and control as outlined in SECY-01-155, "NRC

6f Resources and Schedule

1 Review of current practices and lesson learned from ABWR and N4 control system development

FY	2002	2003	2004	2005	2006
FTE	0.2	0.2	0.1	0.1	0.1
$K	$50K	$100K	$50k	$50K	

2 New risk models for instrumentation and control systems in advance reactors

FY	2002	2003	2004	2005	2006
FTE	0.2	0.3	0.3	0.3	0.3
$K	$300K	$300K	$200K	$150K	$150K

3. Analysis of the requirements and potential issues involved with high temperature gas cooled reactor instruments

FY	2002	2003	2004	2005	2006
FTE	0.2	0.2	0.2	0.1	0.1
$K	$50K	$250K	$100K	$100K	$100K

III.C.4. NUCLEAR ANALYSIS

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(a) Description of Issues

The term "nuclear analysis" describes all analyses that address the interactions of nuclear radiation with matter. It, thus, encompasses the analysis of: (a) fission reactor neutronics, both static and dynamic, (b) nuclide generation and depletion, as applied to reactor neutronics and to the prediction of decay heat generation, fixed radiation sources, and radionuclide inventories potentially available for release, (c) radiation transport and attenuation, as applied to the evaluation of material damage fluence, material dosimetry, material activation, and radiation protection, and (d) nuclear criticality safety, i.e., the prevention and mitigation of critical fission chain reactions ($k_{eff} \geq 1$) outside reactors.

This version of the HTGR nuclear analysis research plan addresses only the nuclear analysis issues associated with reactor neutronics and in-reactor decay heat generation. Future revisions of the plan will address additional nuclear analysis areas as they pertain to anticipated issues arising in the evaluation of HTGR reactor safety and material safety. The following subsections begin with a brief discussion of the nuclear data libraries that are fundamental to all areas of nuclear analysis and continue with discussions of HTGR reactor neutronics and decay heat analysis issues.

Nuclear Data Libraries

All areas of nuclear analysis make use of nuclear data libraries derived from files of evaluated nuclear physics data, such as ENDF/B in the U.S., JEF in Europe, or JENDL in Japan. The nuclear data files include, for example, fundamental data on radionuclide decay as well as neutron reaction cross sections, emitted secondary neutrons and gamma rays, and fission product nuclide yields, all evaluated as complex functions of incident neutron energy. The neutron reaction evaluations also provide cross-section uncertainty information in the form of covariance data that can now be processed and used with advanced sensitivity and uncertainty analysis techniques, as developed in recent years under RES sponsorship, to assist in the identification and application of appropriate experimental benchmarks for problem-specific code validation.

Many of the processed nuclear data libraries in use today were developed in the 1980s or earlier. For example, the PBMR design team in South Africa now relies on the German VSOP reactor physics code with multi-group nuclear cross section libraries derived in the early 1980s from the evaluated physics data in ENDF/B-IV. Pre-1990s cross section libraries are similarly being used for preparing the nodal data used by the NRC's reactor spatial kinetics code, PARCS, and for the criticality, depletion, and shielding analysis sequences in the NRC's SCALE code system. While these legacy cross section libraries have proven largely adequate in a variety of applications, they have known limitations and shortcomings and cannot be described as state-of-the-art.

In response to a 1996 user need memorandum, RES sponsored a program at ORNL to upgrade the AMPX code suite for its eventual use in creating new cross section libraries that would take full advantage of the expanded resolved resonance ranges and the improved/corrected nuclear data and covariance evaluations now available in the latest releases of ENDF/B-VI and its foreign counterparts JEF2.3 and JENDL-3. With the recently completed AMPX upgrades as well as continued improvements to the NJOY nuclear data processing codes, both opportunity and motivation now exist to produce and test state-of-the-art nuclear data libraries for use in the analysis of reactor safety and material safety issues associated with conventional and advanced reactor technologies.

Reactor Neutronics and Decay Heat Generation

The nuclear heat sources of importance in all reactor safety analyses are from nuclear fission and the decay of radionuclides produced by nuclear fission and neutron activation. Reactor neutronics codes are used to predict fuel burnup and the dynamic behavior of neutron-induced fission chain reactions in response to reactor control actions and system events. Under subcritical reactor conditions, where the self-sustaining fission chain reactions have been terminated by passive or active means, the decay of radioactive fission fragments and activation products becomes the dominant nuclear heat source. Note that the 1986 Chernobyl accident involved a rapid power excursion from a prompt supercritical fission chain reaction, whereas the 1979 Three Mile Island accident involved inadequate removal of decay heat from a subcritical reactor.

The defining features of HTGRs include their use of fission-product retaining coated fuel particles, graphite as the moderator and structural material, and neutronically inert helium as the coolant. Both the PBMR and GT-MHR are modular HTGR designs that are fueled with low-enrichment uranium (LEU) instead of high-enrichment uranium (HEU) and thorium used in earlier HTGRs. Both also have long annular core geometries and locate control and shutdown absorbers in the graphite reflector regions. Therefore, in many respects, the PBMR and GT-MHR designs have similar code modeling and validation issues for the prediction of reactor neutronics phenomena and decay heat generation.

Reactor neutronics and decay heat analysis issues unique to the PBMR relate mainly to the following: (i) its use of multiple-pass on-line fueling; (ii) its pebble-bed annular core with statistical packings of fuel pebbles of widely varying burn-ups; (iii) the intermixing of graphite pebbles and fuel pebbles near the boundaries between the fueled core region and the central graphite region, and (iv) the potential for misloading events, anomalous packing and clustering of pebbles, and anomalous flow patterns of pebbles through the core such as those might be caused by localized pebble bridging, jamming of chipped or fractured pebbles, unanticipated funneling effects near the core exit, or unanticipated radial gradients of pebble flow velocity resulting from the strong temperature dependence of pebble-to-pebble friction (e.g., as seen in the THTR-300 pebble bed reactor). Core physics analysis issues unique to the GT-MHR relate mainly to the following: (i) the effects of burnable poisons; (ii) the presence of both "fissile" and "fertile" coated fuel particles (with 19.9% enriched and natural uranium, respectively) in the fuel compacts; (iii) reactivity control for cycle burnup effects; and (iv) the power shaping effects of zoned fuel and poison loadings.

Potentially safety-significant core physics analysis issues anticipated for the PBMR and GT-MHR designs include the following:

(1) <u>Temperature coefficients of reactivity</u>: The staff's safety evaluations should be able to confirm that the reactivity effects from temperature changes in the fuel, moderator graphite, central graphite column, and outer reflector graphite elements are adequately treated in the applicant's safety analyses. Based on sensitivity analyses and validation

against representative experiments and tests, the evaluations should assess and account for computational uncertainties in the competing physical phenomena, including for example the positive contributions to the fuel and moderator temperature coefficients associated with ^{135}Xe and bred fissile plutonium.

(2) Reactivity control and shutdown absorbers: The reactivity worths of in-reflector control and shutdown absorbers are expected to be sensitive to tolerances in the radial positioning of the absorbers. The evaluations for reactivity control and hot and cold shutdown should also account for absorber worth variations through burn-up cycles (GT-MHR) and the transition from initial core to equilibrium core loadings as well as absorber worth variations caused by temperature changes in the core and reflector regions, xenon effects, pebble flow aberrations, and accidental moisture ingress.

(3) Moisture-ingress-induced reactivity: Although the absence of high-pressure, high-inventory water circuits in closed Brayton cycle systems makes this less of a problem than in earlier steam cycle HTGRs, the effects of limited moisture ingress will nevertheless have to be evaluated for depressurized or underpressurized accident conditions in both the PBMR and GT-MHR. Effects to be evaluated include the moisture-induced reactivity itself (i.e., from adding hydrogenous moderator to the undermoderated core) as well as the effects of moisture on temperature coefficients (e.g., from spectral softening), shortened prompt-neutron lifetimes (i.e., faster thermalization), and reduced worths of in-reflector absorbers (i.e., fewer neutrons migrating to the reflector).

(4) Reactivity transients: T/H-coupled spatial reactor kinetics analyses will be needed for assessing axial xenon stability as well as reactivity transients caused by credible events such as overcooling, control rod ejection, rod bank withdrawal, shutdown system withdrawal or ejection, seismic pebble-bed compaction, and moisture ingress. For loss-of-cooling, passive-shutdown events with failure of the active shutdown systems (i.e., anticipated transient without scram (ATWS)), the delayed re-criticality that occurs after xenon decay may also require spatial kinetics analysis models to account for the unique spatial power profiles and feedback effects caused by the higher local reactivity near the axial ends and periphery of the core where temperatures and xenon concentrations are lower.

(5) Pebble burnup measurements and discharge criteria: The PBMR designer states that selected fission-product gamma rays will be measured to determine the burn-up of each fuel pebble and that this measured burnup will serve as the criterion for discharging the pebble or passing it back through the reactor. The particular burn-up value used as the discharge/recycle burn-up criterion will be chosen to limit the maximum pebble burnup, which is stated as nominally 80 GWd/t. Therefore, determining a suitable value for discharge/recycle burn-up criterion (<80 GWd/t) will require consideration of in-core pebble residence time spectra, together with supporting neutronics calculations, in order to statistically characterize the maximum burn-up increment that might accrue during a pebble's final pass through the core. Burnup measurement uncertainties will also have to be considered. Furthermore, since pebble burnup measurements (unlike the pebble reactivity measurements used in THTR-300) cannot distinguish pebbles with different initial fuel enrichments, the same discharge burnup criterion must be applied to the initial charge of 4%-enrichment fuel pebbles as to the 8%-enrichment pebbles that are added

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in transitioning to an equilibrium core. Neutronics calculations will be needed to bound the higher neutron fluence experienced by the 4%-enrichment pebbles in reaching the maximum burnup levels allowed in the transitional cores.

(6) Pebble-bed hot spots: The results of melt-wire experiments conducted in the German AVR test reactor demonstrate the existence of local hot spots under normal operating conditions in pebble bed cores and that such hot spots determine the maximum normal operating temperatures of the fuel. These hot spots may arise from a combination of higher local power density (e.g. due to moderation effects near the reflector wall or from chance clustering of lower burn-up pebbles), lower local bed porosity due to locally tight pebble packings, and reduced local helium flow due to the increase of helium viscosity with temperature. Whereas the slow evolution of loss-of-cooling heat-up transients in the PBMR will tend to wash out any effects of pre-accident local flow starvation on subsequent peak fuel temperatures, the effects of higher local fission power densities will be retained throughout the heatup transient in the form of higher local decay heat powers. Therefore, the effect of decay-power hot spots, in particular, may need to be considered in evaluating the maximum fuel temperatures arising in pressurized or depressurized loss-of-cooling accidents.

(vii) Pebble fission power densities and temperatures: While the normal overall flux and power profiles in a PBMR may be well approximated by skewed axial cosine and radial Bessel functions, each computational node in the core model (e.g., an R-Theta-Z mesh) will generally contain fuel pebbles with a wide range of burnups (i.e., a statistical combination of 1st-pass, 2nd-pass,..., Nth-pass pebbles) and, hence, a wide range of pebble power densities. The computational models may therefore need to account for pebble-to-pebble burn-up and power variations within nodes. Note that in calculating operating temperatures inside a pebble, the reduction of pebble power with pebble burnup may tend to be offset by the reduction of graphite thermal conductivity with neutron fluence.

(7) Pebble decay heat power densities: Much as with fission power densities (see previous item), each node in the core calculational model will contain pebbles with a broad range of decay heat power densities. Computational studies may therefore be needed to establish technical guidance (and possibly a technical standard analogous to ANS 5.1 for LWRs) on accepted modeling approximations and assumptions (e.g., nodal averaging methods) for calculating decay heat sources in pebble bed reactors while accounting for validation uncertainties associated with the shortage of applicable experimental data.

(8) Graphite annealing heat sources: Although continuous annealing effectively prevents any significant build-up of Wigner energy at the high operating temperatures of HTGR graphite, there is nevertheless a significant accumulation of higher-energy graphite lattice distortions that anneal out only at the elevated graphite temperatures encountered in loss-of-cooling accidents (e.g., conduction cooldown events). This high-temperature annealing heat source should be evaluated and, where significant, added to the nuclear decay heat sources used in the analysis of loss-of-cooling heatup events. (Note that the recovered thermal conductivity caused by high-energy lattice annealing during slow graphite heatup accidents can substantially reduce the peak fuel temperatures reached

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during the accident, an effect that has traditionally been credited in the heat removal models used for MHTGR accident analyses.)

Other Nuclear Analysis Issues

Other areas where nuclear analysis issues may arise relate to (a) in-reactor radiation shielding analysis and fluence dosimetry and (b) the criticality safety analyses needed for the higher-enrichment fuels used by the PBMR and GT-MHR designs.

With regard to radiation shielding, issues may arise concerning the prediction and monitoring of local fluence peaks that could result from radiation streaming in any gaps the may develop over time between graphite reflector blocks.

Enrichment plants, fuel fabrication facilities, and transportation packages for LWR fuel assemblies are presently designed, analyzed, and licensed to handle uranium enrichments up 5 wt% ^{235}U. Criticality validation issues are expected to arise due to the shortage of evaluated critical benchmark experiments involving neutron moderation by graphite, materials with 5 to 20% enrichment, and particle fuel geometries. In addition, technical guidance may be needed on the modeling of particle fuel forms, which are generally much more reactive than would be predicted by simplified computational models that smear the fuel into a homogeneous mixture.

(b) Risk Perspective

The results from accident sequence analyses provide information that may be used in plant PRAs for assessing event consequences and their probabilities. Core neutronics codes, generally coupled with thermal-fluid dynamics and/or severe-accident systems codes, are needed for evaluating the dynamic progression of accident sequences that involve reactivity transients. For accident sequences in which the self-sustaining fission chain reaction is terminated by active or passive means, the codes used in evaluating the thermal response of the subcritical system (e.g., maximum fuel temperatures) must employ algorithms that adequately represent the intensity, spatial distribution, and time evolution of the decay heat sources.

(c) Related NRC Research

Related past, ongoing, and planned NRC research efforts include the following:

(1) RES in-house analysis and contractor projects conducted in the late 1980s and early 1990s in supporting the staff's preapplication safety evaluation of the DOE-supported MHTGR.

(2) Recently completed RES-sponsored work on (1) upgrading the AMPX code system for use in creating state-of-the-art nuclear data libraries and (2) the development of sensitivity and uncertainty analysis methods that utilize cross section covariance data.

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(3) Ongoing RES projects and tasks: (1) Modular HTGR Accident Analysis (ORNL), (2) TRAC-M model development for modular HTGRs, (3) Initial PARCS code modifications to incorporate the R-Theta-Z geometry needed for PBMR analysis,

(4) Future NRC research on the HTGR technical areas described in other chapters of this research plan (e.g., thermal-fluid dynamics, severe accidents, graphite).

(d) Domestic and International Cooperation

Opportunities for related domestic and international cooperation include the following:

(1) Establish a cooperative research agreement with MIT that includes sharing of pebble-bed reactor physics codes and models as well as related code development and analysis tasks.

(2) Acquire HTGR physics benchmark data from the international HTR-PROTEUS program conducted in the early 1990s at PSI, Switzerland.

(3) Acquire HTGR physics benchmark data from Russia, including GROG and ASTRA experiments as well as any newer physics experiments supporting the design and safety analysis for the Pu-burning GT-MHR in Russia. Also pulsed test data on fresh HTR fuel.

(4) Evaluate feasibility and technical merits of acquiring existing benchmark data from British Magnox, AGR, and early HTR programs, including BICEP, Dungeness B, and various HTGR-related experiments done in the 1970s by Winfrith and British Energy.

(5) Acquire existing HTGR core physics benchmark data from the AVR tests and KAHTR experiments in Germany and the CESAR experiments in France.

(6) Acquire existing and new HTGR core physics benchmark data from HTR-10 in China.

(7) Acquire existing and new HTGR core physics benchmark data from VHTRC and HTTR in Japan.

(8) Join and add new core physics benchmarking activities to the IAEA's ongoing CRP on safety performance of HTGRs. Such activities could include code-to-code benchmarks, but might also introduce additional experimental benchmarks taken from various sources such as recent and planned benchmark measurements at HTR-10 in China and HTTR in Japan, as well as a number of potentially relevant past experiments and operating tests from British experience with Magnox, AGR, and HTR technology. Note that the proposed additional benchmarking efforts would fill a number of validation gaps not addressed by programs to-date, including the international HTR-PROTEUS experiments described in IAEA TECDOC--1198.

(9) Participate in existing and propose new physics benchmarking efforts within the OECD/NEA's Nuclear Science and/or Nuclear Safety activities related to HTGRs.

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4 Development of models of Autonomous control methods

FY	2002	2003	2004	2005	2006
FTE	0.2	0.2	0.2	0.2	0.2
$K	$50K	$200K	$200K	$200K	$200K

5 Analysis control systems used to integrate the control of multiple module plants

FY	2002	2003	2004	2005	2006
FTE	0.2	0.2	0.2	0.2	0.2
$K	$100K	$100K	$200K	$200K	$200K

6 Analysis of on-line monitoring systems and methods and advanced diagnostic methods needed to support high temperature gas reactors

FY	2002	2003	2004	2005	2006
FTE	0.1	0.1	0.1	0.2	0.2
$K	-	$50K	$100K	$200K	$100K

7 Review of advance control algorithms for application to high temperature gas cooled reactors

FY	2002	2003	2004	2005	2006
FTE	0.1	0.1	0.2	0.2	0.2
$K	-	-	$150K	$150K	$150K

6g Priorities

1) Review of current practices and lesson learned from ABWR and N4 control system development

This effort is **HIGH** priority, since it will support as a basis for determining what other work would need to be done.

2) New risk models for instrumentation and control systems in advance reactors

This work is **HIGH** priority, since it would result in methods and tools that could be used in licensing as well as feeding into other ongoing work for operating reactors.

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3) Analysis of the requirements and potential issues involved with high temperature gas cooled reactor instruments

This work is **HIGH** priority, since it would result in the development of issues that would be needed to advice the vendors as to possible design issues with there plants. The review guidance developed in this research will also provide NRR with the information needed to review these instruments and there application to high temperature gas cooled reactors.

4) Development of models of Autonomous control

This work is **HIGH** priority, since it will result in needed information on the use of autonomous control in high temperature gas cooled reactors which will be an important aspect of licensing these reactors.

5) Analysis control systems used to integrate the control of multiple module plants

This work is **HIGH** priority, since it will result in needed information on the integration of control systems in multiple module advance reactors which will be an important aspect of licensing these reactors

6) Analysis of on-line monitoring systems and methods and advanced diagnostic methods needed to support high temperature gas reactors.

This work is a **MEDIUM** priority, because it will support other on-going research in this area both at the NRC and internationally. This work will need to be specialties to the specific issues of advance reactors but will follow on to the work that is currently planed else where.

7) Review of advance control algorithms for application to high temperature gas cooled reactors

This work is a **MEDIUM** priority, because significant has already been done in this area and NRC could use this research to develop a general approach to the review of these method. However almost all of the research to date on these methods has been on there efficiency and not on there failure modes additional research is needed.

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III.C.7 PRA FOR HIGH-TEMPERATURE GAS-COOLED REACTORS

(a) Description of the Issue(s)

Both industry and the NRC are utilizing PRA in the licensing process for advanced reactors. As envisioned by Exelon Corporation, risk information will play a crucial role in the licensing process for the proposed PBMR. Also given the NRC shift to risk-informed regulation, increased reliance on risk information is expected for licensing the GT-MHR. However, there is little PRA experience with these new reactor types for either industry or the NRC staff. This lack applies to limited scope PRAs such as Level 1 or 2, as well as full scope PRAs, including external events and diverse operational states. Therefore, the research summarized below is required for the staff to be able to independently review the PRAs included as the basis of future HTGR license applications. This research will provide the technical basis needed to assure that the submitted PRA is both accurate and complete.

(b) Risk Perspective

Maintaining safety, increasing public confidence, and making more effective decisions are three strategic performance goals in NRC's Strategic Plan (NUREG-1614). Whenever a new technology has been introduced into the nuclear reactor industry or a new phenomenology has been found, the NRC conducted an independent investigation of that technology or phenomenology. HTGRs are new designs and, therefore, the current LWR-related PRA experience is of limited use to HTGRs. The limitations of current PRA experience applies to the scope of the PRA (e.g., there is limited PRA experience for low power and shutdown); to the risk metrics used (e.g., core damage frequency or large early release does not seem to be applicable to these designs); and most importantly, to the design, materials, systems and safety approach applied. Future licensees have indicated that PRA and PRA insights will be an integral part of their license application. For the new HTGR designs, justifications regarding design safety, safety margins, and defense-in-depth will be based largely on results and insights from PRA evaluations. The staff needs to gain an understanding of the new designs, contributors to risk, and associated uncertainties. The tasks described in this research plan will enable the staff to achieve this objective and thereby contribute to meeting the agency's strategic performance goals discussed above.

(c) NRC Research Objectives and Plans

The objective of this plan is to identify and develop PRA methods, data, and tools needed to support an independent staff review of HTGR PRA submittals.

This plan is comprised of three tasks. The first task is to develop methods, data, and tools for modeling HTGR (PBMR or GTMHR) design and operational characteristics which are fundamentally different from those of a conventional LWR. The second task is to perform a PRA for the PBMR design (this is the first HTGR design likely to be submitted to the NRC, and no PRA has been done or reviewed by the staff). This will allow the staff to do a comparison of the models and results of its PRA with that of the licensee's submittal and, thus, gain an independent and more complete understanding of the safety issues associated with the proposed design. The third task will provide guidance to the staff based on the results of Tasks 1 and 2, for the review of HTGR PRAs.

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1. PRA Supporting Analyses. There are fundamental tasks that need to be addressed to support either performing an independent PRA or reviewing the submitted PRA. These tasks are described below.

1.1 Risk Metrics. The concepts of core damage frequency (CDF) and large early release frequency (LERF) do not appear to be directly applicable to HTGR designs. Therefore, the question of what should be the subsidiary figures of merit (i.e., analogs of CDF and LERF) for HTGRs, consistent with NRC top level safety goals, needs to be addressed. New risk figure(s) of merit are needed for risk evaluations as well as for developing regulatory criteria for design review and acceptance.

1.2 Initiating Event Identification and Quantification. The events that challenge a LWR are not applicable to the HTGR. It is critical for a PRA to correctly and comprehensively identify those events that have the potential to initiate an accident. Therefore, understanding what events can occur (as a result of design weaknesses, equipment failures, and human errors) that challenge the plant operation comprise the first step in assessing the risk associated with a given reactor design.

1.3 Accident Progression and Containment Performance (including radiological source term). The likely accident progression phenomena will have to be determined based on on-going research, previous experiments, experience in other industries, and expert judgment. A combined deterministic/probabilistic approach, with elicitation methods similar to those used for the liner melt through and direct containment heating issues in some LWRs, may be possible. The accident progression effects of distinct design difference need to be understood. For example, hydrogen should not be generated during the course of an accident, but the potential generation of other combustible gases needs to be examined. In addition, the loss of helium and the effects of air ingress on the accident progression will have to be considered (as well as water ingress, for any credible scenario in which this could occur).

An accident progression (level 2) analysis should assess the benefits of having a containment for the reactors, or of having only a confinement but with a filtered venting system to provide additional protection against release of fission products. (The confinement concept has been successfully modeled in past PRAs, although not for commercial reactor designs.) The benefits of having a containment could include those gained in terms of airplane crashes or other external attacks. Furthermore, we could study the benefit of complete underground siting instead of the partial underground siting now proposed for some HTGR designs. These analyses would directly feed into any safeguards and security work being done with respect to a license application.

Source term work will be performed as part of thermal-fluid dynamics and severe accident work of this RES plan. The knowledge of the fuel performance is a prerequisite to perform an independent review of the PRA. We need to understand how the core behaves during accidents such as overheating or immersion in media other than helium (in air or, if possible, in water). This behavior should be understood not only for fresh fuel but also for fuel near the end of its life to see if burn-up matters.

1.4 System Modeling. Passive systems are used more extensively in HTGRs than LWRs. In addition, the advanced designs will incorporate digital systems. Both need examination.

Passive systems have been treated in PRAs as either initiators (e.g., LOCAs) or complete failures. As a result, current PRAs model only the performance of active systems using a binary logic which is suitable for such purposes. It is not clear that this approach would be suitable for modeling passive systems exhibiting slow evolutionary behavior during accidents. Therefore, the failure modes of such systems need examination including whether and how existing modeling approaches can be amended.

Digital systems typically have not been considered in the past PRAs. In advanced reactors, however, digital instrumentation and control (I&C) systems will be the norm. The reliability of digital systems is being addressed in another part of this RES plan (see Section III. C.6). PRA modeling issues concerning digital system performance should be addressed here. Methods should be developed for incorporating digital system failure in the PRA logic.

The uncertainties associated with the development of modeling the failures of passive and digital system also needs to be addressed.

1.5 Data Collection and Analyses. The HTGRs employ different systems and components, and operate at significantly different conditions than LWRs, hence, the LWR data will not be applicable. The use of appropriate data is crucial in the assessment of the risk associated with a given reactor type. Therefore, collecting and analyzing data applicable to HTGRs is essential. The Fort St. Vrain experience and some of the German AVR data may be applicable.

Furthermore, this task includes addressing the uncertainties associated with the data. Understanding the uncertainties is a very important aspect for any PRA; it is much more crucial for these types or reactors given the limited or lack of operating experience and the significant use of the PRA in the HTGR licensing process.

1.6 Human Reliability Analysis. The operator role in the new reactors (to be built on the premise that they will be human-error free and that, if an event occurs, human intervention will not be necessary for at least 30 minutes) is not well understood. Issues related to the needs for reliable performance (e.g., staffing and training) are part of a different activity of the RES plan (see Section III.B). New human reliability methods, such as ATHEANA, were developed to assess the impact of human performance on plant safety when dealing with long-term and slowly evolving accidents, such as those expected to be predominant in accident sequences related to graphite-moderated reactors. Under this subtask these related efforts will be coordinated to determine if (and what) modifications are warranted to appropriately incorporate the impact of human performance in an HTGR PRA.

1.7 Other Events (internal flood, fire and seismic). As with any design that uses digital I&C, failure possibilities of electronics will need to be addressed. Specifically, the response of digital electronics in a fire or flood is expected to be quite different than that of electromechanical components. The differences may not be just in probability but also in the kinds of failures that could potentially occur (e.g., can an active failure occur?). Furthermore, in light of the new design, all of the external events must be applied to the HTGR from a scoping perspective to see if anything unique might occur.

1. 8 Quantification The SAPHIRE code would be used in the performance of an independent PRA or in examining the applicant's PRA results. The code could use modifications, some of which are of particular interest when contemplating a full scope PRA (external and internal events, full and low power). Such a PRA will generate many more "cut sets" than are comfortably handled now. In addition, the rationale developed for other designs for pruning the results may not be appropriate for an HTGR design in which low frequency sequences may dominate the risk. Other computational capabilities need to be addressed to see if methodological steps in them need altering for a different design with an expected different risk profile. These capabilities include current radiological source term and consequence analysis tools, and the development and use of new modeling techniques, for example, dynamic modeling.

1.9 Other Operational States. The design and intended operation of the HTGR must be examined so that any unique characteristics of them during less than full power operation can be correctly accounted for in the PRA review.

1.10 Multiple Modules. Current PRAs are usually performed for a single unit or sometimes for two sister units. The plan for the PBMR is to operate up to 10 modular units together at a facility with a centralized control room. The PRA in the license application will have to address potential interactions among the multiple modules, and NRC will need to be prepared to review these issues. Examples are: (1) what are the common systems? power, helium storage, cooling water, etc.; (2) what are the cross ties, if any, between units? (3) what is the effect of limited operator staffing and shared systems when accidents involve more than one module (as could be the case for common cause initiators, such as seismic and external flood events)? In addition, human reliability models need to be examined to understand if errors of commission or omission are more likely when multiple modules share the same control room.

1.11 Safeguards and Security. As mentioned above in Section 1.3, there are some portions of this work where explicit information can be generated regarding the safeguards and security for the design. We need to explore how this can be accomplished in the most efficient manner and what other areas of the PRA studies can assist in this endeavor.

2. PRA. Performing an independent full-scope level 3 PRA is proposed in this task for the PBMR design. Performing a PRA will provide the staff more confidence in evaluating and reviewing such a design. Subtasks to be done are the same as a PRA for an LWR: hazard identification; initiating event identification; accident progression analysis including success criteria, accident phenomena, and accident sequence delineation; systems analysis; data analysis; human reliability analysis; other events such as internal flood, fire and seismic; other operational states; quantification and results analysis including consequence analysis and uncertainties.

3. PRA Review Guidance. Under this task guidance will be developed explaining how the results of Tasks 1 and 2 can be used by the NRC staff for its review of an PBMR or HTGMR PRA submittal.

(d) Related NRC Research

The study of the performance of the unique fuels used in the HTGRs and the source term work are discussed in other parts of this RES plan. The knowledge of the fuel performance is a prerequisite to perform an independent review of the PRA. Knowledge of the proposed thermal hydraulics and severe accident work is required for this work. In addition, ongoing work under the programs ATHEANA (A Technique for Human Event Analysis) and the SAPHIRE PRA code will support this effort.

(e) Related International Cooperation

The possibility for cooperating with the South Africa Nuclear Regulatory Commission will be explored as part of this plan.

(f) Resources and Schedule

The estimated resources and schedule to performs the necessary work in preparation to review the licensee's PRA for this five-year research plan are given the following pages. The schedule indicates that an initial supporting analysis (Task 1) will be completed in about 6 months from the initiation of the work, and a scoping PRA for the PBMR (Task 2) will be completed within a year. The steps of performing supporting analysis and more detailed PRA are iterative. That is, as additional supporting analysis tasks are completed, the results will be fed into revising the PRA. Initial review guidance will be prepared at about 2 ¼ years so the staff will be ready to review the PRA which will be submitted along with the construction and operating license (COL) submitted in early 2004. The draft PBMR PRA will be completed at about 4 years after the initiation of the work. This will provide a basis for performing comparisons with the PBMR PRA (submitted as part the license application). Final review guidance will be prepared at about 5 ½ years to support the staff in the design certification process anticipated to be submitted in 2006.

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Task	Resources	
	Staff-months	Dollars*
1. PRA Supporting Analyses (PBMR and HTGMR)		
1.1 Risk metrics	6	$150
1.2 Initiating event identification and quantification	12	$300
1.3 Accident progression and containment performance (including source terms)	see other research	
1.4 System modeling including uncertainties	9	$225
1.5 Data collection and analyses including uncertainties	24	$600
1.6 Human reliability analysis	6	$150
1.7 Other events (internal flood, fire and seismic)	6	$150
1.8 Quantification	6	$150
1.9 Other operational states	6	&150
1.10 Multiple modules	6	$150
1.11 Safeguards and security	6	$150
Subtotal	93	$2325
2. PRA (PBMR)		
• Risk metrics/Hazard identification	see (1.1) above	
• Initiating event identification and quantification	see (1.2) above	
• Accident progression (success criteria, accident phenomena, accident sequence delineation)	30	$750
• Systems analysis	36	$900
• Data analysis	see (1.4) above	
• Human reliability analysis	9 (also see other research)	
• Other events (internal flood, fire, seismic)	18	$450
• Quantification and results analysis (including consequence analysis and uncertainties)	24	$600
• Other operational states	see (1.9 above)	
Subtotal	117	$2925
Project Management for Tasks 1 and 2	45	$1,125
3. PRA Review Guidance	6	150
TOTAL	**255**	**$6,375**

*Dollars are estimated assuming 1.0 staff-month of an FTE costs $25K.

Schedule

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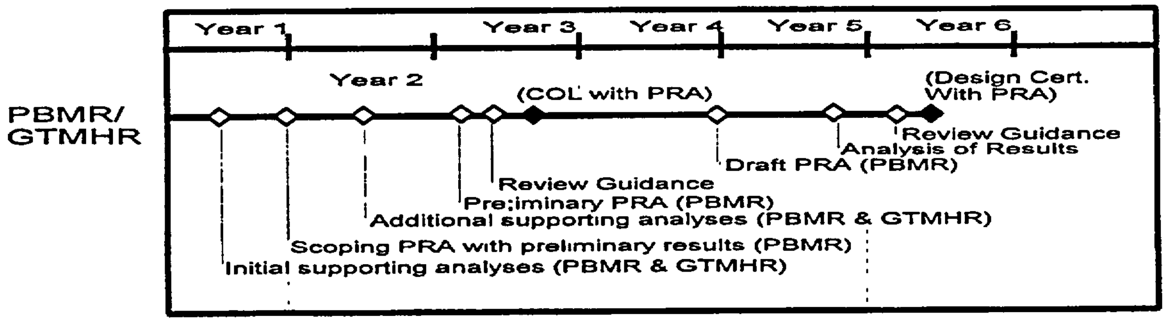

(g) Priorities

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III.D.1. THERMAL HYDRAULICS

Thermal-hydraulics of advanced light water reactors (ALWRs) is relatively well understood because of the experimental and analytical efforts made to investigate the performance of conventional light water reactor systems. Advanced reactors however still pose significant challenges to engineering analysis due to several unique design features. Understanding the effects of these features on local and system-wide thermal-hydraulics is necessary in order to confirm and quantify the expected safety margin of the proposed LWR. This section discusses those features and the thermal-hydraulic issues for advanced light water reactors.

Specifically, two ALWR systems are discussed; the Westinghouse AP1000, and IRIS (International Reactor Innovative and Secure). Both designs rely on passive safety systems to insure adequate core cooling and prevent core uncovery. Preliminary assessments show that for each of these designs, the passive systems adequately remove decay heat following a pipe rupture. Confirmation of this apparent safety margin depends on assessing the performance of these passive systems, and quantifying uncertainties associated with the thermal-hydraulic processes which they utilize.

a. Description of Issues

The AP1000 relies on passive safety systems for decay heat removal. Pipe breaks throughout the primary system must be considered as part of the design basis, as they are in conventional PWRs. The most critical accident scenarios in AP1000 have been defined as part of the AP600 Design Certification work. Thus, the major thermal-hydraulic issues for AP1000 are due to those thermal-hydraulic processes that are strongly dependent on the higher core steam production rate expected during an accident.

The major thermal-hydraulic issues for the AP1000 include:

(i) <u>Entrainment from horizontal stratified flow</u>: Higher core steam production increases steam velocities in the hot leg and automatic depressurization system (ADS) during later phases of a small break loss of coolant accident. Sufficiently high steam velocities can entrain water from the hot leg and carry droplets into the ADS. This increases the pressure drop between the core and containment, and delays injection from the IRWST. New experimental data and models to predict this process are necessary.

(ii) <u>Upper plenum pool entrainment and de-entrainment</u>: High core steam production may entrain a significant amount of water from the pool in the upper plenum during a small break LOCA. This may result in core uncovery for accident scenarios where the two-phase level drops below the bottom of the hot legs. Experimental data for prototypical upper plenum geometry is needed, and analytical models to account for entrainment and de-entrainment in the upper plenum are needed.

(iii) <u>Low pressure critical flow</u>: Transition from high pressure phases of a small break accident to the IRWST injection period occurs while steam is vented through the ADS. Because of the rapid depressurization, the flow remains critical with an upstream pressure that is much lower than pressures maintained in previous experiments used to examine critical flow.

(iv) Direct vessel injection: Flows from the core makeup tank (CMT) and IRWST are injected directly into the down-comer in the AP1000. This design feature is intended to reduce ECC bypass during a large break loss-of-coolant accident (LOCA). Validation of models to predict bypass flows is made difficult because of the lack of experimental data for this injection geometry. Satisfactory resolution of ECC bypass for direct vessel injection may require new experimental data, and additional code validation.

(v) Passive containment cooling: The AP1000 containment is cooled by internal natural circulation and heat transfer to a thin film on the exterior. Experimental data and reliable analytical models for estimating AP1000 containment response are needed.

The IRIS design is a modular light water reactor with a power of up to 335 MWe. It makes use of passive safety systems to insure adequate core cooling, but because of the system design, the possibility for many of the conventional design basis accidents is eliminated. The steam generator, pressurizer, and coolant pumps are all internal to the reactor pressure vessel (RPV), which is contained within a relatively small containment shell. A loss of coolant from the RPV is expected to cause a rapid increase in containment pressure, which will subsequently reduce the rate of vessel inventory loss.

Because of the unique vessel design and intimate coupling between the vessel and a small containment, risk significant accident scenarios are not well defined. Few evaluations have been performed to identify the worst break location and failure conditions or to explore system response to a wide range of accident conditions.

The major thermal-hydraulic issues for IRIS include:

(i) Two-phase flow and heat transfer in helical tubes: The in-vessel steam generators for IRIS are of a modular helical coil design. The coils are located in the annular space between the core barrel and the vessel wall. Each of coil has an outer diameter of approximately 1.6 m. During loss of coolant accidents, heat transfer by the steam generators are an important mode of heat removal. Flow conditions may vary significantly on the outside of the tubes, as the conditions change from forced flow to natural circulation during an accident. Prototypical experimental data will be needed to determine internal, external and overall heat transfer coefficients for accident conditions. These data will be necessary to develop analytical models for computer codes to predict system response.

(ii) Two-phase natural circulation: The IRIS design operates with a high level of natural circulation, with more than 40% of the total core flow caused by natural convection. During a LOCA, natural circulation through the core and within the vessel will be responsible for decay heat removal. Experimental data is needed to benchmark and verify computer codes to predict IRIS behavior during accident conditions.

(iii) Containment - RCS interaction: A major difference between IRIS and conventional PWRs is the strong coupling between its small, passively cooled containment, and the primary system. Rapid pressurization and flooding of the containment are important processes in mitigation of a LOCA. The rapid change in pressure differential across the break will pose unique problems to code development. New experimental data for critical break flow, and to evaluate system response due to rapidly changing containment back-pressure will be needed. Modeling the vessel - containment interaction will require a coupling between thermal-hydraulic codes for system response and containment response.

(iv) Parallel channel flow instabilities: Because the IRIS has an open lattice core, the core is essentially composed of many parallel channels with boiling taking place in the upper part of the core. As such, the system may be prone to two-phase flow instabilities. An experimental investigation of conditions that might lead to instabilities in IRIS is warrented.

(v) Pump performance: The IRIS coolant pumps are "spool type" pumps, which have not previously been used for PWR primary flows. Performance of this type of pump for two-phase conditions needs to be established.

b. Risk Perspective

Based on preliminary evaluations, the AP1000 design appears to be robust. No core uncovery is expected for design basis accidents. Even if uncovery were to occur, the expected clad heat up and peak cladding temperature would remain well below 10 CFR 50.46 limits. The AP600 design did not exhibit core uncovery even for many beyond design basis scenarios.

Because it is a relatively new design, the performance of the IRIS to design basis accident scenarios has not been extensively investigated. Beyond design basis accidents have not been considered in great detail. Thus, the risk perspective of IRIS has not been established.

c. Related NRC Research

The NRC has maintained an active, confirmatory thermal-hydraulics research program to investigate poorly understood phenomena that are important to advanced passive plants such as the AP1000. Central to this effort has been the experimental program conducted at Oregon State University using the APEX facility. APEX is a scaled integral effects facility, which has been used to simulate a wide range of accident scenarios applicable to the AP1000. The facility is currently being upgraded to operate at higher power levels.

The NRC has also maintained an active experimental program using the PUMA facility. This facility is a scaled representation of an SBWR and has most recently been used to obtain experimental data for low pressure critical flow.

Separate effects test facilities have been established at Penn State University to investigate rod bundle heat transfer, and at Oregon Sate University to investigate entrainment from the hot leg to branch lines. Both of these facilities are expected yield experimental data important in predicting advanced plant behavior.

In addition to the experimental programs, the NRC is actively developing the TRAC-M thermal-hydraulics code for application to advanced passive plants. This code is applicable to the AP1000, and has nearly all of the features necessary to model and simulate IRIS.

d. Related International Cooperation

Currently, the NRC does not have an International Cooperative agreement to perform research or evaluations of the AP1000. Likewise, there are no International efforts in conjunction with the NRC to investigate IRIS. In should be noted however, that there is significant international interest in IRIS. The consortium led by Westinghouse to design IRIS includes several active members from Europe and Japan. It should be feasible for the NRC to reach an international agreement with interested stakeholders to perform confirmatory studies of IRIS.

e. NRC Research Objectives and Plans

The NRC research objectives for AP1000 and IRIS are to perform the experimentation and code development necessary to confirm compliance with 10 CFR50.46 and to determine if there are conditions or accident scenarios that have unacceptable risk. For the AP1000, an integral effects test facility exists, and separate effects tests are being conducted to develop data for models of critical importance. To fulfil these objectives for the AP1000, a series of confirmatory tests, run under both design basis and beyond design basis accident conditions should be conducted in the APEX facility. These tests should be run at a power scaled to the AP1000, and should be used as part of code development and validation for TRAC-M.

To meet these objectives for IRIS, a comprehensive confirmatory test and analysis program should be conducted. Improved models for two-phase flow and heat transfer in helical coils need to be developed and implemented in the TRAC-M code, and the capability to predict the overall system performance must be demonstrated. To simulate transients with strong vessel - containment interaction, it will be necessary to couple TRAC-M to a containment code such as CONTAIN. Models in the CONTAIN code for passive cooling, condensation, film coverage and non-condensable distribution would need to be assessed and improved.

f. Resources and Schedule

The following Table summarizes resource and schedule requirements for advanced light water reactor research:

Task	Comment	Start Date	End Date	Cost Estimate
APEX-AP1000 Confirmatory Integral Testing	Perform series of integral effects tests to confirm plant resonse to design basis and beyond desing basis events.	11/02	6/04	$650,000

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AP1000 Model Development and Separate Effects Testing	Perform separate effects tests in ATLATS and PUMA facilities. Develop code models for AP1000 phenomena.	6/02	12/03	$250,000
AP1000 code development and assessment.	Develop and assess TRAC-M using APEX-AP1000 confirmatory data. Perform large and small break calculations of AP1000 with assessed code version.	6/02	12/04	$350,000
IRIS code development and preliminary assessment.	Couple TRAC-M and containment code. Develop preliminary models for IRIS unique features. Perform preliminary calculations of IRIS using TRAC-M.	1/04	6/05	$450,000
IRIS Helical SG thermal-hydraulics	Construct separate effects facility to investigate two-phase flow and heat transfer in large scale helical coils. Obtain experimental data suitable for code and model development.	1/05	12/07	$3,000,000*
IRIS Integral Testing	Construct a scaled, multipurpose integral effects facility to investigate IRIS vessel and containment response during accidents. Obtain experimental data suitable for code assessment and to determine plant response to design basis and beyond design basis events.	6/05	12/09	$50,000,000*
IRIS code and model development	Develop improved models for helical steam generator thermal-hydraulics, and IRIS internal circulation, and vessel-containment interaction. Perform independent assessment of IRIS performance during accidents using the assessed version of TRAC-M.	1/06	6/10	$2,500,000

* If necessary, would pursue cooperative efforts to share cost.

g. Priorities

The highest priority is for AP1000 confirmatory testing and analysis due to the near term possibility of a submittal for design certification. The IRIS related efforts have a lower priority because of uncertainty in the design and licensing schedule.

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III.D.1. THERMAL HYDRAULICS

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To meet these objectives for IRIS, a comprehensive confirmatory test and analysis program should be conducted. Improved models for two-phase flow and heat transfer in helical coils need to be developed and implemented in the TRAC-M code, and the capability to predict the overall system performance must be demonstrated. To simulate transients with strong vessel - containment interaction, it will be necessary to couple TRAC-M to a containment code such as CONTAIN. Models in the CONTAIN code for passive cooling, condensation, film coverage and non-condensable distribution would need to be assessed and improved.

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* If necessary, would pursue cooperative efforts to share cost.

g. Priorities

The highest priority is for AP1000 confirmatory testing and analysis due to the near term possibility of a submittal for design certification. The IRIS related efforts have a lower priority because of uncertainty in the design and licensing schedule.

III.D.2 SEVERE ACCIDENT CODES, INCLUDING RADIOLOGICAL SOURCE TERM 12/27/01

(a) Description of Issues

The phenomena of concern in analyzing severe accident scenarios in any nuclear reactor are those phenomena involved in accidents that are beyond design basis in which significant core degradation occurs. A beyond design basis accident is one that causes the release of fission products beyond an acceptable limit. Since fission products are bound in the reactor fuel any event or action that can cause fuel damage such as chemical, heat, or mechanical damage to the fuel can be considered an important issue for severe accident analysis. Release of the radioactive fission products to the environment (the composition and quantity of which is called the 'source term') may be impeded by particular mitigative features of a plant, such as coolant system piping and other equipment, containment filters, sprays, and other aspects that provide fission product retention. Therefore, aspects of the reactor design that impede or accelerate the release of fission products to the environment also need to be considered when attempting to predict source terms.

Since severe accidents are normally considered high consequence, low probability events, specific sequences leading to a severe accident are usually only of interest to the degree to which they determine the plant state during the postulated severe accident. Accounting for situations where the plant state is in one of these low probability states may require knowledge of phenomena outside of the normal design range of conditions. Therefore, predicting the plant state during a postulated severe accident may or may not require the use of a code that is normally used for design basis analyses. Thus, any computer codes used specifically for analyzing severe accident phenomena would focus mainly on such phenomena and leave the thermal-hydraulic and kinetics phenomena to codes specifically designed for those purposes, except when analyzing situations outside of the valid ranges for those codes.

Currently the MELCOR severe accident code is used to evaluate core damage and source term release accident scenarios for light water reactors. MELCOR has already been used to analyze the AP600 plant design and extension to the AP1000 design is not expected to require the consideration of any new severe accident phenomena. This is not to say that the AP1000 design does not need to be analyzed, but that the MELCOR code should already have the appropriate models necessary to analyze this design. Regarding the IRIS design, we are not currently aware of any phenomenology that is unique to this design that will require changes to the MELCOR code.

Computer codes used to evaluate severe accident phenomena in light water reactors generally focus on core degradation, relocation, and fission product transport within the reactor coolant system. Source term codes typically follow the fission product transport from the reactor coolant system to the environment. Specific aspects analyzed by our primary light water reactor severe accident code, MELCOR, include:

1) Core uncovery (loss of coolant), fuel heat-up, cladding oxidation, fuel degradation (loss of rod geometry).

2) Core material melting and relocation, heat-up of reactor vessel lower head from relocated fuel materials and the thermal and mechanical loading and failure of the vessel lower head, and transfer of core materials to the reactor vessel cavity,

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3) Core-concrete attack and ensuing aerosol generation.

4) In-vessel and ex-vessel hydrogen production, transport, and combustion.

5) Fission product release (aerosol and vapor), transport, and deposition behavior of radioactive aerosols in the reactor containment building, including scrubbing in water pools, and aerosol mechanics in the containment atmosphere such as particle agglomeration and gravitational settling.

6) Impact of engineered safety features on thermal-hydraulic and radio-nuclide behavior.

For the AP1000, one immediate issue is in-vessel retention for the high power case. One needs to demonstrate that in-vessel retention can be achieved for AP1000 in a manner similar to that for AP600.

(b) Risk Perspective

MELCOR was developed to be used to perform level 2 PRA analysis. As a level 2 PRA code, MELCOR must be able to model any phenomena considered important in analyzing a accident sequence postulated by the level 1 PRA studies. The results of the MELCOR analysis are meant to be feed into the MACCS code for level 3 PRA analysis. MELCOR is therefore already an integral part of the NRC's ongoing risk informed regulation paradigm.

(c) Related NRC Research

Results of related NRC research regarding these ALWR designs will need to be reviewed as it becomes available to determine if phenomenological models need to be modified or added to the MELCOR code.

(d) Related International Cooperation

The Cooperative Severe Accident Research Program (CSARP), sponsored by the IAEA, provides a general cooperative framework for sharing severe accident information with international organizations. Any information that is relevant to these ALWR designs being made available through the CSARP should be reviewed for potential application to the MELCOR code.

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(e) NRC Research Objectives and Plans

Currently, to make best use of resources, RES has focused on consolidating and modernizing our severe accident codes in to one code, MELCOR. Based on the information currently available we can identify no phenomenological aspects of the new designs that will need to be addressed by modifications to MELCOR. Aspects, perhaps those dealing with unusual flow properties or heat transfer issues, will need to be addressed as more is learned from new and existing system level experiments and may be from operational experiences.

(f) Resources and Schedule

Validate Bottom Head Heat Flux Models

An issue of concern in the AP600 review was the ability of an external pool of water to keep the bottom head of the AP600 vessel cool and intact in the event that core damage should cause a debris bed to form inside of the vessel. For the AP600 the critical heat flux to the pool was considered likely to be sufficient to provide for the necessary cooling. However, the AP100 core is of considerably higher power density and may cause some concern with regard to the ability of the water pool to carry away enough heat to keep the bottom head of the vessel from failing. This is a severe accident code issue only from the view point of whether or not the heat flux models in MELCOR are sufficiently accurate to do the calculations. Verification of the code models will need to be performed based on previous and perhaps new experimental data. The schedule and resources listed below assume the considerable work involved in reviewing and implementing code models for a new experiment. If the current experimental data are considered applicable then the resources necessary will of course be much lower and the completion date will be much sooner.

Estimated completion date:	January 2003
Level of Effort:	5 staff-months (FY02)
	2 staff-months (FY03)

(g) Priorities

TBD. At this point, only verification of the bottom head heat flux models is proposed. This verification may simply involve review of previous experiment data, or it may require new experiments to be performed. This work should be at least of equal priority to the HTGR modification work for MELCOR as the AP1000 design certification review is expected to begin in early FY2002.

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III.D.3 FUEL PERFORMANCE AND QUALIFICATION

(a) Description of Issue

To be able to effectively review the new ALWR core designs, there is a need to conduct confirmatory research to establish an information base related to the long-term performance and behavior of the fuel assemblies and control rods under temperatures and flux levels expected under normal operating and accident conditions in the IRIS design proposed by Westinghouse and its partners. Furthermore, various performance parameters which would be affected by temperature, radiation (e.g., burn-up, maximum fluence), and oxidation in the event of transients or accidents, need to be examined. NRC should consider conducting studies to obtain confirmatory data to assess behavior at burnups greater than 90 MWd/t.

The first IRIS core will employ standard <5% UO_2 fuel and standard PWR fuel assembly design. This represents current, proven and licensed fuel technology, therefore no licensing issues related to fuel are foreseen by Westinghouse and its partners. A path forward for future fuel cycle enhancement (extending the core life to 8-10 years by increasing the fissile content to about 8% enrichment) has been envisioned, but it will not be part of the initial IRIS design for licensing.

There are several outstanding questions and issues, that should be addressed by the research:

1. What new analytical capability is needed?

2. What confirmatory data would NRC need to develop to have a reasonable confidence for reviewing the acceptability of an applicant's fuel assembly qualification program?

3. What acceptance criteria should be in place for fuel assembly (FA) design, manufacturing, testing, surveillance, inspection and in-service examinations?

4. What international collaborative efforts should NRC participate in for optimum benefit and to leverage cost?

5. Does the long length of the control rod drive lines need investigation? Is a control rod ejection accident more (or less) pronounced than in current PWRs? One should note that Westinghouse is considering in-vessel control rod drives, which would make these concerns moot.

6. Issues regarding greater than 5% U235 enrichment, core neutronics and LHGR peaking factors during the four-year cycle, and boron concentration during the cycle, are covered under the section on neutronics.

(b) Risk Perspective

The IRIS reactor "safety by design" approach attempts to first eliminate the possibility of accident sequences from occurring, and second, to reduce the severity of consequences and/or the probability of occurrence. The integral reactor vessel (RV) configuration, is an ideal layout for implementing this approach. Because the integral reactor vessel contains the steam generators, reactor coolant pumps and the pressurizer, there is no external large loop piping, and therefore there is no possibility of a large LOCA. In addition, the IRIS integral RV configuration results in a tall vessel with elevated steam generators and a low pressure drop flow path, which provides increased natural circulation capability and intrinsic mitigation of Loss of Flow Accidents (LOFAs). The integral RV also provides a large inventory of water above the reactor core, which slows the reactor response to transients and postulated small LOCAs.

But we do need to see if the MSLB or ATWS is more severe (in terms of return to power) than in current PWRs. And even though there is less fuel handling, we should check the radioisotope inventory for the fuel handling accident.

(c) Related NRC Research

Work is expected to start at ANL next FY on Zirconium-1% Niobium alloys, such as Zirlo.

Work on FRAPCON and FRAPTRAN is being carried out at PNNL, but additional effort would be needed to extend code assessment to burnups greater than 75 GWd/t.

(d) Related International Research

HALDEN

The OECD Halden reactor program is performing and will continue to perform research on behavior of high burnup fuel, including fission gas release. Also research on absorber materials and cladding corrosion is being studied.

Cooperation with Other Countries

To be developed.

(e) NRC Research Objectives and Plans

NRC research should be directed towards developing the technical basis to enable the NRC to effectively review various fuel assembly issues. Future research should answer some of the most fundamental questions: What are acceptable performance criteria? What standards should be applied to fabrication and structural design of fuel assemblies? What are the changes in physical properties of the cladding (including oxidation, thermal properties, and mechanical properties) as a function of temperature and irradiation? What in-service examinations, inspections and surveillance should be performed on fuel rods and how should these be done?

To be able to achieve these objectives, research related to the following categories needs to be considered:

- Projected cladding temperatures at full power (for maximum power rod and average power rod) are higher than in current PWRs. For example, the average assembly outlet temperature is 626 F.
- Erbium in UO2 needs to be studied.
- Detection of excessive corrosion should be considered. when core life is longer than four years.

END PRODUCTS

IRIS version of FRAPCON to calculate burnable absorber helium release, very high burn-up fission gas release and fuel conductivity degradation.

(f) Resources and Schedule

1. **Physical Characteristics of Very High Burn-up Fuel:**

   Characterization of the key properties such as fuel thermal conductivity.

   Estimated cost: Part of Halden program
   Period of performance: 24 months.

2. **Cladding Qualification Program:**

   Conduct a review of available high temperature corrosion data.

   Estimated cost: $50K
   Period of performance: 4 months.

(g) Priorities:

Confirmatory research related to very high burn-up fuel is a medium priority item.

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III.D.4 - NUCLEAR ANALYSIS -- ALWRs

(a) Description of Issues

The term "nuclear analysis" describes all analyses that address the interactions of nuclear radiation with matter. Nuclear analysis thus encompasses the analysis of: (a) fission reactor neutronics, both static and dynamic, (b) nuclide generation and depletion as applied to reactor neutronics and to the prediction of decay heat generation, fixed radiation sources, and radionuclide inventories potentially available for release, (c) radiation transport and attenuation as applied to the evaluation of material damage fluence, material dosimetry, material activation, and radiation protection, and (d) nuclear criticality safety, i.e., the prevention and mitigation of critical fission chain reactions ($k_{eff} \geq 1$) outside reactors.

This draft of the ALWR nuclear analysis research plan addresses only the nuclear analysis issues associated with reactor neutronics and in-reactor decay heat generation. Future drafts of the plan will address additional nuclear analysis areas as they pertain to anticipated issues arising in the evaluation of ALWR reactor safety and material safety. The following subsections begin with a brief discussion of the nuclear data libraries that are fundamental to all areas of nuclear analysis and continue with discussions of ALWR reactor neutronics and decay heat analysis issues.

Nuclear Data Libraries

All areas of nuclear analysis make use of nuclear data libraries derived from files of evaluated nuclear physics data, such as ENDF/B in the U.S., JEF in Europe, or JENDL in Japan. The nuclear data files include, for example, fundamental data on radionuclide decay as well as neutron reaction cross sections, emitted secondary neutrons and gamma rays, and fission product nuclide yields, all evaluated as complex functions of incident neutron energy. The neutron reaction evaluations also provide cross-section uncertainty information in the form of covariance data that can now be processed and used with advanced sensitivity and uncertainty analysis techniques, as developed in recent years under RES sponsorship, to assist in the identification and application of appropriate experimental benchmarks for problem-specific code validation.

Many of the processed nuclear data libraries in use today were developed in the 1980s or earlier. For example, pre-1990s cross section libraries are being used for preparing the nodal data used by the NRC's reactor spatial kinetics code, PARCS, and for the criticality, depletion, and shielding analysis sequences in the NRC's SCALE code system. While these legacy cross section libraries have proven largely adequate in a variety of applications, they have known limitations and shortcomings and cannot be described as state-of-the-art.

In response to a 1996 user need memorandum, RES has sponsored ORNL to upgrade the AMPX code suite to enable its eventual use in creating new cross section libraries that would take full advantage of the expanded resolved resonance ranges and the improved/corrected nuclear data and covariance evaluations now available in the latest releases of ENDF/B-VI and its foreign counterparts JEF2.3 and JENDL-3. With the recently completed AMPX upgrades as well as continued improvements to the NJOY nuclear data processing codes, both opportunity and motivation now exist to produce and test state-of-the-art nuclear data libraries for use in the analysis of reactor safety and material safety issues associated with conventional and advanced reactor technologies.

Reactor Neutronics and Decay Heat Generation

The nuclear heat sources of importance in all reactor safety analyses are those arising from nuclear fission and the decay of radionuclides produced by nuclear fission and neutron activation. Reactor neutronics codes are used to predict fuel burnup and the dynamic behavior of neutron-induced fission chain reactions in response to reactor control actions and system events. Under subcritical reactor conditions, where the self-sustaining fission chain reactions have been terminated by passive or active means, the decay of radioactive fission fragments and activation products becomes the dominant nuclear heat source. Note that the 1986 Chernobyl accident involved a rapid power excursion from a prompt supercritical fission chain reaction, whereas the 1979 Three Mile Island accident involved inadequate removal of decay heat from a subcritical reactor.

Reactor neutronics and decay heat analysis issues for AP1000 are essentially identical to those for AP600 and the current generation of PWRs, with, for example, their gradual evolution to the higher initial enrichments and new burnable poison designs needed for higher burnups and longer cycles. The only new research activities anticipated for AP1000 neutronics would be the limited RES assistance that may be needed in preparing, testing, and running a PARCS input model for the spatial kinetics analysis of AP1000 reactivity transients.

Neutronics and decay heat analysis issues specific to the IRIS design include the following:

(i) Fuel depletion modeling: Depletion analysis of the IRIS fuel designs, with their >5% initial enrichments, significantly higher moderator-to-fuel ratios, novel burnable poison designs, and higher design burnup levels, may call for flux-solver methods and modeling practices more advanced than those traditionally used in analyzing conventional PWR fuels. Modeling studies with higher order methods (e.g., Monte Carlo) will be needed to assess such depletion modeling issues and develop appropriate technical guidance.

(ii) Fuel depletion validation: The available experimental data base for validating LWR fuel depletion analysis methods consists largely of destructive radiochemical assays performed in the 1970s and 80s on rod segments from a dozen or so discharged PWR and BWR fuel assemblies. The database includes essentially no data from fuel rods with integral burnable poisons, initial enrichments above 4%, or burnups beyond 40 GWd/t. Sensitivity analyses, based on methods developed in recent years under RES sponsorship, will be needed to help assess the applicability of the existing validation databases to the IRIS fuel designs (with their >5% enrichment, significantly higher moderator-to-fuel ratios, advanced burnable poison designs, and burnup levels to 80 GWd/t) and to assist in the prioritization of further data needs and the estimation of remaining validation uncertainties.

(iii) Neutronics of high-burnup cores: The IRIS concept of a 5- to 8-year straight-burn core without fuel shuffling poses a number of issues concerning the neutronics analysis of its initially highly poisoned and subsequently highly burned core. Current LWR experience makes relatively modest use of burnable poisons and is limited to shuffled core-average burnup values less than 35 GWd/t, whereby fresher fuel assemblies are typically placed in close proximity to those approaching design burnups of 60 GWd/t or less. Cumulative uncertainties associated with poison and fuel burnup effects, even at moderate burnups, will have greater neutronic significance in IRIS than in shuffled PWR cores. Neutronic phenomena affected by such analysis uncertainties would include temperature coefficients, spatial power profiles, control worths, shutdown margins, and kinetic parameters like effective delayed neutron fraction and prompt neutron lifetime.

(iv) Decay heat power: Due to depletion modeling issues and the apparent shortage of available radioisotopic or calorimetric validation data applicable to the IRIS fuel designs at high burnup (see related items i and ii above), specific technical guidance will likely be needed on accepted methods for computing decay heat sources with appropriate consideration of validation uncertainties.

(b) Risk Perspective

The results from accident sequence analyses provide information that may be used in plant PRAs for assessing event consequences and their probabilities. Core neutronics codes, generally coupled with thermal-hydraulic and/or severe-accident systems codes, are needed for evaluating the dynamic progression of accident sequences that involve reactivity transients. For accident sequences in which the self-sustaining fission chain reaction is terminated by active or passive means, the codes used in evaluating the thermal response of the subcritical system (e.g., maximum fuel temperatures) must employ algorithms that adequately represent the intensity, spatial distribution, and time evolution of the decay heat sources.

(c) Related NRC Research

Related past, ongoing, and planned NRC research efforts include the following:

(i) Recently completed RES-sponsored work on (1) upgrading the AMPX code system for use in creating state-of-the-art nuclear data libraries, (2) the development of sensitivity and uncertainty analysis methods that utilize cross section covariance data, (3) modeling and validation guidance for computing radionuclide inventories in high-burnup LWR fuels, and (4) guidance on modeling and validation uncertainties in computing the reactivity of spent PWR fuel.

(ii) Ongoing RES tasks at ORNL to complete the development of 2D-depletion lattice physics analysis sequences (NEWT/ORIGEN-S) in the NRC's SCALE code system for use in exploratory studies and preparing design-specific nodal physics data tables for input to the NRC's PARCS spatial kinetics code.

(iii) Future NRC research on the ALWR technical areas described in other chapters of this research plan (e.g., Thermal Hydraulics, Severe Accidents).

(d) Related Interoffice, Domestic, and International Cooperation
[In addition to international cooperation, we should also mention potential cooperation with other NRC offices as well as domestic cooperation with U.S. entities like DOE, universities, or industry/professional groups like EPRI, ANS, and ASME.]

Potential areas of related interoffice, domestic, and international cooperation include the following:

(i) To acquire relevant insights from recent and ongoing efforts to assess biases and uncertainties in computing the isotopic composition and reactivity of moderate- and high-burnup PWR fuels, RES staff could seek interoffice cooperation with staff in NMSS/DWM and NMSS/SFPO, as well as cooperation with the DOE Yucca Mountain Project, concerning the application of burnup credit in the criticality safety analysis for spent fuel management systems. (See also related NRC research in previous section, items (i)(3) and (i)(4).)

(ii) Identify and acquire relevant LWR physics benchmark data from the international LWR-PROTEUS program now underway at PSI, Switzerland, and explore possibilities for extending the cooperative program to include specific IRIS-related benchmarks.

(iii) Identify and acquire relevant LWR physics benchmark data from the ongoing international REBUS program in Belgium (formerly co-sponsored by RES) and from recent work at the ECOLE and MINERVA facilities of CEA/Cadarache in France, and explore possibilities for cooperative work on additional benchmark experiments to address specific IRIS validation issues.

(iv) Pursue active NRC participation in relevant international programs, including experiments (e.g., items ii and iii above), code-to-data benchmarks, and code-to-code benchmarks, conducted by the IAEA, the European Commission, OECD/NEA.

(e) NRC Research Objectives and Plans

The NRC research objectives are to establish and qualify independent analysis capabilities and develop technical insights needed for assessing the adequacy of the applicant's safety analyses. For analysis issues involving reactor neutronics and decay heat generation in the PBMR and GT-MHR designs, the following research activities are planned:

1. Familiarization with Applicant's Codes and Methods: In coordination with preapplication review activities, gain familiarity with the reactor neutronics codes and decay heat algorithms and associated analysis assumptions, validation data, and uncertainty treatments being used by the applicants. Incorporate insights and questions arising from this familiarization process into the prioritization, planning, and execution of the NRC's overall research efforts in this and related technical areas.

2. Initial Exploratory and Scoping Studies: Use available independent codes (e.g., MCNP/MonteBurns, SCALE/NEWT/ORIGEN-S, SCALE/SENSIT, WIMS/MONK), and available applicant codes as needed, to perform exploratory and scoping analyses on selected IRIS issues such as described in Section (a) of this chapter. Incorporate insights and questions arising from these exploratory and scoping studies into the

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prioritization, planning, and execution of the NRC's overall research efforts in this and related technical areas.

3. Preparation of Modern Cross-Section Libraries: Using the upgraded AMPX code system, supplemented by NJOY as needed, prepare and test state-of-the-art master cross section libraries for use in performing exploratory and confirmatory analyses on reactor safety and material safety issues involving conventional and advanced reactor technologies, including ALWRs.

4. Preparation and Testing of Spatial Kinetics Model: Develop a PARCS input model of the IRIS reactor geometry, and using appropriate lattice physics depletion analysis tools with state-of-the-art cross section libraries (see previous item), prepare the design-specific nodal data tables needed for performing spatial kinetics analyses with the PARCS code (coupled with TRAC-M thermal-hydraulics).

5. Validation for Depletion and Decay Heat Analysis: Review existing and planned validation databases (e.g., spent fuel isotopic assays and decay heat calorimetry) and perform sensitivity analyses, based on methods developed in recent years under RES sponsorship, to help assess their applicability to the IRIS fuel designs and operating parameters and to help prioritize further data needs and assess remaining validation uncertainties. Participate in cooperative programs for new experimental data as well code-to-data and code-to-code benchmarking activities.

6. Validation and Testing for Reactor Neutronics: Review existing and planned validation databases (e.g., critical experiments, worth measurements, reactor tests) and perform sensitivity analyses, based on methods developed in recent years under RES sponsorship, to help assess their applicability to IRIS reactor neutronics phenomena and to help prioritize further data needs and assess remaining validation uncertainties. Participate in cooperative programs for acquiring new experimental data and conducting relevant code-to-data and code-to-code benchmarking activities.

(f) Resources and Schedule

Activity	Estimated Level of Effort	Estimated Completion
1. Familiarization with Applicant's Codes and Methods	2 staff months	July 2003
2. Initial Exploratory and Scoping Studies	6 staff months	January 2004
3. Preparation of Modern Cross-Section Libraries*	12 staff months*	November 2003*
4. Preparation and Testing of Spatial Kinetics Model	9 staff months	May 2005
5. Validation for Depletion and Decay Heat Analysis	12 staff months	December 2005

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6. Validation and Testing for Reactor Neutronics	9 staff months	April 2006

* The same work also appears in the HTGR plan. The resulting master cross section libraries will be generically applicable to all reactor types.

III.D.5 HIGH TEMPERATURE MATERIALS -- ALWRs

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TO BE SUBMITTED LATER

III.D Advanced Light Water-Cooled Reactors

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6 Instrumentation and Control

6a Description of Issues

The new generation of advance reactor concepts, both for high temperature gas-cooled reactors and for advance light water reactors will be the first opportunity for vendors to build new control rooms in this country. The advances that have been made in the development of many of the current generation of operating reactors in other parts of the world will be used in the design and construction of any new plant constructions in the US. The new plant will have fully integrated digital control rooms, at least as modern as the N4 reactors in France or the ABWRs. In addition the desire for much smaller controls room staffs for economic reasons will also push the plants in the direction of more automation as has been seen in the natural gas fire power plants. The use of multiple modular plants will also require significantly more complex control of both the primary instrumentation and control systems and all of the support systems including the switch yard.

These plants will be designed for autonomous operation with a minimum of supervision by plant operators for long periods of time. This will include automated startups and shutdown, and changes of operating modes. The operating crews will be very small compared with current generation nuclear power plants, as few as three operators for ten modules. This will require that not only normal operations but off normal operations and recovery be much more highly automated. To make modular reactor concepts effective the plant must work like a single larger plant. The will require a level of automation and coordination that is heretofore unheard of in the nuclear power industry. How plant control and safety systems will deal with reorganize itself to deal with partial failures of interconnected particularly at the switch yard and the control room will need to be investigated.

Because of the longer fuel cycles and much longer time between maintenance outages the plants will require much more extensive use of on-line monitoring and diagnostics, and predictive maintenance. Instrumentation will be needed to support this increased automated surveillance and the issue of how these systems will integrate with the control systems will need to be investigated. Additionally, because some of the systems that will be used as part of this new generation of light water cooled reactors will be operating in conditions that they have not previously been used at it is expected that several new kinds of sensors will be developed to support these reactors. These new sensors will need to be investigated. There may be conditions, such as the use of ultra long life cores that will require new neutron detectors to be use in this kind of reactor that will require changes in the way we do design and safety calculations (drift, calibration, response time, etc)

It has been seen in the application of highly automated control rooms in other industries that the availability of computer resource to use of modern control theory controllers to increase plant availability, and decreased workload on operators will drive control and diagnostic systems in this direction. It is likely that these new light cooled reactors will use some of these advance modern control methods. These could include simple feed forward controllers, non-linear controllers, neural-fuzzy controllers or even more exotic methods. Likely control algorithms that may be used in high temperature gas cooled reactors will need to be reviewed.

6b Risk Perspective

Instrumentation and control systems have potentially very high risk importance to plant safety, primarily because a failure can prevent operators or automated systems from performing their intended safety function. In most risk analysis performed to date have not identified instrumentation and control systems (including RPS and ESFAS) as risk important primarily due to the multiple redundances in these systems and there general lack of common mode failure. However, a failure in these systems can prevent the safe shutdown of the plant or permit other unattended actions to occur. For this reason these systems tend to be very important in assuring maintaining a set level of risk (high Risk Achievement Worth).

For advanced light water cooled reactors the transient response to accidents such as ATWS will be an important risk contributor. There will also be potentially new instrumentation and control failures that will initiated accidents by causing unexpected actions. The new reactor designs will need to be reviewed to determine if they are vulnerable to potential controller failures.

In all risk studies to date few instrumentation and control systems have been modeled in any detail. To adequately understand how the much more complicated digital instrumentation and control systems will be used and to support the risk informed licensing that is proposed for high temperature gas cooled reactors this area of risk modeling will need to be expanded. This will also be needed to support the interface with control room human interface research. Because of the lack of adequate models and data to support risk analysis the uncertainties in this area are relatively high and can only be reduced by significant new research in this area.

6c Related NRC Research

The NRC Research Plan for Digital Instrumentation and Control (SECY-01-155) outlines current and future research into several areas of emerging instrumentation and control technology and applications that will be used in the high temperature gas cooled reactors. These included smart transmitters, wireless communications, advanced predictive maintenance and on-line monitoring methods and enhanced cyber security issues. The NRC has recently started new research programs in the areas of wireless communications and on-line monitoring. This research will support the development of review guidance for NRR for these new and improve technologies, and will also support there introduction into advance reactors.

6d Related Research and International Cooperation

The national and international research community has been involved with research and development of advance control and monitoring systems for nuclear power plants for many years. The international community, particularly in Europe, Japan and Korea have developed integrated advance control rooms and done much more research in the areas of automation of plant operations and advance plant monitoring and diagnosis that we have done in the US. There will be significant opportunities for international cooperation in this area.

Westinghouse is doing detailed control systems design studies using plant simulators to help optimize control system design. There may be an opportunity to piggy back on to some of these research programs, particularly in the areas of advanced control algorithms and control of multiple plant modules.

One of the major areas of research outlined on the Department of Energy (DOE) Long-term Nuclear Technology Research and Development Plan is the Instrumentation and Control area. Several of the research topics proposed in this plan are of particular interest to high temperature

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gas cooled reactors including advance instrumentation, such as robust communications and wireless sensors, smart instrumentation and condition monitoring. Also of interest were research into distributed computing, condition monitoring, advanced control algorithms, on-line monitoring.

As part of the implementation of this long term research plan DOE has developed NERI six programs in this area. These include research in the areas of automatic generation of control architectures, self diagnostic monitoring systems, smart sensors. It will be important for the NRC to use the results of this work in its

6e NRC Research Objectives and Plans

NRC will need to conduct research into a number of areas in support of both high temperature gas cooled reactor and advance light water cooled reactors. Some of this research will support both of the areas, while some will be needed to support only one of the concept. For ease of presentation and to present double counting of resources all of the research that will be needed to support the high temperature gas cooled reactor research and generic research was described in section III.C.6. Only research that is needed to support advance light water reactors will be discussed here.

1) Analysis of the requirements and potential issues involved with advanced cooled reactor instruments

This effort would review the requirements for and the development of new instruments to support design, construction and operation of advance light water cooled reactors. These will include new neutron detectors needed to support ultra long life cores.

2) Analysis of on-line monitoring systems and methods and advanced diagnostic methods needed to support advance light water reactors.

This effort will review both current methods and investigate the required development of instruments and techniques to support this the current availability and maintenance schedules.

These research program will be supplemented by the ongoing research program in digital instrumentation and control as outlined in SECY-01-155, "NRC

6f Resources and Schedule

1. Analysis of the requirements and potential issues involved with advance light water cooled reactor instruments

FY	2002	2003	2004	2005	2006
FTE	0.1	0.1	0.1	0.1	-
$K	$50K	$150K	$100K	$50K	

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2. Analysis of on-line monitoring systems and methods and advanced diagnostic methods needed to support advance light water cooled reactors

FY	2002	2003	2004	2005	2006
FTE	0.1	0.1	0.1	0.1	0.1
$K	-	$50K	$100K	$150K	$100K

6g Priorities

1) Analysis of the requirements and potential issues involved with advance light water cooled reactor instruments

This work is **HIGH** priority, since it would result in the development of issues that would be needed to advice the vendors as to possible design issues with there plants. The review guidance developed in this research will also provide NRR with the information needed to review these instruments and there application to advance light water cooled reactors.

2) Analysis of on-line monitoring systems and methods and advanced diagnostic methods needed to support advance light water cooled reactors.

This work is a **MEDIUM** priority, because it will support other on-going research in this area both at the NRC and internationally. This work will need to be specialties to the specific issues of advance reactors but will follow on to the work that is currently planed else where.

III.D.7 PRA for Advanced Light Water Reactors

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a. Description of the Issue

Both industry and the NRC are utilizing PRA in the licensing process for advanced reactors. Also given the NRC shift to risk informed regulation, reliance on risk information for licensing ALWRs, and in particular AP1000 and IRIS, will be increasing. However, there is limited PRA experience with these new reactor types. Lack of experience applies to limited scope PRAs such as Level 1 and 2, as well as full scope PRAs, including external events and diverse operational states. Therefore, the research summarized below is required for the staff to be able to independently review the PRAs submitted as part of license applications for ALWRs (AP1000 or IRIS) .

b. Risk Perspective

Maintaining safety, increasing public confidence, and making more effective decisions are three strategic performance goals in NRC's Strategic Plan (NUREG-1614). Whenever a new technology has been introduced into the nuclear reactor industry or a new phenomenology has been found, the NRC conducted an independent investigation of that technology or phenomenology. ALWRs are new designs and, therefore, the current LWR PRA experience may be of limited use. The limitations of current PRA experience applies to the scope of the PRA (e.g., there is limited PRA experience for low power and shutdown); to the risk metrics used (risk metrics in addition to core damage frequency or large early release may be needed for these designs); and most importantly, to the design, systems, and safety approach applied. Future licensees have indicated that PRA and PRA insights will be an integral part of their license application. For the new ALWRs, justifications regarding design safety, safety margins, and defense-in-depth will be based largely on results and insights from PRA evaluations. The staff needs to gain an understanding of the new designs, contributors to risk, and associated uncertainties. The tasks described in this research plan will enable the staff to achieve this objective and thereby contribute to meeting the agency's strategic performance goals discussed above.

c. NRC Research Objectives and Plans

The objective of this plan is to identify and develop PRA methods, data, and tools needed to support an independent staff review of ALWR PRA submittals.

This plan is comprised of two tasks. The first task is to develop methods, data, and tools for modeling ALWRs (AP600 or IRIS) design and operational characteristics which are fundamentally different from those of a conventional LWR. The second task is to develop guidance to the staff for the review the ALWR PRAs.

1. PRA Supporting Analyses. There are fundamental tasks that need to be addressed to support either performing an independent PRA or reviewing the submitted PRA. These tasks are described below.

1.1 Risk Metrics. The concepts of CDF and LERF may be directly applicable to ALWR designs. However, consideration of the Level 3 PRA results are needed for all advanced reactors, including ALWRs. Specifics to be examined include acceptance criteria for methods to estimate both the radionuclide transport, especially for slow developing acci resulting dose to the public. In these two cases, small changes in assumptions models can lead to vastly different results.

1.2 Initiating Event Identification and Quantification. The events that chall generation of LWRs may not be applicable to ALWRs, and especially IRIS. It i PRA to correctly and comprehensively identify those events that have the pote accident. Therefore, understanding what events can occur (as a result of desi equipment failures, and human errors) that challenge the plant operation comp in assessing the risk associated with a given reactor design. Extensive effort i AP1000 is not expected.

1.3 Accident Progression and Containment Performance (including sour likely accident progression phenomena will have to be determined based on o previous experiments, experience in other industries, and expert judgment. S accident progression, and source term for ALWRS will be different from those must be understood. For IRIS, containment for multiple modular units on a sit considered. It is noted that source term issues are addressed in Section D.2 of plan.

1.4. System Modeling. Passive systems are used more extensively in ALWF In addition, the advanced designs will incorporate digital systems. Both need Passive systems have been treated in PRAs as either initiators (e.g., LOCAs) failures. As a result, current PRAs model only the performance of active syste logic which is suitable for such purposes. It is not clear that this approach wou modeling passive systems exhibiting slow evolutionary behavior during accide the failure modes of such systems need examination including whether and ho modeling approaches can be amended.

Digital systems typically have not been considered in the past PRAs. In advan however, digital instrumentation and control systems will be the norm. The reli systems is being addressed in another part of this RES plan (see Section III. D modeling issues concerning digital system performance should be addressed should be developed for incorporating digital system failure in the PRA logic.

The uncertainties associated with the development of modeling the failures of digital system also needs to be addressed.

1.5 Data Collection and Analysis. ALWRs introduce different systems and hence, LWR data may not be applicable. The use of appropriate data is cruci assessment of the risk associated with a given reactor type. Therefore, collec data applicable to ALWRs (especially IRIS) is essential.

Furthermore, this task includes addressing the uncertainties associated with th Understanding the uncertainties is a very important aspect for any PRA; it is m for these types or reactors given the limited or lack of operating experience an use of the PRA in the ALWR licensing process.

1.6 Human Reliability Analysis. The operator role in the ALWRs (to be built on the premise that if an event occurs human intervention will not be needed for an extended period of time) is not well understood. Issues related to the needs for reliable performance (e.g., staffing and training) are part of a different activity of the RES plan (see Section III.B). New human reliability methods, such as ATHEANA, were developed to assess the impact of human performance on plant safety when dealing with long-term and slowly evolving accidents, such as those expected to be predominant in accident sequences related ALWRs. Under this subtask these related efforts will be coordinated to determine if (and what) modifications are warranted to appropriately incorporate the impact of human performance in an ALWR PRA.

1.7 Other Events (internal flood, fire and seismic). As with any design that uses digital I&C, failure possibilities of electronics will need to be addressed. Specifically, the response of digital electronics in a fire or flood is expected to be quite different than that of electromechanical components. The differences may not be just in probability but also in the kinds of failures that could potentially occur (e.g., can an active failure occur?). Furthermore, in light of the new design, all of the external events must be applied to ALWRs from a scoping perspective to see if anything unique might occur.

1.8 Quantification. The SAPHIRE code would be used in the performance of an independent PRA or in examining the applicant's PRA results. The code could use modifications, some of which are of particular interest when contemplating a full scope PRA (external and internal events, full and low power). Such a PRA will generate many more "cut sets" than are comfortably handled now. In addition, the rationale developed for other designs for pruning the results may not be appropriate for an ALWR design in which low frequency sequences may dominate the risk. Other computational capabilities need to be addressed to see if methodological steps in them need altering for a different design with an expected different risk profile. These capabilities include current source term and consequence analysis tools, and the development and use of new modeling techniques, for example, dynamic modeling.

1.9 Other Operational States. The design and intended operation of the ALWR must be examined so that any unique characteristics of them during less than full power operation can be correctly accounted for in the PRA.

1.10 Multiple Modules. Current PRAs are usually performed for single units or sometimes for two sister units. The IRIS is a modular design, which may be deployed as individual units or a group of units. The PRA to be submitted as part of licensing application will have to address potential interactions among the multiple units, and NRC will need to be prepared to review these issues. Examples are (1) what are the common systems? (2) what are the cross ties, if any, between units? What is the effect of limited operator staffing and shared systems when accidents involve more than one unit, as could be the case for common cause initiators such as seismic and external flood events? In addition, human reliability models need to be examined to understand if errors of commission or omission more likely when multiple units share the same control room.

1.11 Safeguards and Security. Developing accident progression and containment performance can generate explicit information regarding the safeguards and security for the design. We need to explore how this can be accomplished in the most efficient manner and what other areas of the PRA studies can assist in this endeavor.

2. PRA Review Guidance. Under this task guidance will be developed explaining how the results of Task 1 can be used by the NRC staff for its review of an ALWR PRA submittal.

d. Related NRC Research

The review of AP600 performed by the NRC as well as other LWR PRA work will be heavily utilized in performing the tasks of this plan. Also the work on source term and human performance discussed in other parts of this RES plan will be used in performing this work. In addition, ongoing work under the programs ATHEANA and SAPHIRE PRA code will support this effort.

e. Related International Cooperation

The possibility for cooperating with the International Consortium for Development of Design (Oak Ridge National Laboratory, Tokyo Tech, and CNEN Brazil) who are or will be working on the PRA for IRIS will be explored.

f. Resources and Schedule

The estimated resources for this five-year research plan are given in the following page. The schedule at the moment is focusing on IRIS because the bulk of this work will focus on IRIS. The work will be performed in an iterative process incorporating new information as it becomes available: Scoping results will be available in about one year from the initiation of the work, initial results within 2 years, and preliminary results in about 2½years. Initial review guidance will be prepared in about 3 years so the staff will be ready to review the PRA which will be submitted along with the construction and operating license (COL). The draft results will be completed in about 4 years after the initiation of the work. This will provide a basis for performing comparisons with the ALWR PRAs (submitted as part the license application). A final review guidance will be prepared at about 5½years to help the staff in the design certification process.

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Resources

Task	Resources	
	Staff-months	Dollars*
1. PRA Supporting Analyses (AP1000 and IRIS)		
1.1 Risk metrics	6	$150
1.2 Initiating event identification and quantification	12	$300
1.3 Accident progression and containment performance (including source term)	see other research	
1.4 System modeling including uncertainties	6	$150
1.5 Data collection and analyses including uncertainties	12	300
1.6 Human reliability analysis	6	$150
1.7 Other events (internal flood, fire and seismic)	6	$150
1.8 Quantification	6	$150
1.9 Other operational states	6	&150
1.10 Multiple modules	6	$150
1.11 Safeguards and security	6	$150
Subtotal	72	$1800
Project Management	18	$450
2. PRA Review Guidance	6	$150
TOTAL	**96**	**$2400**

*Dollars are estimated assuming 1.0 staff-month of an FTE costs $25K.

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IV. NRC PRIORITIES

Recognizing that the realities of budget constraints and competing priorities will dictate how much of the desired advanced reactor research will be done, it is important to establish guidelines for overall prioritization of research activities. These guidelines need to consider such factors as safety significance, uncertainty and schedule as well as industry v. NRC responsibility. Listed below are guidelines that can be used to help establish the plan (each section of this plan also provides description of the relative priorities within an area):

Is the research essential for NRC to make independent findings regarding plant safety?

1. When is the research needed:

   - is it a prerequisite for identifying future needs?
   - is it tied to a review schedule?

2. Is the research directed toward:

   - determining safety margins?
   - reducing uncertainty?
   - establishing that safety criteria (e.g., dose) are met?

6. Could industry be responsible for the research:

   - basis?
   - NRC role (e.g., public confidence)

4. Could it be done in a cost-effective (e.g., cooperative) fashion?

Depending upon the answers to the above, the proposed research can be ranked in a relative fashion.

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V. IMPLEMENTATION

Considering budget constraints and industry schedule, NRC will have to prioritize its research activities by addressing the questions raised in Section IV. Using the guidelines set in that Section, the needed research activities can be ranked and schedules can be established. NRC will have to continue to draw upon the existing international HTGR experience and research. To off-set costs, due consideration would have to be given to future cooperative efforts both domestically and internationally. Some shared research with the industry is also expected. Most importantly, at the pre-application review phase, the applicant will have to provide the NRC with complete and detailed plant-specific design-, safety-, and technology-related information so that at topical meetings, NRC can raise specific issues. The discussion should lead to a clear understanding of where the information gaps exist and what additional tools and data that the NRC would have to develop to review the licensee submittals at the review stage.

VI. Domestic Research
The current interest and the pre-application review of the PBMR and likely request in the near future for pre-application review of the GT-MHR, combined with the meager domestic resources have resulted in many cooperative research efforts among the government, national laboratories, industry, and universities. Sometimes the domestic efforts cannot be separated totally from the international efforts as many foreign organizations have joined ongoing activities. This is not only due to the collaboration efforts but also due to the fact that entities like General Atomics Company (GA) and Exelon are part owners and have commercial interest in plants outside the United States. Within the DOE, through its Office of Nuclear Energy, Science, and Technology, conducts research activities that are carried out by the national laboratories and various US universities. Additionally DOE also funds the industry to conduct other research. A memorandum of understanding exists between DOE and NRC to conduct research in a cooperative and cost sharing process in order to conserve resources and avoid duplication.

DOE Initiatives and Sponsored Research

For many years ending in the early 1990s, DOE sponsored the modular High Temperature Gas-cooled Reactor (MHTGR) Program. This program culminated in a draft safety evaluation review by the NRC of the MHTGR design in 1989 (NUREG-1338). Subsequently, in the late 1990's, due to the continued focus on nuclear energy being a viable energy source, DOE initiated a new program called the Nuclear Energy Research Initiative. NERI is a Research and Development (R&D) program to stimulate universities, industry, and national laboratories to innovate and apply new ideas to old problems. The DOE research money for generic work on both HTGR and ALWR comes from NERI. The NERI budget for FY2002 is $27.1 million. There is strong competition for this pool of money from the international activities, Generation IV activities, and current efforts to optimize the existing nuclear power plants.

The collaborative efforts with DOE will optimize the domestic resources for research needed by the NRC, but the many other areas covered by NERI also compete for resources from advanced reactor research. However, DOE is committed to allocating funds among different research areas. For instance, DOE solicited proposals this fall for an Early Site Permit (ESP) demonstration project, a comprehensive evaluation and identification of the activities, schedule and resource requirements to implement an ESP application for a preferred site. It is possible some of the Generation IV research could be applied to the near deployment plants -- the two HTGRs (PBMR and GT-MHR) and the two ALWRs (AP1000 and IRIS). However, Generation IV activities are relatively new and need to be reviewed for applicability to the near deployment plants.

The cooperative research efforts between DOE and the Electric Power Research Institute (EPRI) focus on advanced light water reactors and research to optimize the operations of the current operating fleet of nuclear plants. EPRI developed, in cooperation with the Nuclear Energy Institute, and other nuclear industry organizations, "Nuclear Energy R&D Strategy Plan in Support of National Nuclear Energy Needs" and provided it to DOE to initiate joint planning and coordination toward common R&D goals. EPRI's HTGR database is to be complied this year.

Industry and University Research

GA has an on-going joint project with Russia to build a HTGR for plutonium disposition. This project will lead to the development, fabrication and demonstration of key GT-MHR components such as the turbo machinery and its major components, reactor vessel and internal materials, and the fuel based on a plutonium oxide coated particle fuel. While the Russian plant is not a commercial venture, the research for this plant could be transferrable to the commercial GT-MHR design.

The Massachusetts Institute of Technology (MIT) is conducting research on a modular high temperature gas cooled pebble bed reactor. Students and faculty are engaged in research on core neutronics design, thermal hydraulics, fuel performance, economics, non-proliferation, and waste disposal. The objective of this research is to develop a conceptual design of a 110 Mwe pebble bed nuclear plant which could be used as a demonstration of its practicality and competitiveness with natural gas. In addition to MIT with its consortium of United States universities, national laboratories, and industries, this research involves international collaborations with Germany, Russia, China, Japan, and South Africa.

NRC Research

Since early 2001, Exelon met with the staff and DOE on several occasions to discuss topical issues for supporting PBMR pre-application review. Discussions during these meetings have identified several areas where research will be needed. In December 2001, GA met with the staff and DOE to initiate discussions on pre-application review activities on the GT-MHR. This meeting affirmed that much research that has already been conducted could be applicable to the commercial GT-MHR. Additionally, some of the PBMR issues are also relevant to the GT-MHR.

The NRC plans to conduct research and testing on new reactor designs to provide an independent capability to judge the safety of the proposed design and confirm information submitted by applicants. Research related to high temperature gas-cooled reactors (HTGR) fuel performance and qualifications, high temperature materials and graphite behavior, and thermal-hydraulic and core heat-up in the HTGR will be needed. The existing thermal-hydraulic and analytical codes may also have to be modified to address design-specific features and phenomena in the new reactors. NRC is considering modifications to the existing codes and to conduct additional code developments in FY 2002 for both severe accident analysis and thermo-fluid dynamic codes for the HTGRs.

For HTGRs, the fuel is the key design safety feature. Research on pebble fuel performance including fuel behavior during heat-up, and fission product release and transport from the irradiated fuel would be useful to confirm the performance of this key design aspect. The existing codes used for analysis of the LWRs can, in some cases, be adapted for the HTGR. However, they would need modification, including the capability to model air and water ingress. For accident analysis for the HTGR, it is expected that fission product release and transport can also be modeled by using the existing codes, with some modifications. Use of different materials and significantly different environmental conditions than the LWRs support the usefulness of materials engineering research to support the review of the advanced reactors. The high-temperature operating conditions, the use of graphite as the moderator and reactor-core structure material, and the use of helium gas as the coolant, raise unique issues with respect to the long-term performance, age related degradations and aging management, and structural

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integrity aspects of safety components, which could be explored via research. Additionally, much data can be obtained through international cooperation.

For the AP-1000 and IRIS, the need for confirmatory research at various facilities may be necessary for thermal-hydraulic testing. The existing codes may have to be assessed for the conditions of operations of these reactor designs and to identify any needed improvements in NRC's thermal-hydraulic codes. NRC plans to conduct some research for FY2002 on thermo-hydraulic research for the ALWR.

Areas of Domestic Research

The following is a list of some of the important areas of research for advanced reactor designs. The list is mostly on HTGR research conducted in the last decade, is currently being conducted, or is planned to be conducted soon. Research on the ALWR has subsided since the NRC certified three of the ALWR designs in the 1990's - System 80+, ABWR, and AP600. The name of the prime organization(s) involved in the research is in parenthesis. With few exceptions the staff has not formally reviewed any of this research and will need to do a very thorough review of the research that is applicable to the current designs. The gathering of the information will also pose a big burden since there does not seem to be a complete listing of all the specific research on HTGR or Advanced Light Water Reactor (ALWR). This issue will be alleviated some since the applicants will provide the references to specific research:

Studies on Power Conversion System (GA)
Irradiated Graphite Strength Tests (PNL, ORNL)
Graphite and Carbon-Carbon Composite Research (ORNL)
Fission Product Release (GA)
Seismic Response of Reactor Core (LANL)
Water Ingress into Reactor Core (ORNL)
High Temperature Metallic Components (GA)
Thermal Stability of Iconel 617 (ORNL)
Fuel Element Structural Design (GA)
Fort St Vrain Operational Experience (Public Service of Colorado)
PRA of HTGR (GA)
Passive Decay and Residual Heat Removal (Bechtel, CE, and GA)
Seismic Analysis of HGTR (Bechtel)
Helium Circulator Design (GA)
Reactor Physics Uncertainties (GA)
Heat Transfer (Bechtel, CE, and GA)
Fission Product Retention in TRISO Coated Fuel (ORNL)
Environmental Aspects (GA, Bechtel, Stone & Webster)
Pressure Vessel Design Codes (GA)
Creep-Fatigue Damage (GA, CE)
In-core Self-Powered Flux and Temperature Probes (ORNL) 2001 NERI Award
High Performance Fuel Design (MIT) 2001 NERI Award
Reactor Physics and Criticality Benchmark Evaluations for Advance Nuclear Fuel (FramatomeTechnologies) 2001 NERI Award
MELCOR Work on HTGR (NRC Funded for FY2002, Sandia)
Thermo-hydraulic Tests for ALWR (NRC Funded for FY2002)
ALWR In-Vessel Heat Retention Studies (DOE, UC/Santa Barbara, Westinghouse)

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ALWR Development Program (DOE, EPRI)
LWR Uranium-Thorium Dioxide Fuels (INEEL), 2000 NERI Award
Advanced Zirconium Based Alloy Corrosion at High Burn-Up (INEEL), 2000 NERI Award
Risk-Informed Assessments for Future NPP (INEEL), 2000 NERI Award