November 9, 2001

MEMORANDUM TO: Michael E. Mayfield, Director, DE:RES
Scott F. Newberry, Director, DRAA:RES

FROM: Thomas L. King, Director, DSARE:RES /RA/

SUBJECT: FUTURE REACTOR LICENSING RESEARCH PLAN

In reference to my memorandum, dated October 17, 2001, to you, attached is the time-line for development of the research plan (Attachment 1). Our objective is to ensure that the NRC has the ability to independently confirm and analyze licensee's safety claims. In developing the research plan, we will assume that the industry plans will proceed as described in the Nuclear Energy Institute letter to Chairman Meserve, dated August 10, 2001, with the exception of the recently announced slips in AP-1000 and Pebble Bed Modular Reactor (PBMR). Furthermore, the plan will encompass infrastructure development only, and will not address the effort to support NRR on review of applications. It should describe RES proposed plans to develop an infrastructure to support licensing of future reactor designs. The objectives of the research plan are to:

- identify what analytical tools and data should be developed to provide the agency with an independent capability to assess the safety of future HTGR and ALWR reactor designs.
- identify experimental work that should be performed to support NRC code assessment and/or to explore safety margins in the designs.
- identify work necessary to establish the technical bases for regulatory requirements appropriate for these new designs.

Such research will also help educate the staff on the technology and thus contribute to a more effective and efficient staff review.

The plan should be developed to describe what needs to be done and it's priority. This will then be consistent with the Future Licensing & Inspection Readiness Assessment (FLIRA) report, and provide information for the budget process where the impact of budget constraints can be assessed. Attachment 2 contains the outline of the research plan.

CONTACT: Raji Tripathi (RRT1), DSARE:RES
(301) 415-7472

H-4

The research plan will contain two separate sections -- one for HTGRs and another for ALWRs. For each reactor type, under various topics, the plan should include tasks, milestones, cost and schedule. For each topic, priorities should be assigned on the basis of relative importance of proposed research work, including justification. The major areas to be addressed are:

- Thermal-hydraulic/Fluid dynamics codes
  - What capabilities are needed and why?
  - What codes and why?
  - How will the codes be validated?
  - Experimental needs.
  - Schedule/milestones/end products
  - Estimated cost
  - Priorities

- Severe accident codes, including source term
  - What capabilities are needed and why?
  - What codes and why?
  - How to be validated (data, etc.)
  - Experimental needs
  - Schedule/milestones/end products
  - Estimated cost
  - Priorities

- Fuel performance models/data
  - Key issues
  - Analytical needs and why
  - Experimental needs and why
  - Schedule/milestones/end products
  - Estimated cost
  - Priorities

- High temperature materials including graphite (HTGRs only)
  - Key issues
  - Codes, standards, and methods needs
  - Experimental needs
  - Schedule/milestones/end products
  - Estimated cost
  - Priorities

- I&C
  - Key issues
  - Research needs and why
  - Schedule/milestones/end products
  - Estimated cost
  - Priorities

- PRA
  - Models/Approach/Data (?)
  - Schedules/milestones/end products
  - Estimated costs
  - Priorities

The staff responsible for various topical write-ups as well as for peer review of selected topics are identified in Attachment 3. Also included is a draft research plan for nuclear graphite that was prepared by Raji Tripathi at my request, and should serve as a model for format (Attachment 4).

If you have any questions, feel free to call me at 415-7499.

cc w/atts.:
A. Thadani/R. Zimmerman

- PRA
  - Models/Approach/Data (?)
  - Schedules/milestones/end products
  - Estimated costs
  - Priorities

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cc w/atts.:
A. Thadani/R. Zimmerman

Distribution w/atts.:

DSARE R/F	J. Muscara
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M. Cunningham	S. Bajorek
N. Chokshi	S. Arndt
D. Dorman	R. Meyer
P. Norian	C. Tinkler
P. Kadambi	F. Cherny
J. Ibarra	A. Cubbage, NRR
S. Rubin	D. Jackson, NRR
D. Carlson	

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OFFICE	DSARE		D:DSARE	
NAME	RTripathi:mmk by TKing		TKing	
DATE	11/09/01*		11/09/01*	

TIME-LINE FOR THE RESEARCH PLAN

Milestone	Completion Date
Division Input on Individual Write-ups	11/30/01
Consolidate Division Inputs	12/07/01
Internal Review of Draft Research Plan	12/14/01
Resolve Comments and Complete Draft Final Research Plan Send to NRR for comment	12/21/01
Send draft to ACRS	01/01/02
ACRS Sub-committee meeting	Early February 2001
Final Research Plan	Mid February 2002

Attachment 1

Outline For Advanced Reactor Research Plan

<u>Executive Summary</u> . . . . R. Tripathi

<u>Introduction</u> . . . . R. Tripathi

<u>Scope</u>: HTGR (PBMR and GT-MHR) and ALWRS (AP-1000) and IRIS

<u>Objectives</u>: . . . . R. Tripathi

<u>Discussion:</u> . . . . R. Tripathi

HTGRs (lead-in paragraph) . . . . R. Tripathi

- Thermal-fluid-dynamics codes . . . . C. Gingrich
  - Description of Issue(s)
  - Risk Perspective
  - Related NRC Research
  - Related International Cooperation
  - NRC Research Objectives and Plans
  - Resources and Schedule
  - Priorities

- Severe accident codes, including source term . . . . C. Gingrich
  - Description of Issue(s)
  - Risk Perspective
  - Related NRC Research
  - Related International Cooperation
  - NRC Research Objectives and Plans
  - Resources and Schedule
  - Priorities

- Fuel fabrication performance and qualification . . . . S. Rubin
  - Description of Issue(s)
  - Risk Perspective
  - Related NRC Research
  - Related International Cooperation
  - NRC Research Objectives and Plans
  - Resources and Schedule
  - Priorities

- Neutronics – (core physics/decay heat removal) . . . . D. Carlson
  - Description of Issue(s)
  - Risk Perspective
  - Related NRC Research
  - Related International Cooperation
  - NRC Research Objectives and Plans
  - Resources and Schedule

Attachment 2

– Priorities

Attachment 2

- Materials – High Temperature Materials .... J. Muscara
  Materials – Nuclear-Grade Graphite .... R. Tripathi
  – Description of Issue(s)
  – Risk Perspective
  – Related NRC Research
  – Related International Cooperation
  – NRC Research Objectives and Plans
  – Resources and Schedule
  – Priorities

- I&C .... S. Arndt
  – Description of Issue(s)
  – Risk Perspective
  – Related NRC Research
  – Related International Cooperation
  – NRC Research Objectives and Plans
  – Resources and Schedule
  – Priorities

- PRA .... A. Rubin
  – Model/Approach/Data ?
  – Resources and Schedule ?
  – Priorities

ALWRs (lead-in paragraph) .... R. Tripathi

- Thermal-hydraulic codes .... S. Bajorek
  – Description of Issue(s)
  – Risk Perspective
  – Related NRC Research
  – Related International Cooperation
  – NRC Research Objectives and Plans
  – Resources and Schedule
  – Priorities

- Thermal-hydraulic Experiments .... S. Bajorek
  – Description of Issue(s)
  – Risk Perspective
  – Related NRC Research
  – Related International Cooperation
  – NRC Research Objectives and Plans
  – Resources and Schedule
  – Priorities

- Severe accident codes, including source term . . . . C. Gingrich
  - – Description of Issue(s)
  - – Risk Perspective
  - – Related NRC Research
  - – Related International Cooperation
  - – NRC Research Objectives and Plans
  - – Resources and Schedule
  - – Priorities

- Fuel performance models/data . . . . R. Lee
  - – Description of Issue(s)
  - – Risk Perspective
  - – Related NRC Research
  - – Related International Cooperation
  - – NRC Research Objectives and Plans
  - – Resources and Schedule

- Neutronics . . . . D. Carlson
  - – Description of Issue(s)
  - – Risk Perspective
  - – Related NRC Research
  - – Related International Cooperation
  - – NRC Research Objectives and Plans
  - – Resources and Schedule
  - – Priorities

- Materials . . . . J. Muscara
  - – Description of Issue(s)
  - – Risk Perspective
  - – Related NRC Research
  - – Related International Cooperation
  - – NRC Research Objectives and Plans
  - – Resources and Schedule
  - – Priorities

- I&C . . . . S. Arndt
  - – Description of Issue(s)
  - – Risk Perspective
  - – Related NRC Research
  - – Related International Cooperation
  - – NRC Research Objectives and Plans
  - – Resources and Schedule
  - – Priorities

- PRA . . . . A. Rubin
  - – Model/Approach/Data ?
  - – Resources and Schedule ?
  - – Priorities

NRC Priorities .... R. Tripathi/J. Flack

Implementation .... R. Tripathi
- International Cooperative Research .... R. Tripathi
  - International organizations (IAEA, NEA)
  - Cooperative efforts w/other countries, including European Union
- Domestic Research .... R. Tripathi
  - DOE-sponsored, NRC-sponsored, Industry efforts

Schedules and Resources .... R. Tripathi

Staff Responsible for Individual Topical Write-ups

High-Temperature Gas-Cooled Reactors (PBMR/GT-MHR)

Division:	Staff	Topic	Peer Review
DSARE:	D. Carlson	Neutronics	
		Decay Heat Removal	
	C. Gingrich	Thermodynamic Codes	
		Severe Accidents	C. Tinkler
		Source Term	C. Tinkler
	S. Rubin	Fuels	R. Meyer
	R. Tripathi	Graphite	J. Muscara
		International research	
		Priorities	
		Implementation	
		Resources	
		Research Plan Consolidation	
DRAA:	A. Rubin	PRA	
DET:	S. Arndt	I&C	
	J. Muscara	Codes & Standards	F. Cherny
		High Temperature Materials	F. Cherny

Advanced Light-Water-Cooled Reactors (AP1000 and /IRIS)

Division:	Staff	Topic	Peer Review
DSARE:	S. Bajorek	Thermal-hydraulic Experiments	
		Thermal-hydraulic Codes	
	D. Carlson	Neutronics	
		Decay Heat Removal	
	C. Gingrich	Severe Accidents	C. Tinkler
		Thermodynamic Codes	
	R. Lee	Fuel (IRIS)	R. Meyer
DET:	S. Arndt	I&C	
	J. Muscara	Materials	F. Cherny

Nuclear-Grade Graphite[1] Research Plan

(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 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:

1. 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?

2. 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?

3. Since "new" graphite will be produced with "old" graphite technology because that is the only available experience and information base, various physical characteristics, such

[1] In HTGRs, nuclear-grade graphite is used in protective coatings of the fuel balls as well as for the moderator balls and in the structural elements, including the side and top reflectors, in the core region. The fuel carbon is not fully graphitized (it is typically, processed to below 1000 °C) and the side and bottom reflector is graphitized to about 2000 °C. Therefore, the available data for one is not necessarily applicable to another.

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.

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

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

6. What national and international standards should be developed for physical characteristics and operational performance of nuclear graphite?

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

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

(b) <u>Risk Perspective</u>

In HTGR, 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, temperature, 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) <u>Related NRC Research</u>

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

(d) <u>Related International Research</u>

IAEA

TECDOC-901 contains 27 papers presented at a 1997 international specialists meeting held in Bath, UK. Each of the representatives presented the safety issues and graphite experience in their country. Recognizing world-wide loss of collective knowledge because of impending retirements of experts, the IAEA decided to establish a centralized system for collection, storage and dissemination of an electronic graphite database.

With support from Japan, South Africa and the United Kingdom, the IAEA has established a database related to irradiated nuclear graphite properties. The objective of this effort is to preserve the existing world-wide knowledge on the physical and thermo-mechanical properties of the 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. Development of a site on the Internet for the database, with direct access to unrestricted data, is also in progress. Completion schedule of this effort is not presently known.

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 (others are: R&D on very high burn-up fuel, R&D and component testing of high efficiency recuperator designs, and materials development for turbine blades up to 900 °C for long creep life). The duration of the Coordinated Research Program (CRP) is from 2000 through 2005. Current status unknown.

NEA

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." The conference was co-sponsored by JAERI. Component behavior, including graphite performance, under normal and accident conditions were discussed. (Have the proceedings on the net.)

On October 10 –12, 2001, an NEA/OECD conference was 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. (No one from the NRC attended – Proceedings to be obtained.)

International Standards

International cooperation is also crucial in establishing consensus standards, as well as for developing acceptance and performance criteria, for nuclear-grade graphite.

Cooperation with Other Countries

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. There is a need for confirmatory research in some areas, and priorities were assigned. Additionally, it was noted that regulators from countries, such as RSA, who also have the same challenge as the NRC of the PBMR design review for licensing, may benefit from mutual cooperation.

(e) <u>NRC Research Objectives and Plans</u>

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 and 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 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 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 (NII and BNFL)?

What creep stress has been employed in prior UK graphite irradiation creep experiments (NII)?

What product QA requirements are imposed on the suppliers of AGR fuel sleeve graphite (BNFL)?

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 (biggest uncertainties), and (ii) what graphite materials property data is needed to define property distributions for the purpose of these analysis?

2. Graphite Qualification Program – Need to study irradiation behavior of graphite and also to measure graphite oxidation. Questions to be addressed include:

What is the 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?

Does sufficient graphite irradiation data (presumably from fuel sleeves) exist to apply to PBMR operation?

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

In the PBMR context, is the irradiation behavior of the new fuel sleeve graphite (Nitetsu coke), as observed from post irradiation examination of discharged fuel sleeves (CEGB/Nuclear Electric data), similar to that of the earlier VFT pitch coke graphite (NII and BNFL)?

What graphite materials test program is necessary?

Are there high dose ($>1x10^{22}$ n/cm^2 EDN) data for fuel sleeve graphite from MTR experiments available in the UK (NII and BNFL)?

3. Surveillance, Inspection and In-service Examinations

Should irradiation samples be taken from large (production) blocks of fuel sleeve graphite or from smaller, pilot plant scale billets (NII and BNFL)?

NRC needs to have 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. Additionally, there should be a clear understanding of the manner in which the surveillance/in-service inspection/sampling will be done so that there is reasonable confidence that the program will achieve its objectives.

4. International Consensus Standards for Nuclear-Grade Graphite

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

(f) Resources and Schedule

1. 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.

2. Graphite Qualification Program:

(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, and (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.

3. Development of Consensus Standards for Graphite Design and Fabrication

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: $?
Duration: ?? years

Priorities: Confirmatory research related to nuclear grade graphite is a high priority item. (Add discussion from the Workshop Summary Report)