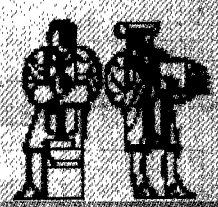
Idaho National Engineering and Environmental Laboratory

INEEL/EXT-99-00694
MIT-Nuclear Engineering
July 1999

STRATEGIC NUCLEAR RESEARCH COLLABORATION

FY99 ANNUAL REPORT



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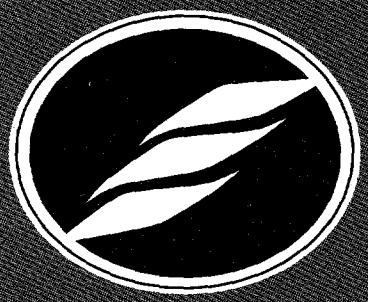

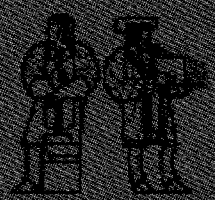

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IDAHO NATIONAL ENGINEERING AND ENVIRONMENTAL LABORATORY

AND

MASSACHUSETTS INSTITUTE OF TECHNOLOGY

STRATEGIC NUCLEAR RESEARCH COLLABORATION

FY99 ANNUAL REPORT

JULY 1999

Executive Summary

In a 1997 report entitled, "Federal Energy Research and Development for the Challenges of the Twenty-first Century," the President's Committee of Advisors on Science and Technology (PCAST) recommended inclusion of nuclear energy in the U.S. energy research and development portfolio. At the same time, mounting concerns over the role of greenhouse gas emissions in global climate change have sparked serious discussion of a proposed "carbon tax" or other governmental policy mechanisms that would serve to make electricity generated from non-carbon-emitting sources (such as nuclear energy) more economically competitive in the future. In the Kyoto agreement signed last year, Vice-President Gore committed the United States to work toward reducing greenhouse gas emissions to levels at least 5% lower than those emitted in 1990. Meanwhile, worldwide demand for electricity is expected to grow at a rate of nearly 3% per year through the year 2020. At this rate of growth, nearly two new 1000 megawatt generating plants will have to be built *each week* to meet this worldwide demand growth. These forces are combining to lead decision makers in the United States and in other countries to consider carbon free sources of safe, clean, economical electricity. Nuclear power has enormous potential to play a major role in the world's energy mix as well as in the United States, but only if existing US technical and policy challenges can be satisfactorily resolved. Significant research is needed now to provide a sound technical basis for addressing these challenges.

It is ~~against this backdrop~~ that the INEEL has created the Strategic Nuclear Research Collaboration. The SNRC brings together some of America's finest laboratory and university nuclear researchers in a carefully focused research program intended to produce "breakthrough" solutions to the difficult issues of nuclear economics, safety, non-proliferation, and nuclear waste. This integrated program aims to address obstacles that stand in the way of nuclear power development in the US. These include fuel cycle concerns related to waste and proliferation, the need for more efficient regulatory practices, and the high cost of constructing and operating nuclear power plants.

Funded at an FY99 level of $2.58M, the SNRC is focussing the efforts of scientists and engineers from the INEEL and the Massachusetts Institute of Technology to solve complex nuclear energy challenges in a carefully chosen, integrated portfolio of research topics. The result of this collaboration will be research that serves as a catalyst for future direct-funded nuclear research and technology development and which preserves and enhances the INEEL's role as America's leading national laboratory for nuclear power research. In its first year, the SNRC has focussed on four research projects each of which address one or more of the four issues facing further nuclear power development (economics, safety, waste disposition and proliferation-resistance). The projects are:

- Modular Pebble Bed Gas Cooled Reactor Development
- Actinide Transmutation by Lead-Bismuth Cooled Reactors
- Regulatory Excellence: Performance –Based Regulation for DOE Facilities
- Proliferation-Resistant, Low Cost Thorium Dioxide-Uranium Dioxide Fuels for Light Water Reactors

In its first year, there have already been some impressive returns on the SNRC investment. The

SNRC has, for example, been instrumental in earning the INEEL the designation as the Department of Energy's "Lead Laboratory" supporting the department's Office of Nuclear Energy and in winning an impressive share of the DOE's Nuclear Energy Research Initiative (NERI) program awards. The program, however, does far more than just position the laboratory to pursue new direct-funded nuclear R&D programs. The SNRC reflects INEEL's leadership responsibility to initiate research aimed at creating the next generation of nuclear technology options and solutions. This research is strategically focused so as to deliver enabling, catalytic solutions. It is substantial enough to deliver integrated, systematic solutions that have value on a national scale. It is built on a close and genuine collaboration between MIT and INEEL experts who can leverage one another's capabilities into "breakthrough" solutions.

This Annual Report describes technical work and accomplishments during the first year of the SNRC's existence.

The Origin of INEEL/MIT Strategic Nuclear Research Collaboration: Nuclear Power Issues and Opportunities

Currently, nuclear energy accounts for approximately 22% of U.S. electricity generation with zero emission of greenhouse gases. In fact, the use of nuclear power in the U.S. avoids the production of more than 147 Million tonnes of CO_2 each year (20% of the carbon emissions from the electrical generation section.) The growing awareness of the connection between our energy policy and our environmental goals is forcing policy makers to examine all sources of electricity more objectively than ever before. This is leading to a new appreciation of nuclear energy's importance as a part of our nation's energy supply portfolio and as an essential element in our environmental strategy.

In its November, 1997 report entitled "Federal Energy Research and Development for the Challenges of the Twenty-first Century," the President's Committee of Advisors on Science and Technology (PCAST) recommended strongly the inclusion of nuclear energy in the U.S. energy R&D portfolio. Some quotes from the PCAST report signal a recognition among top level science advisors that nuclear energy must be viewed as an integral part of any viable solution to increasing demand for electricity in an increasingly greenhouse-constrained world:

- "Nuclear fission currently generates about 17% of the world's electricity. If this electricity were generated instead by coal, world carbon dioxide emissions from fossil fuel consumption would be almost 10% larger than they currently are."
- "To write off fission now, as some have suggested, instead of trying to fix it where it is impaired, would be imprudent in energy terms and would risk losing much U.S. influence over the safety and proliferation resistance of nuclear energy activities in other countries. Fission belongs in the R&D portfolio."
- "It is important for the administration to acknowledge nuclear power as an energy option that could contribute substantially to meeting national and international emissions goals, if the concerns around it are resolved."

These sentiments, along with other social and political developments, have served to focus great interest on what may amount to a renewed commitment to nuclear energy as part of the overall U.S. energy mix. Still, there are a number of key obstacles that must be overcome before nuclear power can be viewed as a viable part of our future energy mix. Foremost among these are concerns about the economics of nuclear power, safety concerns, non-proliferation issues, and unresolved questions about nuclear waste.

Whereas the operating costs of many nuclear power plants are competitive with natural gas direct cycle electricity production, it is widely accepted that the capital cost of building new nuclear plants is so high as to prohibit new nuclear construction in a deregulated market. Continued improvements in nuclear operating economics relative to other power sources and major reductions in capital costs must therefore be achieved before nuclear can be presented as a serious contender for a long-term place in the U.S. energy mix. Concerns over greenhouse gases have sparked serious debate about the environmental costs of electricity generation by fossil fuels that may result in legislation or other policy-driven actions to level the economic playing field for nuclear and other non-emitting power sources. Technological and/or regulatory breakthroughs may also serve to significantly decrease the cost of nuclear power. Likewise, concerns about safety, non-proliferation, and nuclear waste may prove to be manageable through a focused R&D program that is specifically directed at ensuring that the benefits of nuclear

power are available to future generations of Americans.

A direct outgrowth of a joint INEEL/MIT workshop that was held April 13 and 14, 1998, the Strategic Nuclear Research Collaboration is strategically focused to provide breakthrough solutions to some of the major issues that are standing in the way of the advancement of nuclear power. Three broad areas are addressed in this research program: (1) Regulatory Excellence: Framework for DOE Facilities; (2) Advanced Reactor Technology; and (3) Global Fuel Cycle Research. Together, these broad areas address the inter-related issues of economics, safety, non-proliferation, and waste in an integrated and systemic way. Four specific projects covering these broad areas of research, were initiated in fiscal year 1999.

In the sections that follow, the conceptual framework of the INEEL/MIT SNRC, and atechnical overview of the projects undertaken are covered. Detailed appendices provide descriptions of the initial phase of research in each of thefour projects .

Strategic Nuclear Research Collaboration – Conceptual Overview

The basic concept of the SNRC is a simple one. The program consists of an integrated portfolio of nuclear research projects that are intended to meet the following criteria:

- strategically focused so as to specifically address the major obstacles standing in the way of further nuclear power development,
- results in "leapfrog" incremental technology improvements to create the next generation of nuclear technology options and solutions,
- large enough to produce important findings of value on a national scale, and
- a genuine and close collaboration between laboratory and university researchers from the INEEL and MIT to meet the tests of technical excellence and practicality.

The program constitutes a fundamentally new type of laboratory/university collaboration which brings together some of the brightest minds in America's laboratories and universities in a highly focused and highly leveraged way. The SNRC responds to the findings and recommendations of the President's Committee of Advisors on Science and Technology (PCAST) report on Federal energy research and development. As a first step in developing this response, the INEEL sponsored a joint INEEL/MIT workshop entitled "Nuclear Energy for a Greenhouse Constrained World." Held on the MIT campus in Cambridge, MA April 13 and 14, 1998, the workshop involved chosen representatives of the INEEL and MIT as well as selected outside experts. The workshop consisted of a mix of structured presentations from experts representing the technical areas of fuel cycle technology, nuclear regulation, and advanced reactor technology, as well as discussions about potential paths forward for nuclear power. The workshop was followed by discussions between INEEL and MIT experts in each of the three areas for the purpose of identifying a portfolio of research. Each individual element of the portfolio, as well as the integrated portfolio as a whole, was rigorously selected to meet the criteria identified above.

It is intended that the SNRC serve both as a catalyst for renewed commitment to nuclear energy development, and as an opportunistic response to an improving climate for nuclear development. As such, the SNRC must produce results that directly address the major obstacles which must be overcome prior to any serious consideration of future nuclear development in the United States, and must do so in a thoroughly practical way. At the same time, this portfolio of research must serve to increase the visibility of the INEEL and MIT and cement our positions as the nation's

leading institutions for future nuclear energy research and development.

With these goals in mind, the research outlined in this annual report was developed, and should be evaluated, as an integrated research portfolio. The elements of the research portfolio were specifically selected based on a systematic analysis of the weaknesses that currently impede the development of nuclear power and the recognition that the path forward must be based on an integrated treatment of these recognized weaknesses. This integrated research portfolio consists of four currently funded projects. Each project has been specifically chosen for its ability to address the inter-related issues of nuclear economics, safety, non-proliferation, and waste.

Strategic Nuclear Research Collaboration – Project Overview

The four projects funded during FY99 are identified below and are described in detail in the sections that follow.

Modular Pebble Bed Reactor Project

This project consists of a detailed conceptual analysis of the pebble bed high temperature gas reactor (PBHTGR) as a candidate source of safe and economical electric power for the future. The passively safe, modular gas reactor concept has received considerable attention as a potential source of electric power for developing countries. This research focuses primarily on improvements in fuel particle design, manufacturing, and reliability; PBHTGR spent fuel disposal; PBHTGR non-proliferation issues; development of core neutronics models; PBHTGR control issues; safety, risk, and licensing issues; manufacturing and producibility issues; and potential uses of waste heat from PBHTGR operation. The objective of the research is to evaluate the economics, safety, and proliferation characteristics of this promising reactor design as a source of future nuclear power and to identify potential solutions for recognized weaknesses.

Actinide Transmutation by Lead-Bismuth Cooled Reactors

This project explores the potential for using reactor transmutation of spent fuel to greatly reduce the quantities of long-lived radionuclides in spent fuel. Transmutation has the potential to reduce the time horizon for concern about environmental issues associated with spent fuel disposal from the current 100,000 years to time periods on the order of only 1,000 years. This project can result in improved public acceptance of nuclear technology by addressing concerns about long-lived radioactivity in nuclear waste. Current work focuses on neutronics, fuel selection, materials issues, and plant engineering.

Regulatory Excellence: Performance-Based Regulatory Framework for DOE Facilities

This project is directed at developing and demonstrating criteria and processes to make the regulation of nuclear technology more "performance based." That is, a philosophy of regulation that is more focused on the outcomes that are to be achieved than on the prescriptive means by which those outcomes are attained. The concept of performance based regulation is widely viewed as having the potential to significantly reduce operating and capital costs for both

existing and new nuclear facilities. Under this regulatory philosophy, licensees would be granted broad latitude to define the means by which required outcomes are attained. This added flexibility could potentially result in significant cost savings. The purpose of this research is to help pave the way for performance based regulation by defining criteria, processes, methods, and models that can ensure effective regulation and demonstrating these ideas in trial applications for selected DOE facilities.

Proliferation-Resistant, Low Cost Thorium Dioxide-Uranium Dioxide Fuels for Light Water Reactors

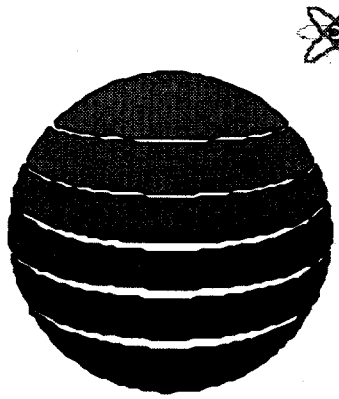
This research project explores new fuel configurations, consisting of mixed thorium dioxide and uranium dioxide, for light water reactors. The goal of the work is to develop light water reactor fuel that is less expensive than the current UO_2 fuel, that allows longer refueling cycles and higher sustainable plant capacity factors, that is highly resistant to proliferation of weapons materials, that results in a more stable, and insoluble waste form, and that generates reduced volumes of high level waste.

Detailed descriptions of each of the research projects, and a discussion of the status of each project are provided in the appendices that follow.

Advanced Reactor Technology

Modular Pebble Bed Reactor Project

Massachusetts Institute of Technology
Idaho National Engineering & Environmental Laboratory



First Annual Report

July 1, 1999

Submitted by

Andrew C. Kadak - MIT
Ronald G. Ballinger - MIT
John M. Ryskamp - INEEL

Table of Contents

Modular Pebble Bed Reactor Project

Andrew C. Kadak—MIT
Ronald G. Ballinger—MIT
John M. Ryskamp—INEEL

June 22, 1999

1. ABSTRACT

The Advanced Reactor Technology Project has as its research objective the establishment of a fundamental conceptual design for a Modular Pebble Bed Reactor (MPBR) electric generating station that is cooled by helium gas. Based on preliminary research into future nuclear energy options, the Massachusetts Institute of Technology (MIT) student work, which began in January of 1998, concluded that this technology provided the best opportunity to satisfy the safety, economic, proliferation, and waste disposal concerns that face all nuclear electric generating technologies. The areas of research for this project are aimed at addressing some of these fundamental concerns to determine whether the small 110 MWe modular gas-cooled pebble bed plant can become the next generation of nuclear technology for worldwide deployment.

In his most recent speech on the subject at the American Nuclear Society meeting in Boston on June 7, 1999, William Magwood, Director of Nuclear Energy Programs at the Department of Energy, called for the development of a "Generation 4" nuclear energy plant. The criteria he sited were small, modular, quick to build, naturally safe, proliferation resistant, and competitive nuclear plants. The MPBR is exactly that type of plant. Our intention is to develop the technical and economic basis of this plant to a point where a serious proposal can be made to build a research power plant at the INEEL to test the various new concepts in fuels, controls, balance of plant equipment, natural safety (no meltdown), and process heat applications.

Although gas reactors have been tried in the past with limited success, the innovations of modularity, factory manufacture, onsite assembly, and integrated state-of-the-art control systems coupled with improved fuel design and a pebble bed core make this design potentially very attractive from an economic perspective. The key issue for successful deployment, after having demonstrated the unique safety advantages, is whether this plant can compete with natural gas fossil units. Preliminary cost comparisons performed by MIT and supported by ESKOM, the South Africa utility, indicate that the modular pebble bed design is competitive with natural gas and, in ESKOM's analysis, the numbers show lower costs. Should these conclusions continue to hold as a result of the research work underway on this project, the modular pebble bed design could revolutionize how electricity is supplied to a growing and developing world population in an environmentally acceptable manner. The INEEL/MIT team is investigating six specific areas of research.

Fuel Particle Design: MIT accomplished the following tasks: (1) a complete review of the current state of technology for coated particle fabrication and performance; (2) a review of the current state of the technology for coated particle failure modeling and; (3) based on the evaluation of the in-reactor test data for coated particle failure, a plan has been developed to improve the current failure model. Reviews were conducted of past INEEL tests of the General Atomics microsphere fuel compacts. The FUEL code was received, tested, and evaluated to determine its suitability for the work being planned. The INEEL used the ABAQUS stress analysis code to compute fuel particle stress distributions in particles with a variety of properties and defects, such as cracks in the outer and inner pyrolytic carbon layers and debonding between layers.

Based on the analysis conducted thus far, the characteristics of the next generation failure model have been identified. It has been decided that a completely new fuel performance model will be developed. The primary motivation for this decision is the recognition of the importance of the incorporation of anisotropic effects in physical and mechanical properties for the pyrocarbon layers. The model will consider the thermal, chemical, and mechanical evolution of the fuel with time and will assess fuel reliability. The reliability model will be based on a probabilistic fracture mechanics approach for the initiation and propagation of layer failure. Additionally, the model will account for non-symmetric loading, which may result from localized debonding and/or layer cracking, as well as non-spherical layers in the as-fabricated

kernels. Improved properties distributions, which more accurately represent bounds for as-fabricated layer dimensions, will also be incorporated. Lastly, finite element modeling of the details of crack initiation, propagation and SiC/pyrocarbon layer failure will be used to validate the methodology used in new fuel model. The INEEL and MIT teams will jointly develop the new model.

Atomistic Basis for Radiation Damage of Fuel Materials: The MIT effort on this task centered on the understanding of radiation damage in graphite. The graphite will be used as a test case for atomistic modeling of the effects of radiation damage of materials because there is a great deal of data available. The objective is to develop an atomistic model that can be used to predict behavior of fuel materials over time in the reactor environment. This information will be used as input to the fuel behavior model of the previous task. To date simple models have been developed for the simulation of long term behavior.

Spent Fuel Disposal Characteristics: This MIT task evaluated the loading requirements of pebble bed fuel in the Yucca Mountain repository. Since the pebble bed reactor is a low power density reactor, the volume of fuel is 10 times larger than that of equivalent light water reactor spent fuel. A Master's Thesis has been completed, which concludes that the actual space required in a the Yucca Mountain repository is a factor of 7 less than light water reactor fuel for the equivalent amount of electrical generation due to the higher burnup and lower heat generation rates of the pebble bed fuel. The graphite waste form for direct disposal was evaluated and found to be an excellent material with low corrosion, low leach rates, and chemical stability, which greatly simplifies overpack requirements and disposal of this fuel form. An experiment was also conducted to determine the attributes of the silicon carbide to chemically retard distribution of significant radioisotopes of the waste. The experiment indicated that silicon carbide could be a good material for fission product retardation.

Methods to Enhance Nonproliferation: A bachelor's thesis was completed on this topic to assess the nature of weapons material and the potential to produce such material from the pebble bed fuel cycle. The design of the on-line fuel handling system was reviewed to assess the opportunities for diversion. International Atomic Energy Agency (IAEA) safeguards requirements and systems were reviewed to determine whether these plants pose a proliferation risk. We concluded that with the high burnup targets established for the fuel (> 90,000 MWd/Mt), the material was essentially useless for weapons production. The ability to divert fuel from the system at lower burnups was also assessed in light of IAEA safeguards requirements. Over 750,000 fuel spheres would need to be diverted at the best burnup time for a UO2 fuel cycle, which could easily be detected. Thus, we concluded that pebble bed reactors provide better proliferation resistance than light water reactors and require no reprocessing. Should additional proliferation resistance be desired, these plants could and have operated on thorium fuel cycles.

Core Neutronics Modeling: The INEEL identified existing codes that can be used to model the neutronics of the MPBR. The COMBINE code was used to generate cross sections and neutron spectra for a standard (MIT) fuel pebble design and three different fuel particle types (HRB-21, NPR, and HFR-K5). Models of the three types of particles were incorporated into the pebble models. A pebble temperature distribution was computed using a two-region, one-dimensional spherical conduction-convection model and coolant temperatures based on ESKOM reactor design data. INEEL constructed an EXCEL spreadsheet that generates number densities, pebble temperatures, and other data required by COMBINE. The spreadsheet allows the user to vary enrichments, layer thickness, and other reference data in order to generate variations on the reference COMBINE model. The INEEL also developed a new core physics calculation method to find a self-consistent solution for the neutron flux, the burnup distribution in the core, and the fissile fuel concentration in the core. For the simple case of once-through fueling, we have demonstrated a method for a one-dimensional, one-neutron-group model. We must now extend this new methodology to two dimensions and few neutron energy groups.

The MIT effort has focused on using MCNP to model the core. An MCNP model of the startup core has been prepared and is being used to generate preliminary flux and power distributions, and reactivity worths of control and shutoff absorbers, and to assist in reactor design. The MCNP model of the core, which will be used to benchmark predictions by VSOP and the INEEL diffusion code, will be validated against PROTEUS experimental data. The MCNP code will be linked to ORIGEN and a pebble-dynamics code via MOCUP to provide an independent modeling capability of the equilibrium MPBR core. A Core Neutronics Strategy was developed to outline an integrated approach to model the steady state and transient

neutronic behavior of the core and its integration with the thermal hydraulics model of the reactor plant. This document integrates the work of MIT and INEEL.

Thermal Hydraulics Modeling: A reference design of the complete MPBR plant has been established on a major component level. A steady state thermal hydraulics model has been produced with key parameters established for the conditions at all major components. Development of an integrated plant model to allow for transient analysis on a more sophisticated level will be developed. The INEEL performed a literature search on MPBR thermal hydraulics, installed the RELAP5/ATHENA code onto an INEEL HP712 workstation, calculated the pressure drop across the pebble bed, and set up a RELAP5/ATHENA model of the pebble bed core. Using this model, fuel pebble and particle kernel temperatures were computed.

International and University Collaborations: There is a great deal of interest and work being performed on an international level on advanced gas reactor technologies. Agreements for collaboration are either already signed or are being developed with Russia (Kurchatov Institute), China (INET), Germany (Jeulich), South Africa (ESKOM), Netherlands (Petten), and Japan (JAERI). In addition, student design projects supporting the pebble bed design have been conducted at Ohio State, the University of Cincinnati, and the University of Tennessee, with more university collaborations planned in the future.

Web Site Development: The INEEL has developed a web site to enhance communication among project participants. The site contains a home page, a participant contact list, a status page, schedule information, archival documents, a topic search capability, and a discussion page. The site is open to the public. **http://id.inel.gov/Pebble_Bed**

2. General Information

Requested Information	Your Response
FULL PROJECT TITLE	**Advanced Reactor Technology—Modular Pebble Bed Reactor Project**
PROJECT NUMBER	**B211**
PRINCIPAL INVESTIGATOR'S NAME	**John M. Ryskamp Andrew C. Kadak Ronald G. Ballinger**
INSTITUTION	**INEEL MIT MIT**
MAILING ADDRESS	**P. O. Box 1625 Idaho Falls, ID 83415-3885** **77 Mass Ave. Room 24-207 Cambridge, MA 02139-4307**
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PHONE NUMBER	**208-526-7643 617-253-0166 617-253-5118**
COLLABORATOR'S NAMES AND INSTITUTIONS	
FUNDING DATE	**December 1998**
REPORT PREPARATION DATE	**July 1, 1999**

3. Project Description

The Advanced Reactor Technology Project has as its research objective the establishment of a fundamental conceptual design for a Modular Pebble Bed Reactor (MPBR) electric generating station that is cooled by helium gas. Based on preliminary research into future nuclear energy options, the Massachusetts Institute of Technology (MIT) student work, which began in January of 1998, concluded that this technology provided the best opportunity to satisfy the safety, economic, proliferation, and waste disposal concerns that face all nuclear electric generating technologies. The areas of research for this project are aimed at addressing some of these fundamental concerns to determine whether the small 110 Mwe modular gas cooled pebble bed plant can become the next generation of nuclear technology for worldwide deployment.

In his most recent speech on the subject at the American Nuclear Society meeting in Boston on June 7, 1999, William Magwood, Director of Nuclear Energy Programs at the Department of Energy called for the development of "Generation 4" nuclear energy plants. The criteria he sited were small, modular, quick to build, naturally safe, proliferation resistant and competitive nuclear plants. The MPBR is exactly that type of plant. It is the intention of this project to develop the technical and economic basis of this plant to a point where a serious proposal can be made to build a research power plant at INEEL to test the various new concepts in fuels, controls, balance of plant equipment, ultimate safety and process heat applications. The proposal would be lead by a university collaborative supported by national laboratories and selected companies to assist in detailed design.

Although gas reactors have been tried in the past with limited success, the innovations of modularity, factory manufacture, onsite assembly and integrated state of the art control systems coupled with improved fuel design and a pebble bed core make this design potentially very attractive from an economic perspective. The key issue for successful deployment, after having demonstrated the unique safety advantages, is whether this plant can compete with natural gas fossil units. Initial preliminary cost comparisons performed by MIT and supported by ESKOM, the South Africa utility, indicate that the modular pebble bed design is competitive with natural gas and, in ESKOM's analysis, the numbers show lower costs. Should these conclusions continue to hold as a result of the research work underway on this project, the modular pebble bed design could revolutionize how electricity is supplied to a growing and developing world population in an environmentally acceptable manner.

4. Specific Research Areas

The investigation targets the following specific areas of research:

Improved Fuel Particle Design

The reason this area is so important is that fuel performance is key to the safety of the pebble bed plant. Past US made fuel has had a spotty record of performance. The task being researched is to understand the fundamental scientific bases for microsphere fuel behavior and to develop a model to predict fuel performance for the high burnup target of over 90,000 Mwd/Mt.

Spent Fuel Disposal Characteristics

Without a clear understanding of what is going to be done with the spent fuel, no new advanced design plant will be built in this country. Pebble bed reactors rely on low power density for its natural safety. As a result, the volume of fuel, despite its much higher burnup than light water reactors, is significantly higher, which would have a negative impact on its use. This area of research explored the net amount of spent fuel that could be stored in Yucca Mountain as a function of electricity generated. Also addressed in this topic is the question of waste form. Is the graphite pebble bed a superior waste form for disposal, not requiring special overpacks to assure containment for 10,000 years? Lastly, a special experiment was conducted to determine whether the silicon carbide coating of the microspheres provides any chemical retardation of the movement of key fission products.

Methods to Enhance Nonproliferation

Non-proliferation of any nuclear technology is key to its deployment. A study was conducted focusing on the proliferation risks of the pebble bed fuel cycle and means to prevent and detect diversion of spent and fresh fuel. Even though the plant uses relatively low 8% enriched uranium and anticipates high burnups up to and exceeding 90,000 MWd/Mt, an assessment was made of the risks of premature removal of pebbles and the bomb making potential at various burnup levels.

Thermal Hydraulics Modeling

Since this project is going to develop a conceptual design of the complete modular pebble bed plant, it was important to begin establishing the conceptual design of the balance of the plant using the indirect helium-to-helium heat exchanger design. The task is to develop a model to analyze steady state and transient behavior of the integrated plant. This then will be used as a reference for safety analysis and the design of an integrated state of the art control system.

Core Neutronics Modeling

Modeling of the reactor core, which utilizes on-line refueling with continued movement of pebbles, is a challenging task. It is particularly important in licensing since there is no ability to directly measure incore power distributions. As such, the importance of a validated core neutronics model is essential. At present an MCNP approach is being used to develop a core model which will be validated with critical experiments and ultimately used to validate more general diffusion theory codes such as VSOP which are traditionally used in such analysis. It is also planned to investigate alternative fuels such as thorium cycles and no-online refueling options.

These areas were selected on the basis of their importance to the ultimate success of the project. As the project develops, additional topics will be added to address safety, economics and the development of an integrated control system. These topics will be added as resources are available and sufficient technical bases developed to allow for work in these areas.

5. Project Accomplishments

5.1. Fuel Particle Design

During this reporting period the following has been accomplished: (1) A complete review of the current state of technology for coated particle fabrication and performance; (2) a review of the current state of the technology for coated particle failure modeling and; (3) based on the evaluation of the in-reactor test data for coated particle failure, a plan has been developed to improve the current failure model. Reviews were conducted of past INEEL tests of the General Atomics microsphere fuel compacts. The FUEL code was received, tested, and evaluated to determine its suitability for the work being planned.

The INEEL used the ABAQUS stress analysis program to compute stress distributions in fuel particles having defects such as shrinkage cracks in the inner and outer pyrocarbon layers and debonding between the inner pyrocarbon and silicon carbide layers. The purpose for these computations was to determine whether these defects contribute significantly to failure of the particles. Results of these analyses show that significant concentrations of tensile stress develop around these defects early in the irradiation of the particle, especially around the tip of a shrinkage crack. To gain further insight as to whether these stress concentrations lead to particle failure, an ABAQUS model was developed to calculate stress intensities around the tip of a radial shrinkage crack in the inner pyrocarbon layer that can be used in a fracture mechanics evaluation of the particle.

Based on the analysis conducted thus far, the characteristics of the next generation failure model have been identified. It has been decided that a completely new fuel performance model will be developed. The primary motivation for this decision is the recognition of the importance of the incorporation of anisotropic effects in physical and mechanical properties for the pyrocarbon layers. The model will consider the thermal, chemical, and mechanical evolution of the fuel with time and will assess fuel reliability. The reliability model will be based on a probabilistic fracture mechanics approach for the initiation and propagation of layer failure. Additionally the model will account for non-symmetric loading, which may result from localized debonding and/or layer cracking as well as non-spherical layers in the as-fabricated kernels. Improved properties distributions, which more accurately represent bounds for as-fabricated layer dimensions, will also be incorporated. Lastly, finite element modeling of the details of crack initiation, propagation and SiC/pyrocarbon layer failure will be used to validate the methodology used in new fuel model. The new model will be jointly developed by the INEEL and MIT teams.

5.2. Fuel Kernel Performance

Figure 1 shows a schematic of a typical TRISO coated particle or "microsphere". The fuel kernel, which may be UO_2 or a Uranium Oxycarbide is surrounded by a number of layers of graphite and at least one layer of silicon carbide or, in advanced fuel kernel designs, another carbide such as zirconium carbide. The inner-most graphite layer is a low density region designed to accommodate changes in fuel kernel dimensions due to swelling, to accommodate fission gas release, and to absorb any radiation damage from fission product recoils. The next three layers, consisting of an inner and outer pyrocarbon layer with a SiC layer in between, constitute the primary barrier to fission product release and acts as the pressure boundary for the system.

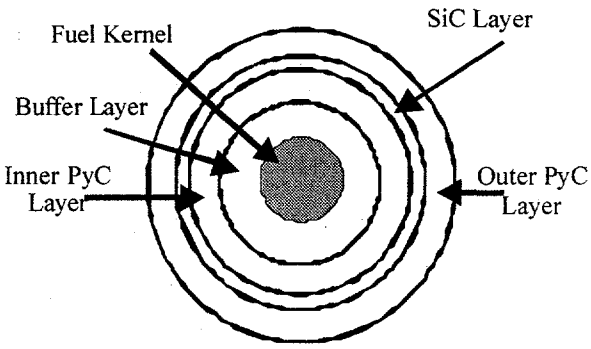

Figure 1. Schematic of typical "TRISO" coated particle.

Figure 2 shows the effect of irradiation on the structure of the microsphere. The fuel kernel has undergone significant microstructural changes, the most obvious of which is the development of fission-gas-induced swelling. The low density buffer layer has undergone a significant amount of densification to the extent that obvious void regions now exist. While these changes are significant, producing gas pressure buildup within the kernel, the most significant changes can occur in the PyC and SiC layers.

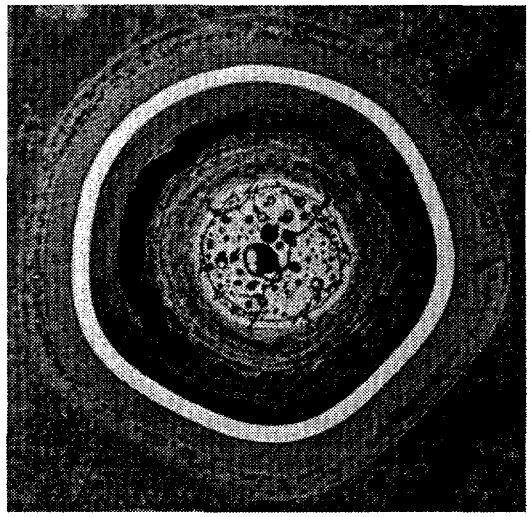

Figure 2. Irradiated fuel microsphere showing effects of exposure

As Figure 2 shows, the inner and outer PyC layers undergo shrinkage and, in some cases, debonding from the SiC layer. Moreover, this shrinkage if it occurs asymmetrically can, and does in the figure illustrated, result in cracking of one or both of the PyC layers. In extreme cases, the SiC layer also cracks. When this occurs, the fuel kernel releases fission products to the primary system environment. While failure of the SiC and PyC layers constitute failure of the pressure boundary, a second release path for specific fission products exists. A few fission products, silver in particular, are highly mobile and diffuse through intact PyC and SiC layers. Release of silver by this mechanism becomes significant at temperatures approaching 1600°C.

A comparison of the idealized geometry shown in Figure 1 with the actual kernel, shown in Figure 2, is instructive. The optimum fuel kernel geometry consists of spherical layers of

uniform thickness with, in the ideal case, identical dimensions. However, the as-fabricated kernels exhibit distributions in layer thickness as well as in degree of sphericity. Moreover, along with distributions in physical dimensions, there will be distributions in physical and mechanical properties for each of the kernel layers. Figure 3 shows a plot of a measured degree of anisotropy of crystal structure for the inner PyC layer for a number of nominally identically fabricated fuel microspheres. The degree of anisotropy is significant.

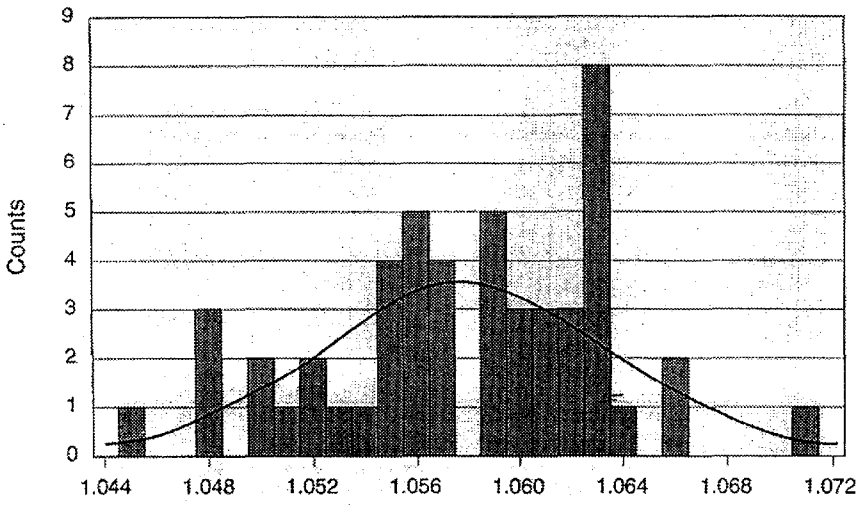

Figure 3. NPR Inner Pyrocarbon layer Anisotropy-Number of Particles Versus BAF. Sample Size: 50 Particles

Lastly, the effect of irradiation damage, among other effects, will be to alter the degree of anisotropy. Modeling these complex processes is a difficult one, indeed. However, an understanding of fuel reliability will require that these effects be modeled adequately.

The fuel kernel can be roughly divided into two loosely coupled regions from a fuel performance standpoint. The fuel itself, consisting of the uranium oxide and/or uranium carbide microsphere, is separated from the pressure boundary of the kernel, the PyC and SiC layers, by the low density graphite layer. This arrangement effectively separates the microsphere into two regions from a fuel performance perspective and serves to decouple mechanical interaction between the fuel kernel itself and the pressure boundary layers. This is unlike the situation with light water reactor fuel in which mechanical interaction between the fuel pellets and the fuel cladding will ultimately occur during the life of the fuel. The primary coupling between the two regions is through fission gas and/or CO/CO_2 related pressure buildup, which results in pressure stresses in the pressure boundary layers, and through any chemical interaction between fission products and/or fuel and the pressure boundary layers. The following phenomena will be important within each region:

Fuel/Buffer Layer Region

1. Fuel Swelling
2. Fission Gas Release
3. Fission Product Migration
4. Chemical Potential
5. Buffer Densification

The modeling of fission fuel swelling and fission gas release in UO_2 based fuel systems is well understood from a phenomenological if not a deterministic point of view. We anticipate that the project will be able to make extensive use of existing models and data in this area as a starting point. However, our fuel differs significantly from UO_2 based systems due to the presence of uranium carbide which is added to the UO_2. The uranium carbide (UC-UC_2)-uranium oxide system will act as a buffer for the system and will control the chemical potential. This process takes advantage of in the design of the fuel. The U/C ratio is adjusted such that gas (CO, CO_2) production is minimized during the expected life of the fuel. The evolution of the U-C system has been studied well enough such that the modeling of this area will be straightforward but complicated.

Pressure Boundary Layer Region

1. Pyrocarbon Properties Evolution
2. Pyrocarbon Swelling, Densification, Creep
3. SiC Properties Evolution
4. Inter-Layer Mechanical Interaction
5. Layer Cracking and/or Debonding
6. Effects of Anisotropy of Initial Properties and Evolution of Anisotropy

The key to prediction of the fuel failure probability will be successful modeling of the pressure boundary layer region. The current model, used in the FUEL code, treats this region as a symmetric problem with isotropic materials properties. With these assumptions, it is possible to develop a closed form solution to the mechanical problem including the effects of swelling and creep in the pyrocarbon layers. Failure is predicted to occur when the fracture stress of the SiC layer is exceeded. Variability in layer properties (dimensions, fracture stress, etc.) are accounted for through distributions on input.

The current treatment of the mechanical problem has proven to be unsatisfactory. The model underpredicts the fuel failure probability. The current treatment does not adequately account for phenomena that play a key role in the failure process. These phenomena include swelling, shrinkage, and cracking and/or debonding of the pyrocarbon and SiC layers as a result of irradiation exposure. Additionally, the assumption that the system is isotropic in properties and their evolution is not correct. This is especially true for the pyrocarbon layers. Figure 3, mentioned earlier, shows the distribution of a measure of anisotropy for the inner pyrocarbon layer for New Production Reactor fuel particles. The degree of anisotropy is significant. Perhaps even more importantly, the degree of anisotropy EVOLVES with irradiation exposure. Thus, the elastic mechanical problem, is anisotropic if we assume isotropic swelling and creep (which we cannot). The evolution of anisotropy with irradiation exposure further complicates the situation.

The situation with the SiC layer, while less complicated, still presents significant challenges. SiC is isotropic in its mechanical properties. However, data from testing of SiC layer strengths shows that there is a significant variation in the fracture strength within a nominally identically fabricated batch. There appears to be a bimodal fracture strength distribution with some fraction of a batch exhibiting very low fracture strength. Current treatment of SiC fracture strength distributions using a Weibull distribution is not adequate for the actual particles.

The development of the model for the pressure boundary region has required that the mechanical problem be reformulated to allow for anisotropic mechanical properties and their evolution during irradiation exposure. As a result of this, a closed form solution to the mechanical problem will not be possible.

The reformulation of the mechanical problem in the pressure boundary region will allow for a more realistic calculation of the layer stress distributions and their evolution with irradiation exposure. This, in combination with initial distributions of layer physical properties, degree of anisotropy, and fracture strengths, will allow for a much improved fuel reliability model.

Another aspect of this project is to develop a model for radiation damage of the fuel particle using fundamental atomistic principles. Early work centered on the understanding of radiation damage in graphite which was used as a test case for the atomistic modeling of the effects of radiation damage of materials. Work is now underway to understand the behavior of the pyrolitic carbon layers. It is quite clear that grain structure will be a very important feature in determining the integrity of the fuel microspheres which means that the control of the manufacturing process will be especially critical to good fuel and the natural safety feature of the plant.

5.3. Spent Fuel Disposal Characteristics

Waste issues effecting the Pebble Bed Reactor were addressed in a recent Master's Thesis [1]. The thesis examined three primary topics: (1) Storage requirements of spent pebble fuel compared with storage requirements of spent fuel from a pressurized water reactor, (2) An analysis of graphite as a waste form, and (3) Interactions of silicon carbide with uranium. Each of these topics are extremely important to understanding the long-term impact should pebble bed reactors be implemented on a wide scale in the near future. The characteristics examined in this report favor the use of pebble bed reactors over light water reactors. The results and conclusions of each of the three main topics are summarized below.

Storage Requirements

The conclusion of the thesis is that it will take less space to store spent pebble fuel than it will take to store spent fuel assemblies from a pressurized water reactor. In fact, the space saving potential is almost 7.5 times as great for the pebble bed reactor based on megawatts of electricity available to the consumer. However, this savings in area comes at a price of more canisters needed. This report assumes that spent pebble fuel will be placed in the same size canister with the same protective material as the spent fuel assemblies of the light water reactors. With this as the case, the spent pebble fuel will require 9.60 times more waste packages than light water reactor spent fuel.

There are significant costs associated with the increase in the required number of waste canisters. The materials used in the manufacturing process of the canister are the most significant cost contributors to the overall design. The exotic materials needed for the protection of spent fuel assemblies are not needed for spent pebble fuel. A licensing process is necessary that credits spent pebble fuel for its already superior packaging in the form of the protective outer layer made of graphite. The expensive canister material adds only 1,000 years to the corrosion resistance of the graphite matrix which will last on the order of 1 billion years. Therefore, the most expensive part of the canister provides virtually no added protection for the spent pebble fuel. Perhaps a less expensive material can be used. Because all proposed canister materials will corrode at rates that are orders of magnitude faster than graphite, it is reasonable to attempt to save cost on materials that add no benefit to the corrosion resistance of the graphite waste form. Table 1 is a summary of the results on storage requirements.

Table 1. Storage Area Requirements for a PBR and PWR at 1000 MW(e)

Issues	PBR	PWR
Fuel elements/waste package	41957	21
Fuel elements discarded/year	1150538	60
Years of operation	30	30
Fuel element storage requirement	34516125	1800
Waste packages needed	823	86
MTHM/fuel element	3.47×10^{-6}	0.45
MTHM/waste package	0.1456	9.45
MTHM/acre	90	90
kW/waste package	1.53	15.53
Waste packages/acre	617	9
Acres needed for storage	1.333	10.056
Storage Requirement (m^2/MW(e))	5.33	40.70

5.4. Graphite as a Waste Form

There is overwhelmingly positive support for graphite as a waste form from all researchers that have specifically looked at this question [2,3]. No contradictions to these conclusions were found in any of the reports that compared graphite to the metal alloys associated with PWR spent fuel. The exceptional durability, thermal conductivity, thermal expansion, and inertness of graphite make it a superb barrier. The graphite already exists on the fuel. The graphite is placed on during the fuel manufacturing process and remains with the fuel throughout the fuel cycle including final disposal. Conveniently, spent pebble fuel already exists in a matrix that is far superior to any that can be reasonably placed around the spent fuel assemblies of light water reactors.

Silicon Carbide and Uranium

For the Yucca Mountain Viability Assessment, four isotopes were identified at potential long term dose vectors: ^{99}Tc, ^{129}I, ^{234}U, and ^{237}Np. Additionally, the role of secondary phase formation at a retaining mechanism is identified as an important issue for acceptance of the repository license. The work covered in this task examined the role of SiC in forming insoluble secondary phases with U(VI). The results clearly indicate that there is less soluble uranium in a system with silicon carbide than a system without silicon carbide (Figure 4). Above pH 4, the amount of uranium present in the aqueous phase is reduced by the presence of SiC. The differences are strongest around pH 7, conditions which are expected in the repository environment. Along with the experimental data, speciation calculations with literature data is presented [4]. The decrease of aqueous uranium is well correlated with the presence of uranium hydroxide species ($UO_2(OH)_x$). This indicates the sorbed U species on SiC are hydroxides. This can be used to evaluate the thermodynamic constants which describe the sorbed species.

The silicon carbide layer of the spent fuel element of a pebble bed reactor will act as a barrier to migration of radionuclides throughout the fuel cycle. Most significantly, it will act as a retention mechanism for millions of years during final disposal in a repository. Further work should be directed to characterize the ability of the silicon carbide to retain other radiologically significant isotopes.

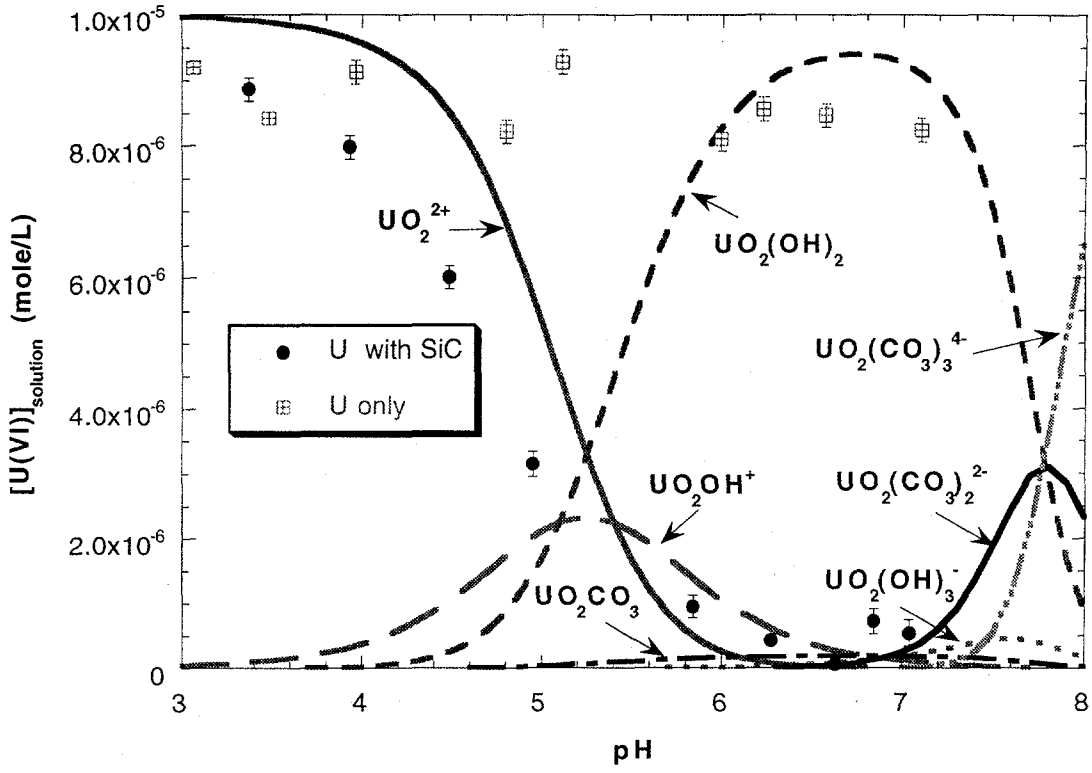

Figure 4. Behavior of Uranium(VI) in the Presence of SiC. The Speciation of Uranium(VI) is also Presented

5.5. Methods to Enhance Nonproliferation

A Bachelor of Science thesis was completed on this topic by Jennifer M. Anderson [5]. Since the MPBR design is dependent on an online refueling system, the proliferation resistance of the plant is an important issue. This thesis addressed the following topics:

1. The nature of weapons materials and the potential to produce such from the pebble bed and light water reactor plants.
2. The design of the fuel handling system and potential points for diversion.
3. IAEA safeguards adequacy to ensure that the number of pebbles required for weapons materials can not be diverted without detection.

Weapons Potential

Shown on Table 2 is the plutonium content of the fuel as a function of burnup starting with an 8% enriched UO_2 core. This table identifies the key isotopes of interest in weapons and proliferation concerns. These calculations were performed by Dr. Thomas Cochrane of the Natural Resources Defense Council, who helped on this project.

Table 2. Plutonium content of MPBR fuel

	Burnup (MWD/kg)										
Isotope	0	10	20	30	40	50	60	70	80	90	94
Pu-238	0.0%	0.02%	0.10%	0.28%	0.60%	1.17%	2.14%	3.66%	5.58%	7.00%	7.28%
Pu-239	100%	91.49%	82.34%	72.96%	63.82%	55.32%	47.77%	41.28%	35.97%	32.19%	31.17%
Pu-240	0.0%	7.91%	15.23%	21.56%	26.48%	29.62%	30.70%	29.59%	26.93%	24.38%	23.67%
Pu-241	0.0%	0.56%	2.13%	4.47%	7.19%	9.80%	11.67%	12.24%	11.53%	10.52%	10.22%
Pu-242	0.0%	0.02%	0.20%	0.73%	1.90%	4.08%	7.73%	13.23%	19.98%	25.98%	27.66%
kg Pu/ton	0.0	0.8036	1.507	2.147	2.753	3.349	3.965	4.640	5.391	6.886	6.296

Shown on Table 3 are the key plutonium characteristics of various weapons grades of plutonium. As can be seen by comparing the two tables, the MPBR high burnup fuel does not even classify as reactor grade weapons material.

Table 3. Isotopic mixes of various plutonium grades

	Isotope				
Grade	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
Super-grade	-	98.0%	2.0%	-	-
Weapons-grade	0.012%	93.8%	5.8%	0.35%	0.022%
Reactor-grade	1.3%	60.3%	24.3%	9.1%	5.0%

Table 4 which is a predetonation capability comparison, indicates that the spontaneous fission rate for the MPBR fuel is so high that there is less than a 50% chance of a weapon even reaching 4% of its design yield.

Table 4. Predetonation Probabilities

Probability of exceeding x% of design yield	Super Grade	Weapons Grade	MPBR
4%	98.51%	95.70%	48.70%
5%	97.49%	92.81%	29.52%
6%	96.55%	90.19%	18.49%
8%	94.82%	85.55%	7.79%
10%	93.27%	81.49%	3.52%
15%	89.86%	73.07%	0.59%
20%	86.93%	66.29%	0.12%
25%	84.32%	60.61%	0.03%
50%	74.08%	41.44%	0.00%
99%	60.53%	22.90%	0.00%

It is quite clear from this review and analysis that persons or nations wishing to make a weapon from pebble bed spent fuel will look elsewhere for the source of their material. The largest threat to proliferation is early removal of the balls after approximately 12,000 Mwd/Mt burnup, which is addressed in the next sections.

Comparison with LWR uranium oxide fuel cycles indicates that the pebble bed fuel is superior in proliferation resistance largely due to higher burnups. Should additional proliferation resistance be required, the MPBR could be operated on a thorium cycle which may be preferable.

Design of the Fuel Handling System

The fuel handling system of the MPBR consists of a series of tubes as depicted in Figure 5. These tubes channel the pebbles from the bottom of the core into the top, with one fresh fuel entry point and one discharge point to the spent fuel storage canisters. A review of the ESKOM and AVR (German research reactor) was made to identify possible diversion paths with the following conclusions. The most likely diversion points are at the locations of the largest quantity of fuel – fresh fuel (not interesting due to low enrichment), scrap pile (small amount of mixed burnups) and the spent fuel canisters (burnups very high and not interesting weapons material). Are there physical barriers to prevent diversion and do monitoring methods exist to detect any diversion that may take place? In general, the fuel handling system is a contained system that would require penetration to access the quantity of pebbles required for weapons production.

IAEA Safeguards and Diversion Quantities Required for Weapons

Those nations that have signed the nonproliferation treaty have pledged not to develop nuclear weapons if they do not already possess them. These nations are subject to International Atomic Energy Agency (IAEA) inspections whose purpose is the "timely detection of diversion of significant amounts of nuclear material". IAEA uses mechanical, monitoring, and inspections to detect such diversions. The IAEA processes were reviewed and it is concluded that based on the timely detection requirement using modern detection techniques, pebble bed plants can be safely deployed without fear of proliferation.

To make this point even stronger, the IAEA significant amounts of materials for the following categories are summarized in Table 5 below.

Table 5. Number of Pebbles Required for Weapons

Fuel Burnup	Uranium 235[1]	Plutonium[2]	Percent of Core
Fresh	134,000	-	37%
10,000 Mwd./Mt		1,422,142	About 3.5 cores
20,000 Mwd/Mt[3]		758,365	About 2 cores
90,000 Mwd/Mt[4]		165,968	About _ core

1. 75 kg of Uranium 235
2. 8 kg of Pu
3. Best time for isotopic weapons mixture.
4. Not weapons capable due to high heat generation rate

The conclusion of this review is that detection of fuel diversion is likely due to the huge number of fuel balls that need to be diverted for even the minimal IAEA requirements regardless of the quality of the weapons material.

5.6. Thermal Hydraulics Modeling

During this reporting period the standard "reference" design for the MPBR system has been developed. This reference design has been documented and will be reported in two topical reports. [6,7] Figure 6 shows the schematic of the proposed reference design. The basic design of the plant contains a primary system cooled by the downward flow of helium with a core exit temperature of 850°C going to an intermediate heat exchanger and returning to the inlet at a temperature of 445°C The secondary side of the plant also operates on a helium Brayton cycle in which the helium from the intermediate heat exchanger goes to the high and low pressure turbines powering a 110 Mwe generator.

A detailed steady state thermohydraulic model has been produced. Shown on Figure 7 (at end of report) is a block diagram schematic of key pressures and temperatures at various parts in the plant. A net bus bar efficiency of 45 % is predicted for this type of cycle.

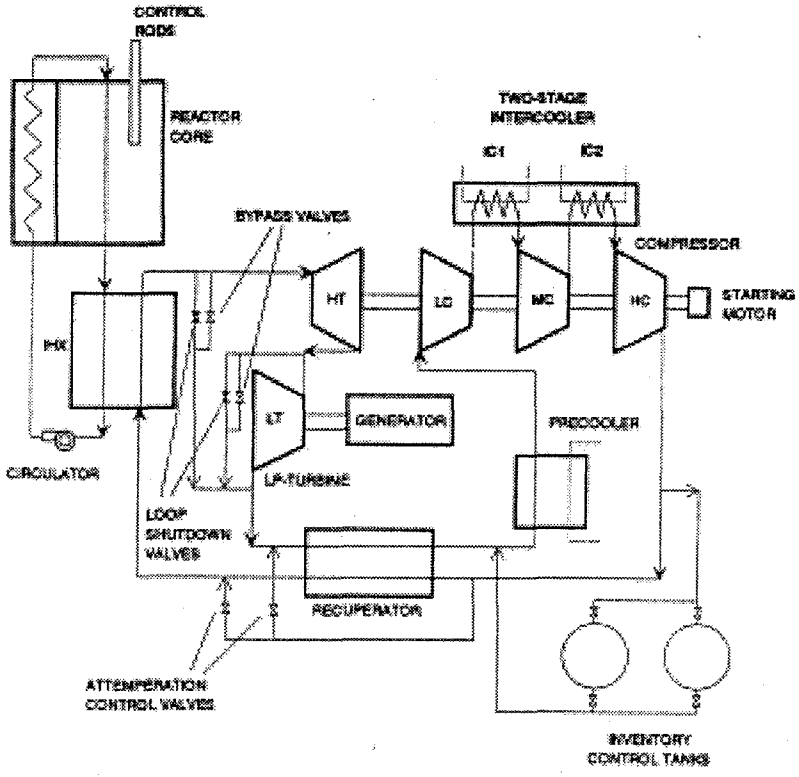

Figure 6 Schematic of Reference Design

As part of thermal hydraulics safety analysis on the pebble bed reactor concept, RELAP5/ANETHA code was used to model the pebble bed core. The model consists of pebble bed, time-dependent volumes and junctions. The bed was divided in five hydrodynamic volumes. Each volume includes a representative pebble and particle at each divided location. The bed can be divided into more hydrodynamic volumes for the detailed calculations. The purpose of these calculations is to determine the temperature distributions within particles and pebbles in the pebble bed to make certain that the hot temperature does not exceed the design temperature.

Based upon a 250 MW thermal power, normalized power distribution, the design flow rate of helium, thermal properties used by the new production reactor, and other input information, the hottest temperature in the particle is 925°C (vs. 1400°C design temperature). The hottest particle is located in the lower hydrodynamic volume because of the downward flow configuration. In this model, the representative particle was coupled with a pebble that is cooled by the helium gas flow. Based on the information on the surrounding hydrodynamics, the pebble temperature was calculated, and then the pebble temperature was used as a boundary temperature to compute the particle kernel temperature by heat conduction.

The near future plan is to perform transient analyses of the loss-of-coolant accident (LOCA). This study will include the entire components such as a reactor core, turbines, compressors, precooler, recuperator, valves, pipings, and helium inventory tank. The safety analysis from this task will demonstrate the effects of the interaction between the reactor core and its energy conversion system.

Lastly system volumes were calculated for the entire plant to be used in future plant modeling.

5.7. Core Neutronics Modeling

The overall strategy for the core neutronics modeling of the MPBR core is based on the complementary use of both diffusion theory and Monte Carlo physics codes. INEEL is developing a core physics and fuel management code system consisting of the COMBINE transport code, the DIFF3D three dimensional diffusion code and the FUPAR pebble flow code. The effort at MIT is focusing on the development of a detailed MCNP model of the reactor, as well as the investigation of pebble bed flow dynamics using other Monte Carlo simulation techniques. The German VSOP ('94) code is also being acquired for use in the project.

The INEEL has developed a new core physics calculation method to find a self-consistent solution for the neutron flux, the burnup distribution and the fissile fuel concentration in the core. The method was successfully demonstrated for a one-dimensional, one-group model with once-through fueling. This methodology is being extended to two dimensions and few neutron-energy groups. The effort at MIT is focusing on the development of a detailed MCNP model of the reactor, the validation of the physics codes, as well as on the investigation of pebble packing and flow dynamics using advanced simulation techniques.

The reactor core consists of a 10 meter high, 3.5 meter diameter cylinder containing over 350,000 fuel pebbles which are continuously circulated as shown in Figure 8. The proposed design contains a central core of graphite (non fuel pebbles to enhance control rod worth. Control rods are located outside the pebble bed in the graphite reflector. Fuel balls are recirculated anywhere from 10 to 15 times to achieve the high target burnups.

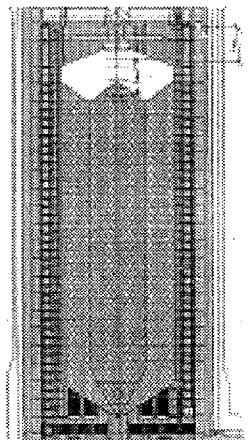

Figure 8 Pebble Bed Reactor Schematic

Two pebble-bed reactors have been previously constructed and successfully operated in Germany (AVR and THTR), and a third reactor is currently under construction in China (HTR-10). While these provide a substantial knowledge base, an independent safety assessment is required in order to license such a reactor in the US. The objective of this work package is to perform physics calculations in support of the nuclear design and analysis of the MPBR, and to provide a licensing basis in the physics and safety areas for US application.

The unique physics challenges posed by this task is shown in Figure 9 . This figure illustrates the flow of pebble through the core. Shown are experimental results form the German program which indicates the relative speed of the pebbles traveling through the bed at different radial locations. Each line represents the same time step which needs to be viewed from the top of the figure. This indicates significantly different rates of movement depending on radial distance from the center. It is obvious tracking each pebble is not possible but some statistical equilibrium averaging technique will be required to simulate this movement.

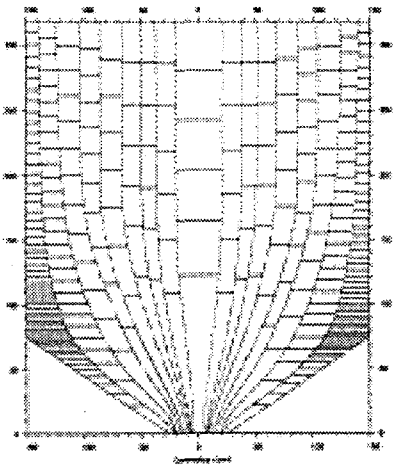

Figure 9 Movement of Pebbles in Core

An important feature of the neutronic modeling of this core is the coupling of the neutron flux and fuel depletion calculations with a dynamic simulation of fuel pebble movement. This is required for an accurate prediction of the power distribution in the core, which will be used in safety analyses that are planned later in the project. Figure 9 illustrates the different pebble flow zones in the AVR core. Due to the inherent safety of this reactor design, the precise knowledge of individual fuel element trajectories is not required. Nevertheless, the reactor physics analysis will include an investigation of the pebble flow dynamics and its effect on the core power distribution.

In order to represent the reactor core, a model was developed which assumed 10 axial and five radial zones. Each zone was assumed to have the same average fuel or graphite material. Figure 10 shows the graphic representation for analysis.

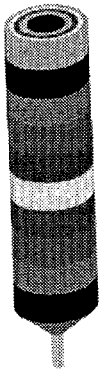

Figure 10 Modeling of Pebble Bed Core

An MCNP model of the startup core has been prepared and is being used to generate preliminary flux and power distributions, reactivity worths of control and shutoff absorbers, and to assist in reactor design. The MCNP model of the core, which will be used to benchmark predictions by VSOP and the INEEL diffusion code, will be validated against PROTEUS experimental data. The MCNP code will be linked to ORIGEN and a pebble-dynamics code via MOCUP to provide an independent modeling capability of the equilibrium MPBR core.

Preliminary startup core power and flux distributions are shown on Figures 11-13 at end of report.

Lastly, a Core Neutronics Strategy [8] was developed to outline an integrated approach to model the steady state and transient neutronic behavior of the core and its integration with the thermal hydraulics model of the reactor plant. This document integrates the work of MIT and INEEL.

5.8. Other Activities

University Collaborations

As a result of lectures given at several universities, three supporting projects have been initiated at these universities. Ohio State is developing a conceptual monitoring system for the reactor leading to an integrated control system. The University of Cincinnati is developing an online burnup meter for the fuel. This is a critical component since burnup limits will be used as part of the safety case for the reactor. The University of Tennessee has reviewed options relative to reducing the size of the reactor vessel to make transportation as a single unit possible. The status of these projects and reports were presented at the June ANS meeting in Boston at a special gas reactor session entitled "The Comeback of Gas Reactors".

International Collaborations

MIT has reached preliminary cooperative agreements have been reached with Russia, Germany, and South Africa regarding cooperation on this design effort. Agreements with China are also being sought. China will have the next pebble bed in operation in 2000 when their 10 MWth research reactor will start up. Visits have been made to Russia (Kurchatov, OKBM, and other technical institutes), Germany (Jeulich) and South Africa (ESKOM) to learn about their past and present activities and to hold technical discussions on ongoing research. A recent trip to China established strong relationships for future cooperative work and also provided an opportunity to view their fuel fabrication facility, test laboratories, and the HTR-10 under construction (the reactor vessel and steam generators are already installed awaiting installation of the graphite reflector).

At the Boston American Nuclear Society meeting in June 1999, international delegates from these countries attended a special session at MIT to have detailed discussions of common research interests and future collaborations.

Presentations on the Pebble Bed Reactor

Presentations on the Modular Pebble Bed Reactor have been made to the following audiences during the term of this contract:

1 Ohio State
2 MIT Independent Activities Lecture

3 University of Cincinnati
4 University of Tennessee
5 Department of Energy
6 American Power Conference
7 Virginia Section of the American Nuclear Society
8 Chicago Section of the American Nuclear Society
9 INET in China
10 Juelich, Germany
11 Kurchatov, Russia
12 Council for Nuclear Safety, South Africa
13 Harvard Kennedy School, Managing the Atom Lecture Series
14 Bryon Nuclear Plant, ANS plant branch
15 ANS November 1998 Meeting
16 "Comeback of Gas Reactors Sessions" on June 9, 1999 at ANS Meeting

Proposal Submitted

A proposal was submitted under the Nuclear Energy Research Institute (NERI) grant solicitation to conduct fuel irradiations at the Advanced Test Reactor (ATR) of Germany made fuel, to compare it against the model predictions and past General Atomic National Production Reactor fuel tests, which were not successful. This proposal requested $ 3 million for a three year effort but was not selected. We will be reviewing the comments of the reviewers for the purpose of resubmitting for this year's solicitation. It has been learned that the Navy was supportive of this work.

6. Path Forward

Fuel Performance Model Development

It was initially felt that this task would build upon already existing model development efforts, codified in the FUEL code, to produce an adequate model for fuel microsphere performance. However, a more detailed analysis of the nature of the kernel failure process, performed during this reporting period, has shown that this is not the case. Indeed, the fundamental failure process is not represented by the model in the FUEL code. Failure due to simple over pressurization of the SiC boundary, exceeding the fracture stress is extremely unlikely. Analysis of the failure process shows that a slow degradation of the PyC layers due to shrinkage, swelling, and/or creep, results in cracking, debonding, or both, of the PyC layers. This process, in combination with initial non-ideal layer shapes and dimensions, results in local stresses that cause cracking of one or more of these layers. Moreover, the evolution of physical and mechanical properties of the PyC layers will play a critical role in the process.

During the next year of the project the focus will be on the development of a new fuel performance model. The process will be divided into the following major tasks: (1) Identification of Model Requirements, (2) Deterministic Model Development, (3) Fracture Model Development, (4) Mechanical and Fracture Model Verification-Finite Element Analysis, (5) Monte Carlo Model Development.

Identification of Model Requirements

While a listing of the phenomena that we expect will need to be modeled is a straightforward task, it will be important that a clear definition of the requirements

for the model be established. Once this has been established, an assessment of the current state of the science for individual phenomena will need to be established. These two tasks will be accomplished through interaction with organizations with knowledge and data as well as with individual experts. This interaction will be initiated through an informal workshop. The participants will include representatives from MIT, INEEL, General Atomics, as well as others that will be identified by the INEEL/MIT team.

Deterministic Model Development

Once the model requirements have been established, the development of the deterministic model will begin. As has been discussed above, the performance model can be divided into a model for the mechanical behavior of the PyC/SiC layers and a model for the performance of the fuel kernel. We anticipate that this will be the development path that will be followed. The mechanical problem will begin with a reformulation of the mechanics of layer performance to include the effects of anisotropy of properties and performance. The reformulated problem will then be benchmarked using finite element analysis.

The fuel thermal and chemistry model will be developed in parallel and then integrated with the mechanical model.

Fracture Model Development

The original fracture model in the FUEL code assumed that failure of the SiC layer occurred due to the hoop stress exceeding the fracture stress. Subsequent analysis of the effect of stress concentration at the PyC/SiC interface due to cracking and/or debonding of the interface has shown that local stresses can be significantly higher than the hoop stress. Additionally, the development of a radial crack in one of the PyC layers without local debonding of the interface will result in the formation of a partially cracked, composite (PyC & SiC) material. For this situation, a fracture mechanics approach to failure of the SiC layer is more appropriate. Additionally, due to the distributions of initial dimensions and microstructure as well as anisotropy of mechanical properties, the fracture problem becomes one where probabilistic fracture mechanics analysis may yield better results. The fracture model for the PyC/SiC layers will be developed using this approach and treating the system as being linearly elastic. For the linear elastic case, superposition of solutions for uniform and non-uniform stress fields will be possible.

Fracture Model Verification

Finite Element analysis will be used to verify stress field and fracture calculations. The finite element analysis will be conducted primarily at INEEL.

Monte Carlo Model Development

The overall model, developed as described above, must be incorporated within a Monte Carlo analysis. During this next year, focus will be on development of the components of the model that will be then used in the Monte Carlo analysis.

Spent Fuel Disposal Characteristics

This task is essentially complete. More work is necessary on the criticality of the spheres in the repository and the corrosion, leachability, and chemical behavior of

irradiated graphite spheres in the repository environment. To conduct this task will require additional resources.

Methods to Enhance Non-Proliferation

This task is essentially complete since the basic issues regarding the pebble bed weapons grade materials and diversion potential has been addressed.

Thermal Hydraulic Modeling

The mission for the thermal hydraulics group is to develop a working balance of plant model incorporating the intermediate heat exchanger with a multishaft turbine compressor system. This model will then be integrated with the reactor plant to create an integrated functional overall thermal hydraulics model of the complete plant. This model will be used to develop a state of the art control system to allow automatic startups, load following and step power changes without operation action.

Core Neutronics Modeling

The overall strategy for the neutronics modeling of the MPBR core is based on the complementary use of diffusion-theory and Monte-Carlo physics codes. INEEL is developing a core physics and fuel management code system consisting of the COMBINE transport code, the DIF3D three-dimensional diffusion code and the FUPAR pebble flow code. The effort at MIT is focusing on the development of a detailed MCNP model of the reactor, as well as the investigation of pebble-bed flow dynamics using other Monte Carlo simulation techniques.

The German VSOP code is intended for addressing fuel-cycle issues and to serve as an initial benchmark for the INEEL code development effort. VSOP will be utilized for production of neutron-diffusion calculations during the initial phases of the project, while the INEEL code system is under development. Although the neutronic design and analysis of the MPBR could be performed in its entirety with VSOP, the development of physics codes in-house is required (a) to produce reliable results independently from those generated with German methods and data and (b) to reestablish US expertise in the nuclear design of pebble-bed reactors.

Such expertise is currently centered in Germany, the Netherlands and Russia. China is also rapidly acquiring this technology through its construction of the HTR-10 prototype reactor [9]. Both the Netherlands and Russia have developed their own analysis tools, although MCNP is retained as a benchmarking standard. In the USA, research and development in this area was carried out at ORNL until the late 1970s but, due to rapid changes in computer technology, the previously developed codes are now obsolete and the associated knowledge base is largely inaccessible [15]. Therefore, the development of physics codes capable of modeling pebble-bed reactors must be included in any long-term strategic planning for the exploitation of such technology in the USA.

DIF3D/COMBINE Analysis

The INEEL physics effort will initially concentrate on the development of a code system based on COMBINE and DIF3D. The stationary distribution of fuel elements in the core will be determined using a simple incompressible-flow model, which will be coupled to the burnup and fuel-cycle analysis capability of the REBUS-3 code system. Once operational, this code will be validated against VSOP and MCNP calculations, and replace VSOP for production calculations.

VSOP Analysis

The VSOP code (Version '94) will be used at MIT to calculate safety-related temperature reactivity coefficients, to perform fuel management studies and to investigate alternative fuel cycles. The fuel management studies will initially generate the equilibrium burnup distribution needed for modeling the core with MCNP. However, for code validation purposes, all subsequent MCNP analysis will be based on burnups calculated independently with FUPAR.

The MPBR affords a variety of fuels (low- or high-enriched uranium, thorium and plutonium) and several fuel management strategies are possible: single-pass, multi-pass, batch refueling, and single-pass with burnable poison. Each scheme results in a different power profile and burnup distribution. The VSOP94 code system can be used to analyze the cost of the entire fuel cycle and calculations will be performed to select the most economical fuel cycle. For example, the once-through fuel cycle, which utilizes low-enriched uranium fuel with B_4C as burnable poison, has recently been proposed as an alternative to the recycling of fuel elements [10].

MCNP Analysis

The MCNP code is increasingly becoming the tool of choice for nuclear reactor design and analysis. This shift away from diffusion-theory codes is largely due to the accuracy of Monte Carlo codes and the availability of low-cost but powerful computing platforms. Nevertheless, diffusion codes are still required to perform most fuel management studies and to calculate temperature-dependent reactor parameters. Due to the high cost of critical core mockups, MCNP is also used to perform calculations that serve as benchmarks for diffusion-code predictions. The MCNP code itself is validated for specific applications using data obtained from relevant experiments.

MCNP modeling of the MPBR will be carried out at MIT. Planned work includes the following activities:

a) The development of a detailed, full-core model with ten axial and five radial burnup zones (plus two in the fuel discharge region at the bottom of the core). The equilibrium burnup distribution will be established with VSOP94 and/or the INEEL code system. This model will be used to calculate key reactor parameters, including flux and power distributions, reactivity worths of absorbers and energy deposition in reactor structures.

b) The determination of the depleted fuel compositions for the equilibrium core using ORIGEN2.1 and MOCUP.

c) The development of a partial-core model for the zone with the highest predicted power density, which will be used to calculate the peak fuel-element power rating. Depending on the requirements of Task 7.1 (Improved Fuel Particle Design and Manufacturing) [11], it may also be necessary to develop an MCNP model of the hottest fuel element. Surface source files will be used to provide an accurate representation of neutron currents at the boundaries of these submodels.

d) The linking of the FUPAR code, which calculates the spatial material density and burnup distribution within the pebble-bed core, to MCNP4B via MOCUP. The goal of this activity will be to develop an MCNP-based, automated, fuel-management code system for the MPBR.

e) The validation of MCNP4B for application to pebble-bed reactors using recent experimental data from the PROTEUS facility at the Paul Scherrer Institute in Villigen, Switzerland [12]. These critical experiments, which were carried out under the framework of an IAEA Coordinated Research Program, were designed to supplement the experimental database and reduce design and licensing uncertainties for small- and medium-sized helium-cooled and graphite-moderated reactors using low-enriched uranium fuel. It may also be possible to include experimental data from the LEUPRO-1 critical facility at the Kurchatov Institute in Russia [13].

Pebble Flow Dynamics

The accurate prediction of hot-spots in the pebble-bed reactor has been identified as potentially a key issue in the licensability of the MPBR in the USA. At present, all simulations are based on a stream-tube model [14]. The pebble bed is treated as a slowly flowing incompressible fluid in which the liquid particles move along streamlines. The motion of the fuel elements is reproduced by the stepwise advance of volumes in the stream tube, loading and unloading being simulated by filling at the top and discharge from the bottom. The probabilistic nature of the pebble-bed reactor is *assumed* to be due to the random manner in which fuel elements are loaded onto the top of the core. The flow velocity is assumed to be one-dimensional in the annular "drop zones."

Experiments performed in the AVR reactor have demonstrated that the incompressible-fluid flow method can predict discharge burnup within 10% [14]. Such accuracy is sufficient for routine neutronic calculations based on zone-averaged power distributions. However, if a finite probability exists for the motion of a pebble to deviate from a streamline, a fresh fuel element could migrate into a high flux region and be overpowered. Since fission-product release from the TRISO coated fuel particles is a function of temperature (with releases beginning to occur above 1600°), the modeling methods currently used for safety analysis may not be conservative.

The correctness of these assumptions will be tested by extending techniques developed at MIT for the atomistic simulation of many-body systems [15] to investigate pebble flow in the pebble-bed reactor. This work will consist of the application of molecular dynamics and/or Monte Carlo techniques to the gravity-driven flow of pebbles through a cylindrical vessel with a conical discharge. Real-time graphics will permit the visualization of individual pebble motion. Recent molecular dynamics studies of granular flow through apertures have demonstrated considerable time-dependent meandering [16]. If similar flow patterns are observed in the pebble bed, it may be necessary to couple this simulation to the MCNP model of the MPBR.

Reactor Kinetics

Reactor kinetics is concerned with the time-dependence of the neutron population in the core, and is mostly governed by the presence of delayed neutrons. The β-decay of fission products can result in the emission of neutrons with time constants that range from milliseconds to minutes. These delayed neutrons are usually grouped according to the β-decay constants of the parent nuclides.

When a nuclear reactor is sufficiently small so that it is neutronically well coupled, the shape of the neutron flux can often be considered to change negligibly during a transient. The MPBR is expected to be such a reactor. In this case, the space-time kinetics problem can be reduced to the point kinetics equations, in which the entire reactor is

treated dynamically as a "point" having certain weighted average properties. In addition to the dynamic reactivity (ρ), these properties are specified by a set of effective neutron kinetics parameters which include the prompt neutron generation time (Λ), and the fractions (β_i) and decay constants (λ_i) of six delayed-neutron groups. These effective kinetics parameters, which depend on the specific fuel composition and core geometry of the reactor, will be calculated using perturbation theory.

The point reactor kinetics model, together with the reactivity coefficients calculated using VSOP or DIF3D/COMBINE, will be used in the simulation of MPBR dynamics. A stability analysis will be performed to confirm that xenon-induced oscillations cannot occur in the small core of the MPBR. Transients involving the asymmetric movement of control and/or shutoff absorbers will be performed initially using the point kinetics model, with temperature reactivity coefficients generated for the perturbed flux shape. A spatial kinetics model, with coupling to a transient thermal-hydraulics code, will be used for the final safety analysis report.

Safety Reviews

This project has not been started but is planned once a suitable student is identified to carry on this research. In essence, this task will require a review of all accident analyses completed to date on pebble and gas reactors in general to identify key safety issues that need to be considered in design. Of significance is the air and water ingress events as are the structural and hydrodynamic effects of blow down during the postulated worst accident and the release of materials that might be plated out on cooler surfaces of the reactor system.

This information will be used to develop a licensing and design strategy to support the "license by test" concept for the power research reactor facility to be built at INEEL. A critical aspect of this work will be the development of the licensing basis for this reactor concept. The South African regulatory authorities utilize a risk (safety) based approach to regulation with an established public health and safety goal principle. US regulations are more prescriptive and not risk based. It is the intention of this project to begin regulatory exchange dialogue with the NRC and the Council for Nuclear Safety in South Africa and the IAEA to develop a uniform standard for the licensing of gas reactors with their unique safety features. This task has been selected by the Collaboration Group of countries which met at MIT on June 8, 1999, as a collaborative effort for all. The countries represented in the Collaboration Group are: Germany, Russia, Japan, China, South Africa, Netherlands and the US through MIT/INEEL and General Atomics.

Economics

Since the fundamental value of this or any other advanced reactor design is whether it will be used, it is vital to estimate the cost of a prototype facility. In today's electricity market, only those power plants that can compete with natural gas will be considered for purchase. The objective of this task is to develop a defensible conceptual cost estimate for the overall plant. Building on the work done at MIT over the last two years and working in close collaboration with ESKOM, a cost estimate will be prepared for the US market for such a project. This cost estimate will be used to show those in Congress and DOE that this project deserves consideration as a future research power facility to be built to test the fuels, safety concepts, integrated control systems and process heat applications. At present, no student has been identified for this task and efforts are underway to identify an interested US architect engineering firm to assist in this effort.

7. Personnel Supported

At MIT the following researchers are and have been supported:

Faculty:

Andrew C. Kadak, Project Manager and Advisor on Non-Proliferation, Core Neutronics, Safety and Economics.
Ronald Ballinger, Co-Project Manager and advisor on fuels and thermal hydraulics.
John Meyer, Thermal Hydraulics
Neil Todreas, Fuel Behavior
Sidney Yip, Fuel Behavior

Students:

Directly Supported:

Julian Lebenhaft – Research Assistant – Core Neutronics
Tamara Galen, Research Assistant – Thermal Hydraulics
Chungyung Wang – Research Assistant – Thermal Hydraulics
Jing Wang – Research Assistant – Fuel Behavior
Guy Snodgrass – Research Assistant – Economics (not for next term)

Working on Project but with their own support:

Paul Owen – Master's Degree – Waste Characterization – completed
Jennifer Anderson – Undergraduate Senior Thesis – Proliferation – Completed
Heather MacLean – Ph.D. Thesis – Fuel Behavior Modeling – ongoing

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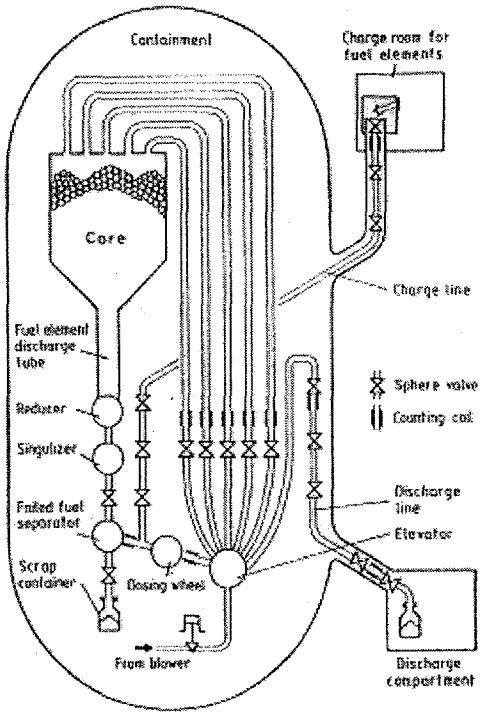

Figure 5. Schematic of the Fuel Handling System

Pebble Bed Gas-Cooled Reactor Balance of Plant Model

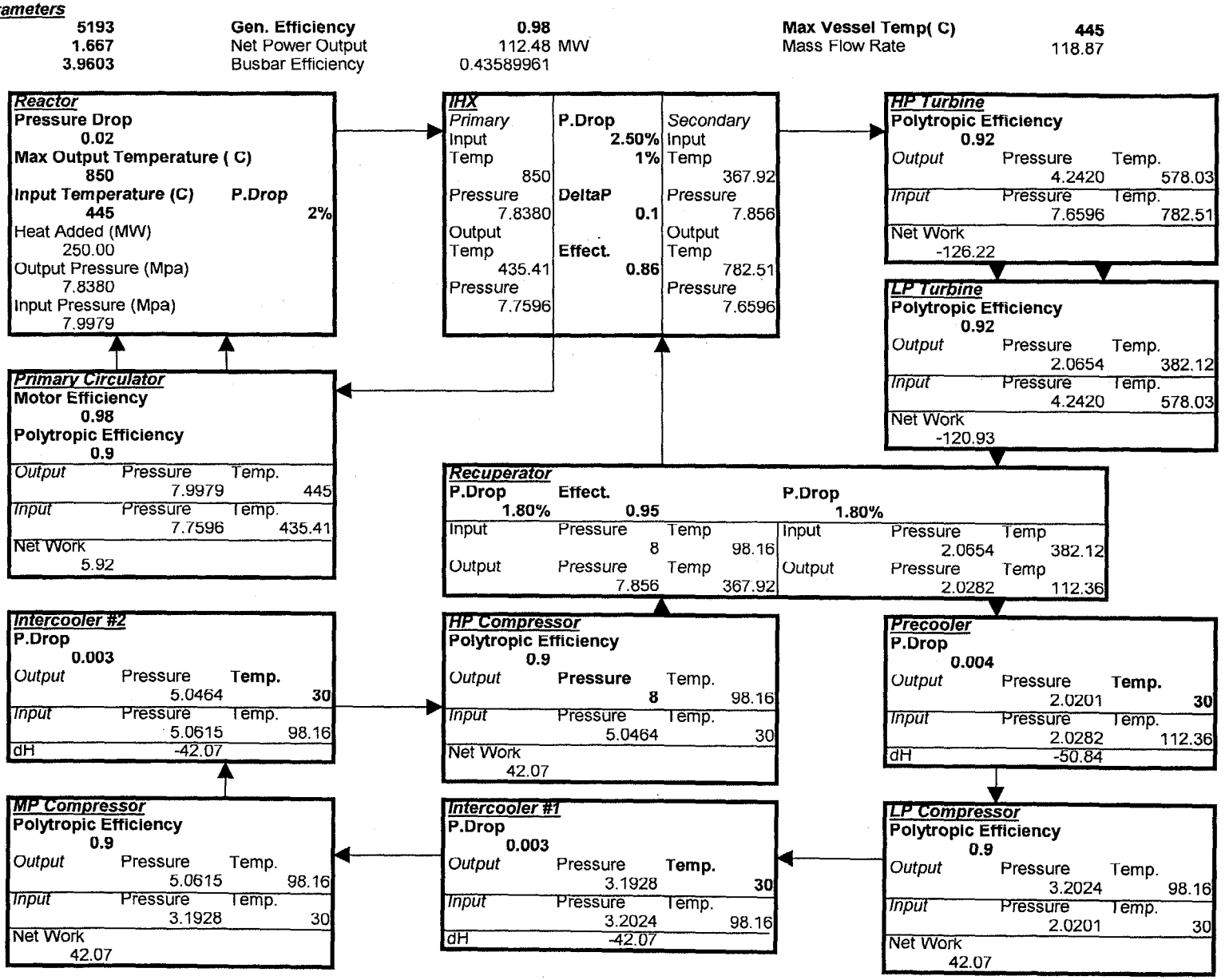

Figure 7 Schematic of Plant System Properties and Flows

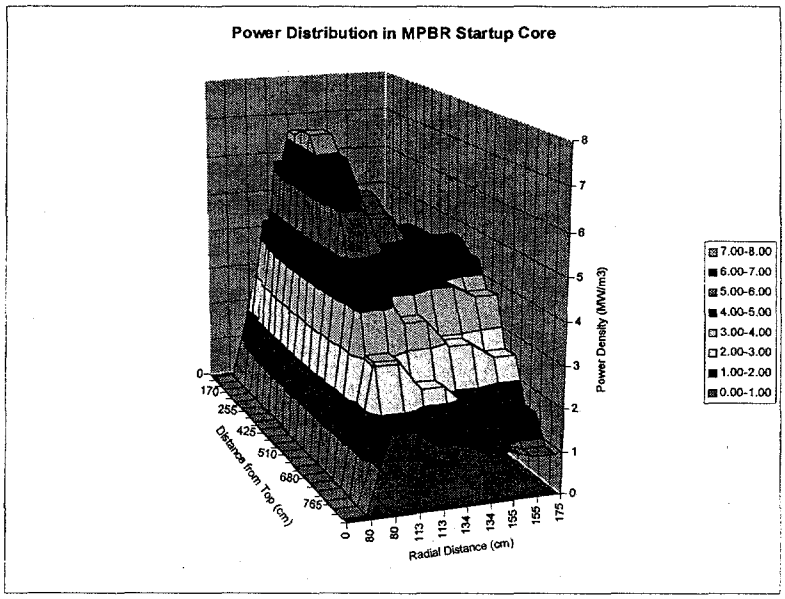

Figure 11. Startup Power Distribution

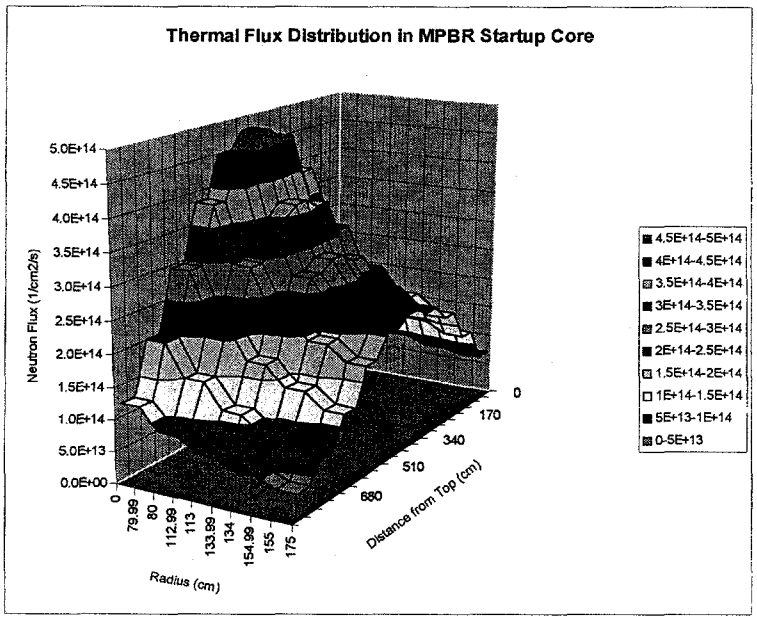

Figure 12. Thermal Flux Distribution

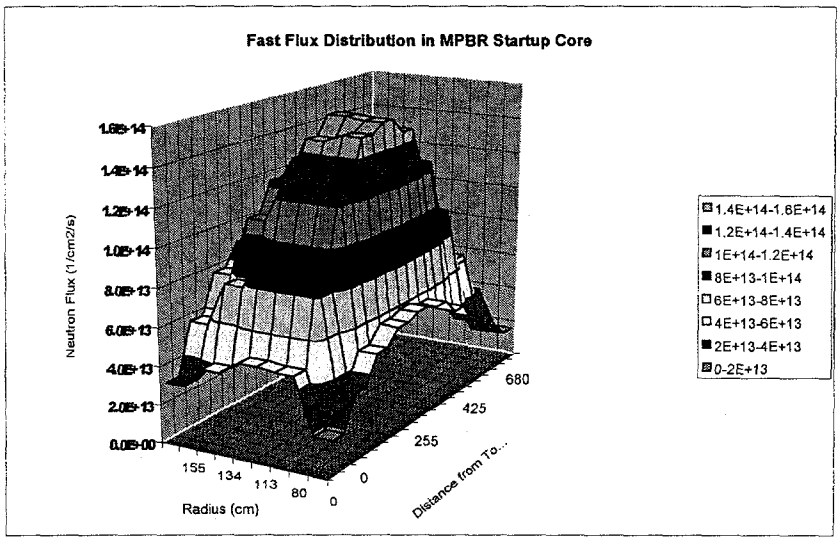

Figure 13. Fast Flux Distribution in Startup Core

Idaho National Engineering and Environmental Laboratory

INEEL/EXT-99-00693
MIT-ANP-TR-065
July 1999

Advanced Reactor Technology

Design of an Actinide Burning, Lead-Bismuth Cooled Reactor That Produces Low Cost Electricity

Annual Project Status Report

Principal Investigators

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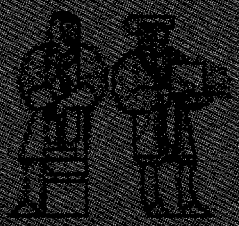
*** Project Leadership**



LOCKHEED MARTIN

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Advanced Reactor Technology

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*** Project Leadership**

TABLE OF CONTENTS

1. Project Overview

The purpose of this project is to investigate the suitability of lead-bismuth cooled fast reactors for producing low-cost electricity as well as for actinide burning. The goal is to identify and analyze the key technical issues in core neutronics, materials, thermal-hydraulics, fuels, and economics associated with the development of this reactor concept.

While considerable design work has been done in the United States, Europe, and Japan on fast reactors, including actinide burners, it has mostly been done for sodium cooled reactors. A lead-bismuth cooled fast reactor was considered in the United States in the 1950s. However, it was abandoned in favor of sodium cooling for two reasons: (1) lead bismuth coolant at the temperatures of interest can be very corrosive to structural materials; and (2) the doubling time of sodium cooled fast reactors can be significantly shorter than that of lead-bismuth cooled reactors as a result of the higher power density achievable in sodium cooled cores. Whereas a short doubling time was considered an important performance characteristic in the fifties, it is of little significance today, as we do not foresee a depletion of low cost uranium resources in the near future, and we have a significant inventory of actinides which can be burned in a fast reactor. Regarding the material compatibility issue, the Russians adopted lead-bismuth for use in their most advanced nuclear submarines, the so-called "Alpha" class submarines which are the fastest in the world. The Russians have built and operated seven Pb-Bi reactors in submarines and two on-shore prototypes. More recently they have studied the design of a variety of lead and lead-bismuth reactors for electric power generation, some of which can operate with one core loading for many years and do not require any fuel reprocessing. Elsewhere, very long-lived core, lead-bismuth cooled, fast reactors have been investigated in Japan, and in the US at the University of California at Berkeley. A lead-bismuth cooled, accelerator-driven, sub-critical actinide burner has been proposed by the Los Alamos National Laboratory for burning the actinides and long-life fission products from spent light water reactor fuel. The Los Alamos system has been labeled Accelerator-driven Transmutation of Waste (ATW). It is envisioned that the reactors investigated in this study could operate in concert with ATWs in a program to both burn the waste from the current generation of light water reactors and produce low-cost electricity. It should also be noted that there exists a synergy between the development of the ATW and a critical system: they share similar coolant and fuel technologies with the result that either system can greatly benefit from improvements achieved for the other.

The choice of lead-bismuth for the reactor coolant in an actinide burning fast reactor offers enhanced safety and reliability. The advantages of lead-bismuth over sodium as a coolant are related to the following material characteristics: chemical inertness with air and water; higher atomic number; lower vapor pressure at operating temperatures; and higher boiling temperature. These basic properties lead to the following advantages for lead-bismuth coolant:

- harder neutron spectrum and, therefore, improved neutron economy, especially when burning actinides;

- better reflection properties, therefore, it is possible to get breeding even without blankets;
- better shielding against gamma-rays and energetic neutrons;
- practically impossible to create a major void in the core due to coolant overheating due to the high boiling temperature and high heat of vaporization of lead-bismuth (a boiling temperature of 1725 C versus 892 C for sodium);
- possibility of eliminating the intermediate coolant loop, and reducing capital costs;
- possibility of using high efficiency heat transfer cycles;
- simpler containment structure due to the impossibility of fires and explosions; and
- small volume change upon solidification.

There are disadvantages to the use of lead-bismuth for cooling an actinide fast reactor in addition to the material compatibility problems. These include: high material cost; higher melting temperature (125 versus 98°C); and the production of Po-210.

Given the status of the field, it was agreed that the focus of this investigation in the first two years will be on the assessment of approaches to optimize core and plant arrangements in order to provide maximum safety and economic potential in this type of reactor. It is envisioned that one pool type plant design would be explored that could accommodate two cores (1) a long-lived core utilizing fertile metal fuel that will consume actinides and (2) a core utilizing non-fertile metal fuel which will maximize the actinide consumption. INEEL and MIT will both work on the plant design. For the first year MIT will focus on the non-fertile fueled core while INEEL will focus on the fertile long-lived core. Four disciplinary-based areas are to be initially investigated:

Neutronic Core Design. Using MOCUP, which combines the MCNP and ORIGIN codes, both MIT and INEEL will investigate the optimum loading of actinides and other materials to create a core that has suitably negative Doppler and void coefficients and a suitably long reactivity lifetime. The work at MIT will initially focus on the use of non-fertile metal fuel. The INEEL will investigate both non-fertile and fertile metal and nitride fuels, looking for optimum economics for an actinide burning, low cost of electricity, reactor design.

Thermal-hydraulic Design Optimization. MIT will investigate pool-type plant designs that will achieve maximum natural convection capability for the core, with the goal of achieving full power removal by natural circulation. MIT will also investigate the economic and safety potential for using a gas turbine power conversion cycle. INEEL will assess the economic and safety potential of the pool type reactor using a steam cycle. INEEL will also assess the plant transient response to various initiating accidents using a modified version of RELAP to account for the coolant and neutronic characteristics.

Fuel and Material Studies. MIT will investigate the suitability of materials for operation over a range of core exit temperatures that will result in economic operation with a secondary side gas cycle. INEEL will investigate the advantages and tradeoffs among various fuel materials such as zirconium-based metallic fuel, oxide fuel and nitride fuel.

Chemical Separation and Waste Assessment. Depending on the fuel materials selected, MIT will perform an assessment of separation technologies and waste disposal strategies in year two.

2. KEY TECHNICAL CHALLENGES

Several technical challenges have been identified that are peculiar to this reactor design.

2.1 Neutronics

Several different fuel types can be used depending on the scope and purpose of the reactor. These fuel types can be lumped into two general categories: fertile and non-fertile fuels. Each type has its own set of challenges, although some of the problems are common to both. The group at MIT has focused on evaluating the behavior of non-fertile fuel, while the group at the INEEL has focused on evaluating the behavior of fertile fuels in a lead-bismuth cooled fast reactor.

To effectively transmute plutonium and minor actinides from LWR spent fuel, it is desirable to minimize the waste of neutrons in order to attain a large surplus available for transmutation. Metallic fuels based on a zirconium matrix provide large excess reactivity due to the low parasitic absorption cross-section of zirconium and due to the hard spectrum achievable because the fuel does not contain any moderating isotope. Nitride fuels may also be suitable.

To maximize the *actinide transmutation* capability of the system, breeding of new fissile material must be minimized hence making the presence of the fertile isotopes U-238 or Th-232 undesirable. The choice of fuel composition for maximum actinide transmutation is then restrained to the to-be-transmuted plutonium and minor actinides (20-30% wt.) and to the zirconium matrix (70-80% wt.), constituting the structural component of the fuel rods. It should be noted that the larger weight fraction of zirconium relative to the heavy metals makes this non-fertile fuel significantly different from the metallic fuel developed by ANL for the IFR project. We envision using dispersion type fuel with IFR type coated particles dispersed in a pure zirconium matrix.

However, to minimize the *cost of electricity* produced, it is desirable to have relatively long refueling cycles so that the plant capacity factors are high and the fuel fabrication costs are low. Long refueling cycles require relatively constant reactivity and, therefore, the use of some fertile material in the fuel. Thorium, with enough depleted uranium to dilute the U-233, is the material of choice, because the end product is not easily separable. Again, a dispersion type metallic fuel with coated plutonium-thorium-uranium particles in a zirconium matrix is envisioned.

The choice of a non-fertile metallic fuel raises three major neutronic challenges:

a) large positive coolant void reactivity coefficient,
b) small Doppler feedback,
c) large rate of reactivity loss with burnup (i.e. the so-called reactivity swing).

In a fertile system, the choice of fuel composition does include the fertile isotopes U-238 or Th-232 (or both), while retaining the to-be-transmuted plutonium and minor actinides present in the non-fertile fuel as the fissile component. The challenge with the fertile fuel is the positive coolant void reactivity coefficient.

Void Reactivity Coefficient. The sign of the coolant void coefficient in fast reactors is the combined result of three conflicting effects upon coolant voiding:

- neutron leakage is increased resulting in reactivity reduction,
- neutron scattering decreases and the spectrum hardens resulting in larger fission-to-capture ratio hence increasing reactivity,
- parasitic captures in the coolant decrease leading to a reactivity increase.

The net outcome is typically a strong reactivity increase due to the latter two effects unless leakage is enhanced enough to offset them.

Doppler Reactivity Coefficient. The amount of fertile isotopes in the non-fertile metallic fuel is very small (mostly Pu-240). Moreover, the hard spectrum leads to a decrease of absorption rate in the resonance peaks. Both these factors result in a very small Doppler feedback and measures to attain a reasonably negative Doppler coefficient must be employed for non-fertile fuel. The fertile fuel can overcome this problem by including enough fertile material to attain a reasonably negative Doppler coefficient.

Reactivity Swing with Burnup. A major consequence of the absence of fertile isotopes is the faster net depletion of the fissionable material and hence a marked reduction of reactivity during a cycle. To ensure criticality at EOL, the reactivity excess at BOL must be large, making this design potentially vulnerable to accidents related to malfunction of the control rod mechanisms (e.g. rapid control rod ejection). In the fertile fuel, the reactivity swing should be very small, requiring less excess reactivity. However, for long-lived fuel the excess reactivity must be sufficient to sustain criticality throughout the life of the reactor.

These challenges are common to most fast reactors and their magnitude is illustrated in **Table 1** for the ANL sodium-cooled, metal-fueled Integrated Fast Reactor (IFR) and for three different fuel compositions.

Table 1. Reactivity swing, void worth, and doppler coefficients in sodium cooled, metal fueled fast reactors.

	Traditional Breeder Fuel	Fertile-free fuel	
Parameter		Weapon Grade Pu Fuel[1]	LWR TRU Fuel[1]
Reactivity Swing ($)	0.09	15.8	7.8
Na void worth ($)	4.78	19.8	6.51
Doppler (¢/K)	-0.05	-0.01	-0.07

2.2 Plant Engineering

2.2.1 Primary System Cost: Enhanced Natural Circulation and Passive Decay Heat Removal System

Key technical objectives are to simplify the primary system, improve its overall reliability and make operation and maintenance less expensive. The development of a passive Decay Heat Removal System (DHRS) and a primary coolant system with enhanced natural circulation capability are potential means to achieve these objectives.

Two options have been considered relative to enhancing natural circulation. One option uses pumps to produce forced convection through the primary coolant system during normal operation, but relies on natural circulation when the reactor is shutdown and just the decay heat is to be removed. The second option relies exclusively on natural circulation in the Pb-Bi, even at full power.

To achieve full power natural circulation in the second option, it may be necessary to design the primary system to allow a substantial relative elevation (i.e. the so-called gravitational head) between the core and the heat sink. Also, friction and form pressure losses must be minimized everywhere in the primary system. These requirements may be in conflict with the need to keep the size of the system within reasonable limits and with the neutronic constraints on the fuel lattice openness[2]. Moreover, a large primary system implies a large inventory of bismuth, which is a relatively costly material.

A shorter reactor pool can be designed if the goal of natural circulation is pursued by mixing the primary coolant with an inert gas[3] in the hot leg to attain a large driving density difference as proposed by Branover [1988]. However, it should be noted that a minimum inventory of primary coolant is required to ensure a sufficient thermal inertia of the reactor pool, which is necessary to smooth transients and to mitigate the consequences of loss of normal heat sink accidents. In this respect it is particularly important to ensure

[1] Fuel with Hf addition to augment the doppler feedback.
[2] A loose lattice may not be neutronically acceptable because it increases the capture in the coolant and worsens the void reactivity feedback.
[3] This is the so-called gas-pump approach.

that enough decay heat can be accumulated in the reactor pool without excessive temperature rise before the DHRS comes online to provide a long-term heat sink.

The low chemical reactivity of Pb-Bi with water and air eliminates the need for the costly intermediate loop typical of sodium cooled fast reactors. However, placing the primary-to-secondary heat exchanger directly in the reactor pool raises a new kind of concern regarding the reactivity response to core flooding by the secondary coolant (i.e. water or helium) in case of heat exchanger tube failure. Thus, the primary coolant flow path in the reactor pool may have to be designed to prevent entrainment of undesirable fluids in the core.

It is clear then that selecting the primary system configuration is an involved challenge driven by the trade-off between forced and natural circulation, neutronics, safety, cost and decay heat removal requirements.

2.2.2 Operating Costs: High Efficiency Power Cycle and Materials Corrosion Control

It is essential to maximize the production of electricity in order to improve the economic acceptability of the system. A suitable conversion power cycle to best exploit the thermal power produced in the core is to be selected: both gas and steam turbine power cycles are potential candidates.

Gas turbine cycles are simpler, easier to operate and, if a chemically inert gas is selected as the working fluid (e.g. helium), they do not pose any major corrosion problems. However, they may have to be operated at high temperature (>550°C) to attain an acceptable thermal efficiency. Moreover, it should be noted that the operating experience with helium turbines is limited.

On the other hand the use of a steam cycle ensures thermal efficiencies above 30% at relatively low temperatures (<350°C), as proven in LWR plants.

In any case a high Pb-Bi temperature in the primary system is desirable since it increases the thermal efficiency of the power cycle. Given the corrosive nature of liquid lead-bismuth at high temperature, the development and selection of compatible structural materials becomes a key issue. Normal austenitic stainless steels cannot be used because nickel rapidly dissolves in bismuth. Current materials investigations focus on the characteristics of a martensitic-ferritic stainless steel (designated EP-823) of low nickel and high silicon content especially developed in Russia for Pb-Bi applications. Moreover, the extremely hard neutron spectrum raises additional concerns regarding the mechanical performance of the reactor materials: the permanent supporting structures in the core (e.g. the core plate, the core barrel) are expected to be irradiated at neutron fluences that may cause significant swelling and embrittlement.

Finally, special emphasis is given to the theoretical determination of the thermo-physical and mechanical properties of the selected fuel because very little data relevant to

this fuel are available in the literature. Nevertheless it should be noted that a fuel mainly composed of Zr (as opposed to a fuel mainly composed of actinides, like the IFR fuel developed by ANL) is expected to yield better overall performance due to the larger thermal conductivity, the superior physical stability (i.e. less changes of phase) and mechanical characteristics of Zr compared to actinides. A coated particle dispersion type of arrangement may further improve the performance of this fuel.

2.2.3 Coolant Activation

The long-lived α-emitter ^{210}Po is produced in the coolant by the following nuclear reaction:

$$^{209}Bi + n \rightarrow ^{210}Bi \underset{t_{1/2}=5days}{\overset{\beta^-}{\rightarrow}} ^{210}Po$$

Preliminary calculations show that the concentration of this radionuclide in the primary coolant can reach several hundreds of Ci per liter of Pb-Bi if not continuously removed by means of an active purification system. Given the high radio-toxicity of this nuclide, its Derived Air Concentration (DAC) limit is very stringent (10 Bq/m^3) and special attention must be paid to ensure that the releases of contaminated Pb-Bi are minimized.

2.2.4 System Analysis Capability

One technical challenge is to develop a thermal-hydraulic system analysis capability that can be used in both the design of the reactor and in the evaluation of its safety. The system analysis capability allows for a simultaneous modeling of the primary and secondary coolant systems, the decay heat removal system, and the reactor neutronics at normal operation and during transients and accounts for the interactions between systems. The approach taken to meet this challenge is to modify an existing thermal-hydraulic system analysis code, based on RELAP5 (LMITCO 1995), to represent a Pb-Bi reactor.

3. Design Progress

This section presents the design results that have been achieved since project inception in October 1998. These design results derive from the effort that has been aimed at exploring and surmounting the design challenges presented in Section **2**.

3.1 Neutronics (MIT)

3.1.1 Void Reactivity Coefficient in a Non-fertile, Metal Fuel Core

A major effort was focused on the development of a non-fertile, metal fuel core with a negative reactivity coefficient when the whole core is voided, or when only a limited region of the core is voided. The latter situation may arise if a steam generator tube ruptures and a subsequent steam or helium bubble is transported into the core, or in the case of cladding failure and gas release from one or more fuel rods. Pursuing the goal of reactivity decrease upon partial core voiding will add complexity to the fuel assembly design; therefore, a detailed analysis of various accident situations must ultimately be undertaken to assess if a negative partial void reactivity coefficient is really necessary.

Several strategies to attain negative void coefficient were explored by means of the MCNP code. They involve:

- Decreasing the pitch to diameter ratio to reduce the magnitude of the positive void coefficient due to smaller change of absorption rate in coolant and smaller spectrum shift from coolant voiding. However, the reduction of pitch to diameter was restricted because of thermo-hydraulics limitations and even tight pitch to diameters were not able to achieve negative coolant density coefficient.
- Loading the ends of fuel pins with stainless steel reflector material followed by B_4C absorber pellets enriched with B^{10} or zirconium hydride to increase leakage/absorption to peripheral absorbers upon coolant voiding. This approach enabled the attainment of negative reactivity change with coolant density. However the drawback was a more complicated fuel rod design and an appreciable reactivity penalty.
- Reducing core height to increase leakage in the axial direction. Core heights of about 1m yielded a negative reactivity change with coolant density.
- Introducing blanket assemblies heterogeneously placed in the core to increase leakage into these blankets upon coolant voiding.

Although the pin shields with absorbing material, pancake cores with small height, or heterogeneous cores with blanket assemblies provided negative reactivity change with decreasing coolant density*, none of these strategies appeared to be effective when only the central region of the core was voided. Therefore, an effort has been made to design a

* Reactivity changes were calculated by varying the core-average coolant density, which was assumed uniform over the entire core, while coolant in the lower-plenum, in the radial reflector and in the chimney above the core were kept constant.

core that can avoid reactivity spikes in the event of partial core voiding. This can be done if neutrons can leak directly from the core center.

Two approaches were found to be effective. The first design involves a pressure-tube reactor similar to the CANDU design with gas-filled space between the pressure tubes. However, pressure tubes pose significant material problems in a high-flux fast reactor with a hard spectrum. Also inadvertent flooding of the inter-tube space with coolant from failed pressure tubes results in a reactivity increase. Therefore, the design of a fuel assembly that allows streaming of neutrons in both radial and axial directions from the center of the core was considered. A number of various streaming fuel assemblies and core configurations have been investigated. One of the alternatives giving plausible void reactivity performance is shown in **Figure 1**.

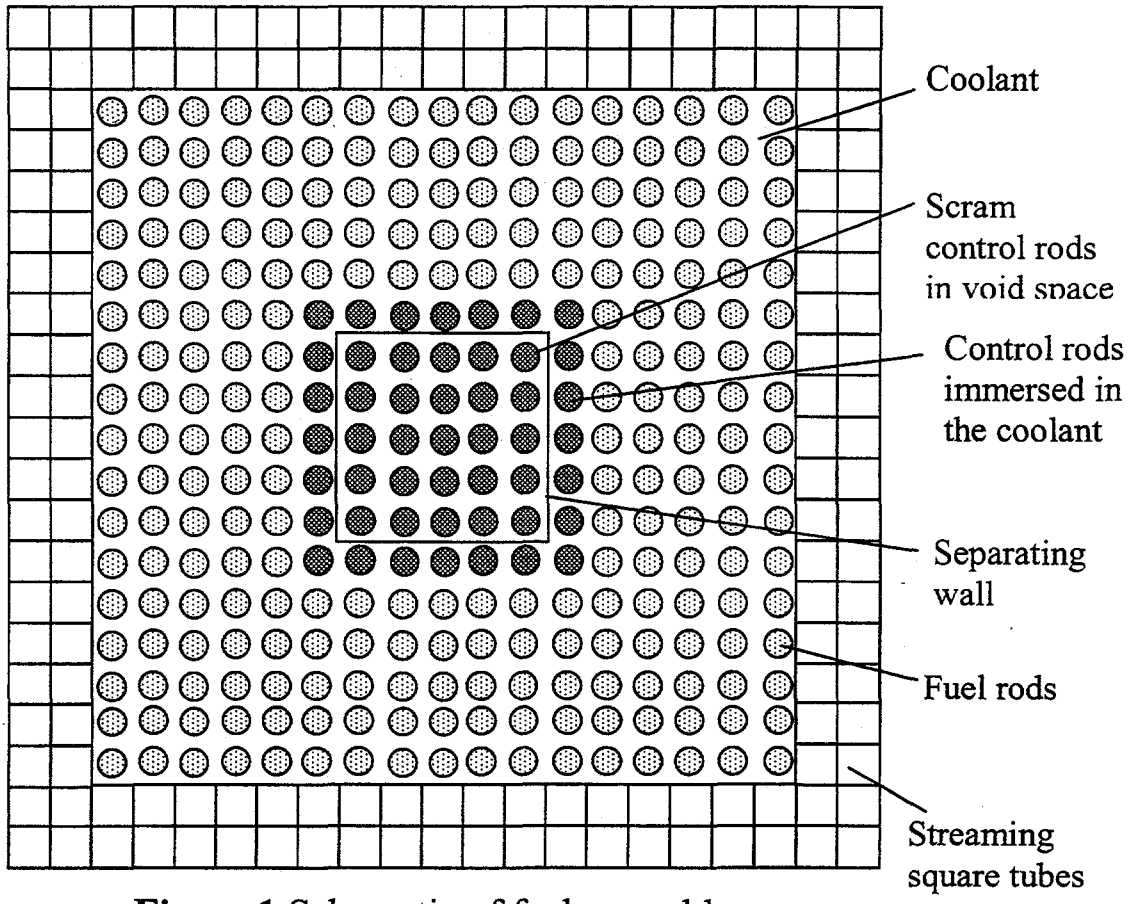

Figure 1 Schematic of fuel assembly with control rod drives

The assembly contains 21×21 positions with 240 fuel rods, 152 square streaming tubes arranged in two rows at the fuel assembly periphery and 7×7 positions in the fuel assembly center. The streaming tubes are filled with gas and sealed. In addition to their neutron streaming function, they also provide structural support for the assembly making possible elimination of separate tie rods. In case of failure of a streaming tube, only one

tube will be flooded with coolant leading to a very small reactivity increase. To achieve a significant reactivity change in such a scenario, a very large number of these tubes would have to be simultaneously flooded. Widespread void flooding is thus extremely unlikely. The thickness of the walls of the square tubes was taken the same as that of cladding, i.e., 0.63mm. The central region of each fuel assembly contains a 5×5 void space to enhance neutron streaming. This space is also used for scram control rods (located outside the core under normal operating conditions). An additional 24 control rods are located around the assembly central region to compensate for long term reactivity decrease and to control the reactor power in transients (for a more comprehensive discussion on the control system see Section 3.4.5).

The core schematic in **Figure 2** shows the location of the fuel rods within the assemblies and the relative power in each fuel assembly. The control rods are made of B_4C with 90% B10 enrichment and are used for both reactivity compensation and control. The main data used for neutronic analysis are summarized in **Table 2**.

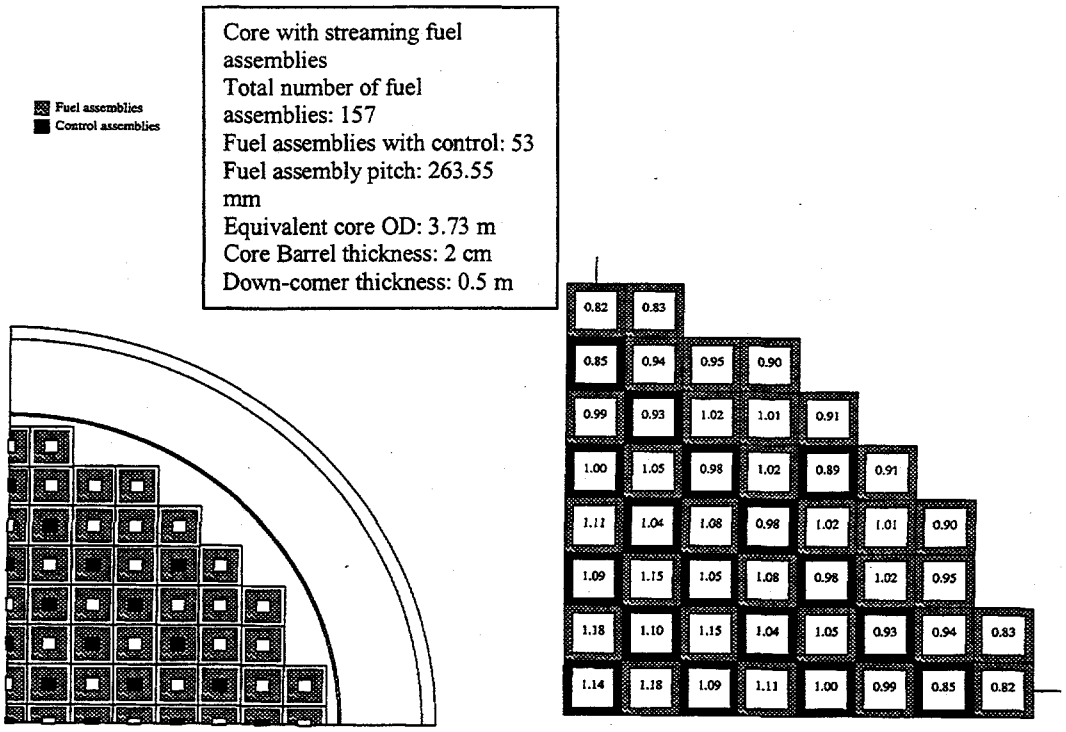

Figure 2 Core layout and fuel assembly power distribution

Table 2 also gives the effective multiplication coefficients for various states of the core. Reactivity exhibits a decreasing trend with decreasing coolant density. Partial core voiding also yields a reactivity decrease, both in case of voiding 50% of 4 fuel assemblies or in case of 50% voiding of only one central fuel assembly. At high burnups, the void reactivity behavior gets slightly worse, as the fuel composition changes to a less favorable state with respect to variation of capture and fission cross sections due to spectrum hardening. However, the reactivity increase at high burnup for the case of partial voiding of 4 fuel assemblies is very small. The calculations also show that flooding either the

entire bank of fuel assembly peripheral streaming tubes (152 tubes) or of the fuel assembly central void region results in reactivity reduction. Hence, the core design with streaming fuel assemblies appears very promising and apparently overcomes the void reactivity problem.

Table 2. Reactivity performance of the fresh core (uniform enrichment)

Case (control rod drives are 84.5cm in the core unless specified otherwise)	k_{eff}
Reference case (core-average coolant density =10.25g/cm^3)	0.999±0.001
Core-average coolant density =8g/cm^3	0.992±0.001
Core-average coolant density =6g/cm^3	0.988±0.001
4 fuel assemblies voided*	0.998±0.001
Control rod drives from the central fuel assembly fully withdrawn	1.004±0.001
Worth of central control assembly	0.005k/k
1 central fuel assembly voided** (for fully withdrawn control rod drives in central fuel assembly)	1.002±0.001
1 central fuel assembly fully flooded –all peripherial streaming tubes (152) flooded	0.996±0.001
Entire central void region with control rod drives 84.5cm in the core in central fuel assembly flooded	0.998±0.001
All control rod drives fully withdrawn - burnup=0GWd/tHM –coolant density=10.25g/cm^3	1.241±0.001
All control rod drives fully withdrawn - burnup=198GWd/tHM+-coolant density=10.25g/cm^3	1.031±0.001
All control rod drives fully withdrawn - burnup=198GWd/tHM+- 4 fuel assemblies voided*	1.032±0.001
All control rod drives fully withdrawn - burnup=198GWd/tHM+-coolant density=8g/cm^3	1.031±0.001

* Partially voided central 50% of 4 fuel assemblies in the core central region
** Partially voided central 50% of 1 central fuel assembly (control rod drives from this assembly withdrawn)
+ 198 GWd corresponds to 632 full power days of exposure, tHM is metric tons of heavy metal

The large reactivity compensation by the control rods results in a relatively large axial power peaking at BOL, as shown on **Figure 3**. Modifications of the control system will be introduced in this effort to lower this peaking.

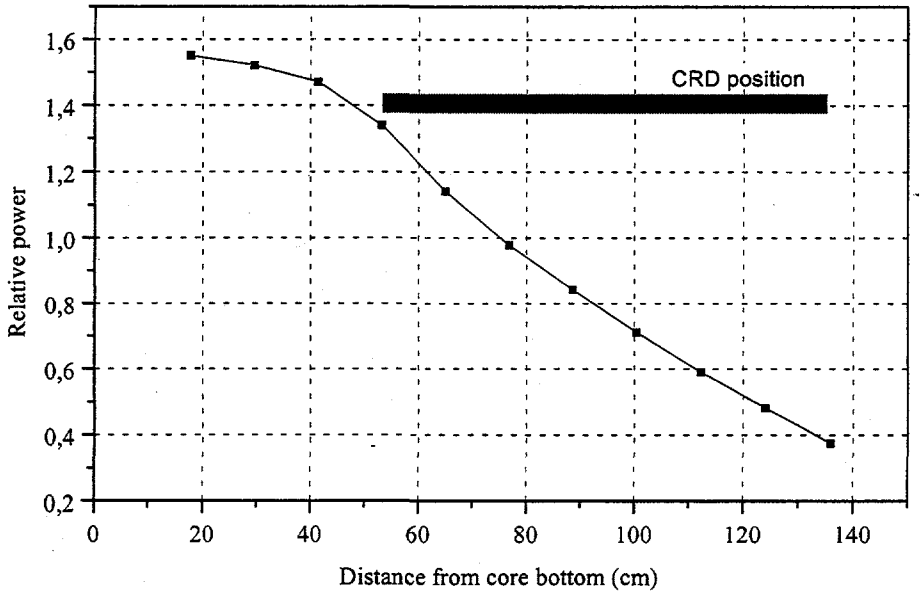

Figure 3 Axial power density profile (average over all fuel rods) at BOL

3.1.2 Reactivity Changes with Burnup and TRU Consumption Rate in a Non-fertile, Metal fuel Core

Changes of fuel compositions during burnup were modeled using MOCUP using a unit cell model. The fuel composition obtained from this unit-cell model at selected burnups was then used in the full core model to obtain k_{eff}. The results are plotted in **Figure 4**.

It can be observed that the core can operate for about 1.7 years, achieving burnup of 190,000MWd/t of heavy metal. Several burnable absorber materials, such as Re, B-10 or Sm-149 were placed in the zirconium matrix and were evaluated to explore options of reducing the slope of the burnup curve as a means to reduce rod worth and extend cycle length. Boron exhibited the best performance, but none of the isotopes investigated appeared very effective. Although the slope of the reactivity slope was reduced, the burnup rate of the absorbers was low due to their small cross section in a hard spectrum yielding a large residual inventory at end of life. Since no materials were identified with significantly larger capture cross sections than those of boron, burnable poisons were discarded in favor of excess reactivity compensation through control rods.

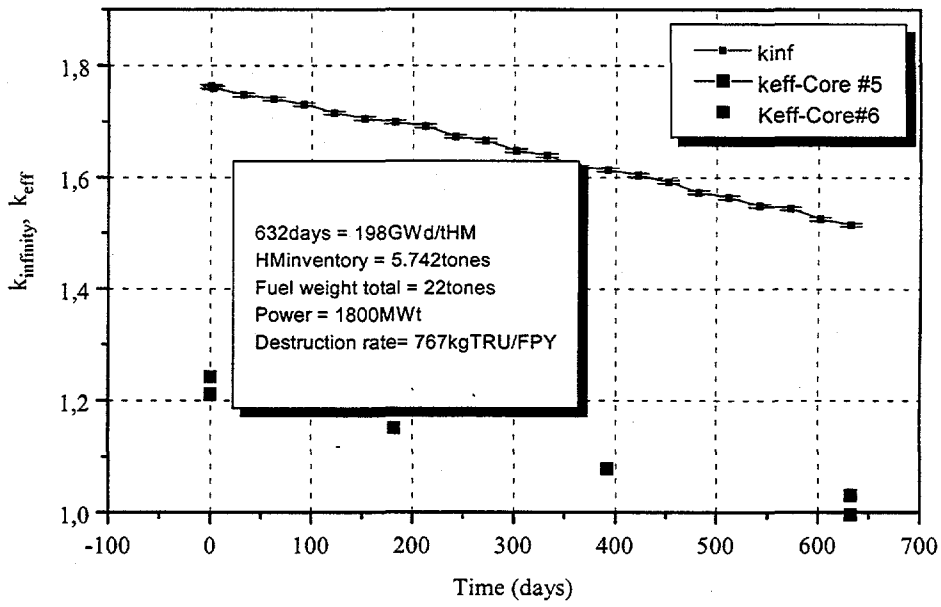

Figure 4. Reactivity change with burnup

The destruction rate of individual actinides per full power year, as obtained from MOCUP, is plotted in **Figure 5**. These results indicate that the proposed Pb-Bi burner with streaming fuel assemblies is quite effective for actinide burning. The total destruction rate of TRUs at an operating power of 1800 MWth is about 767 kg/FPY. This compares with 650 kg/y per 2000 MWth for ATW [Venneri 1998], or 585 kg/y per 1800 MWth. However, this comparison does not include the capacity factor of ATW, which was not known to us. If an ATW capacity factor of 0.8 was assumed, the ATW destruction rate would be 731 kg/FPY. The spallation reactions deposit appreciable heat within the system, hence for the fixed total thermal power, the fission-generated power output, which is the key factor for the incineration rate, is smaller than in the case of a Pb-Bi critical burner. Consequently, the ATW destruction rate is less, which should explain the difference between the values of 767 and 731 kg/FPY above. Hill et al., [1995] studied a sodium cooled fast reactor as a pure burner using metallic fuel and estimated a destruction rate of 231 kg TRU/y per 840 MWth at a capacity factor of 0.75. Scaling their power to 1800 MWth and incorporating the capacity factor yields a destruction rate of 660 kg TRU/FPY. Since the power released per fission is approximately the same for Pb-Bi and sodium-cooled fast reactors, the number of fissions is also comparable for the fixed core power. Therefore, the reason for different TRU consumption rate must come

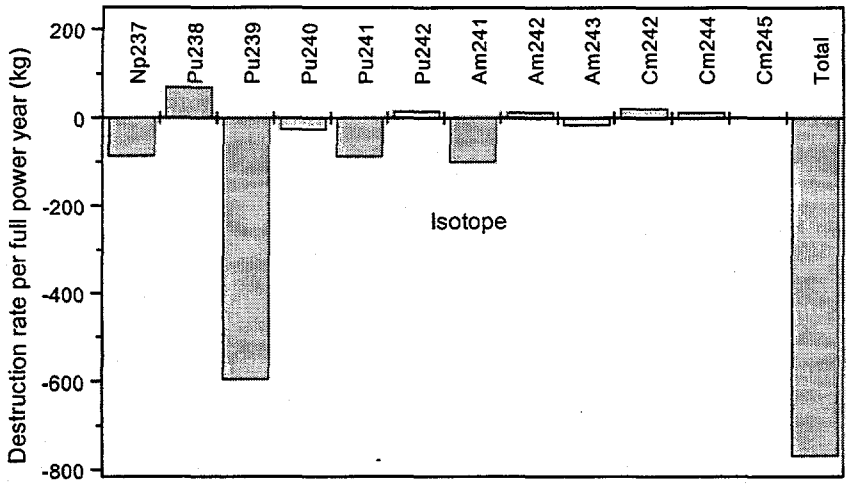

Figure 5. Destruction rate per full power year of various isotopes calculated by MOCUP for a non-fertile, metal fuel core.

from spectral effects. The very hard spectrum of the Pb-Bi burner results in less captures and therefore a smaller conversion ratio. For example, for the given core design, the conversion ratio* is only 0.23. Consequently, the net effect is a higher destruction rate of actinides.

3.2 Neutronics (INEEL)

3.2.1 Comparison of Reactivity Swings with Information Published by Sekimoto

Japanese researchers (Sekimoto and Su'ud) at the Tokyo Institute of Technology have performed a 2-dimensional analysis of possible lead and lead-bismuth reactors using the SLAROM computer code. In their first analysis, they constructed a block core that contained a shield, reflector, outer core, middle core, and central core. A modification, which included blanket material within the central and middle core region, was also analyzed. The fuels used in the calculations were metallic (10% Zr) and nitride, with the nitride fuel slightly outperforming the metallic fuel. Also included in the calculations were different fuel enrichments using plutonium as the fissile component, and a fuel volume percent of 45-50%. The coolants studied were lead and lead-bismuth, with lead-bismuth showing better overall properties. The calculations were performed for a core

* Conversion ratio is defined here as $CR = \sum \sigma_{c,j} N_j \phi / \left(\sum \sigma_{c,j} N_j \phi + \sum \sigma_{f,j} N_j \phi \right)$, where the sums are over all heavy metal nuclides, j.

lifetime that was to exceed 10 years without refueling or fuel shuffling, while maintaining burnup reactivity swings to less than 0.1% Δk/k. Other restrictions include maintaining a negative total-core coolant void coefficient of reactivity over all of the burnup period, omission of an intermediate heat exchanger, and a large natural circulation contribution. Their 2-dimensional calculations show that a reactor can maintain criticality for ~12 years with the above limitations.

This block core was used as a reactivity benchmark to compare a 3-dimensional analysis with a 2-dimensional one. This was accomplished by using MOCUP to perform a time-dependent treatment of nuclides within the reactor. The core was modeled using homogenous fuel blocks, with the pin shields and reflectors also being modeled as homogenous blocks. The burnup steps were performed at one-year intervals for a total burnup of 12 years. Thus far, three cores have been burned using different fuel volume percentages. The Sekimoto results can be seen in **Figure 6**, and the MOCUP results are summarized in **Figure 7**. Sekimoto's results show a decrease in reactivity during the first 3 years, and then an increase between the third and fourth year, with a peak occurring between the eighth and ninth years. He attributes this reactivity swing to the buildup of fissile plutonium from the fertile U-238 in the fuel. This is notably different from the reactivity swing found in the MOCUP results. The slope of the reactivity change with time in all three MOCUP cases is almost identical; there is no gradual increase in the reactivity. By extrapolation, an optimum fuel volume percentage for a 12-year burn can be found at ~46%. This is within the same volume percent range that Sekimoto used to achieve a 12-year critical cycle.

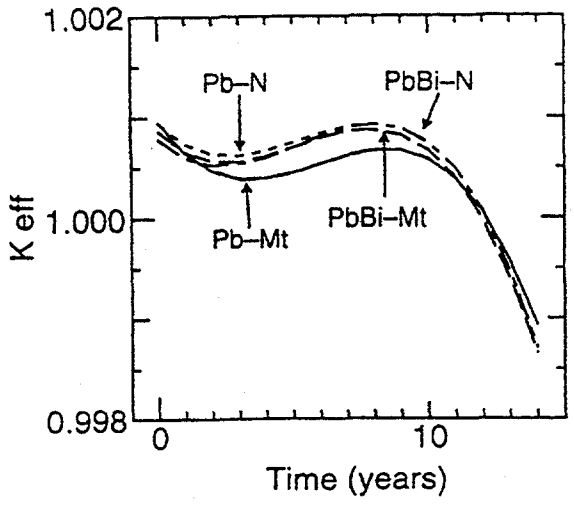

Figure 6. Reactivity of Sekimoto core using different fuels and coolants.

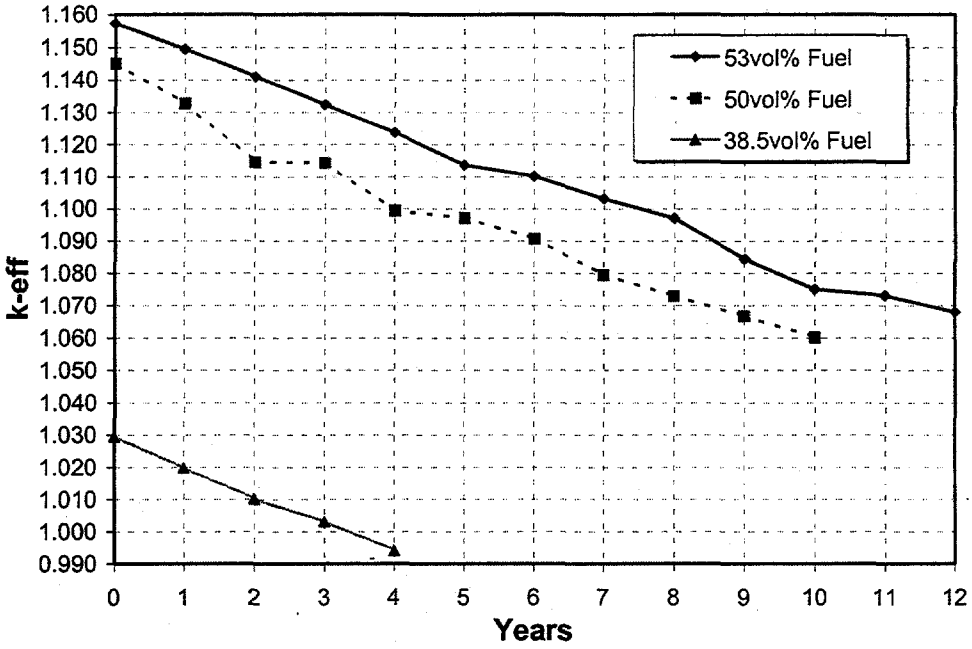

Figure 7. MOCUP calculated reactivity changes with time in the Sekimoto core for three fuel types: 38.5, 50, and 53 volume percent fuel.

3.2.2 Evaluation of Fertile Fuel Options

The advantages of using fertile fuel compared to a non-fertile fuel are:

- better Doppler reactivity coefficient,
- smaller reactivity swing throughout the life of the reactor, and
- longer fuel life.

While these advantages can be significant over a non-fertile system, there still remains the problem of the coolant void coefficient, and the rate at which the actinides can be burned compared to a non-fertile fuel. The void coefficient was addressed in a previous section and will not be re-addressed here. The TRU consumption rate for a fertile fuel will depend upon the reactor power, the TRU content, and the economics involved. The economics will be addressed later.

Because one of the goals of this reactor is to burn actinides, the composition of a fertile fuel must be carefully chosen so as not to produce more actinides than are to be burned, yet the reactor must still be capable of long refueling cycles. This particular requirement would seem to favor thorium over uranium as a fertile material, especially if nonproliferation by way of plutonium production in the reactor is considered an important factor. However, the production of U-233 from thorium must also be considered as a proliferation concern, which would necessitate the addition of some uranium to the fuel (preferably depleted uranium). With these considerations in mind, unit-cell calculations using MOCUP were performed on metallic and nitride fuels. The plutonium and minor actinide composition was held constant at 20% wt., with the

remainder being either uranium/zirconium or thorium/zirconium, and only zirconium in the case of the non-fertile fuel. **Table 3** gives a summary of the fuel composition at BOL.

Table 3. Fertile and non-fertile fuel composition (BOL).

	Non-fertile (wt%)	**Metallic** (wt%)	**Nitride** (wt%)
Plutonium	**16%**	**16%**	**16%**
Pu-238	0.32%	0.32%	0.32%
Pu-239	9.28%	9.28%	9.28%
Pu-240	4.16%	4.16%	4.16%
Pu-241	1.6%	1.6%	1.6%
Pu-242	0.64%	0.64%	0.64%
Minor Actinide	**4%**	**4%**	**4%**
Np-237	1.72%	1.72%	1.72%
Am-241	1.8%	1.8%	1.8%
Am-243	0.36%	0.36%	0.36%
Cm-244	0.12%	0.12%	0.12%
Uranium (nat.) – if used	-	**70%**	**74.5%**
Thorium – if used	-	**70%**	**74.4%**
Zirconium	**80%**	**10%**	-
Nitrogen	-	-	**5.5%** (w/ U) **5.6%** (w/ Th)

The next consideration in using these different fuel types is the reactor power level. This issue becomes important for long-lived cores because the amount of fissile material that will be needed to sustain a critical reactor will have to be adjusted or optimized. The fissile components used for the current analysis are Pu-239 and Pu-241, and were kept constant at a combined weight percent of 10.88%. Although choosing a constant weight percent means that the atom densities for the fissile plutonium will be different for each case, the results of the calculations will be used to extrapolate to an optimal fuel composition. The power levels were chosen at 833 MWth (300 MWe) and 5000 MWth (1800 MWe) with more calculations to be done at 2777 MWth (1000 MWe). The calculations assumed an efficiency of 36%.

The current analysis has been limited to IFR type fuel, i.e., cylindrical pins placed in assemblies. The parameters of the pin can be seen in **Table 4**. The metallic fuel has two interesting properties that are unique to this type of fuel:

- The gas plenum above the fuel.
- The materials used in the gap between the fuel and the cladding.

The gas plenum is used to capture the fission gases as they diffuse from the metallic fuel. This plenum may be reduced somewhat if a dispersion fuel is used in place of the IFR type fuel. The gap material is used to enhance the thermal conductivity of the fuel. However, use of a dispersion fuel would probably eliminate the need for gap materials. It is important to note that other fuel geometries may show better neutronic or thermal hydraulic properties, and will be studied once the fuel composition has been optimized.

Table 4. Parameters of the cylindrical fuel pins.

Design Parameter	**Value**
Fuel OD	0.864 cm
Gap Thickness	0.02 cm
Gap Material (metallic fuel only)	33wt% Pb - 33wt% Sn - 33wt%Bi
Cladding Thickness	0.063cm
Cladding OD	1.03 cm
P/D	1.6
Active Fuel Height	120 cm
Gas Plenum Height (metallic fuel only)	90 cm
Average Coolant Density (Pb-Bi)	10.25 g/cm^3

The burnup time steps taken in MOCUP for all cases were one-year steps with no outages. The results for the non-fertile, plutonium-uranium-zirconium (Pu-U-Zr), and plutonium-thorium-zirconium (Pu-Th-Zr) fuels at 300 and 1800 MWe are shown in **Figures 8 and 9**. Note that the Pu-U-Zr fuel has the highest reactivity at BOL and a smaller reactivity swing than the non-fertile fuel. This fuel has the potential for the longest life for relatively similar amounts of added burnable actinides compared to the other fuel compositions. However, the Pu-U-Zr fuel will increase the inventory of plutonium and other actinides in the fuel rather than decrease (or burn) them and the excess reactivity is high at BOL. Therefore, the actinide burning rate and the excess reactivity become major disadvantages for this fuel composition. These disadvantages far outweigh any advantages gained in reactor life. **Figure 10** shows the results for a comparison between metallic fuel (Pu-U-Zr) and nitride fuel (Pu-U-N). The excess reactivity and reactivity swing in the nitride fuel core appear to be smaller than in the metallic fuel (Pu-U-Zr) core, but the same problems exist. The nitride fuel (Pu-U-N) will still breed rather than burn actinides, and still has a higher BOL excess reactivity than the non-fertile fuel.

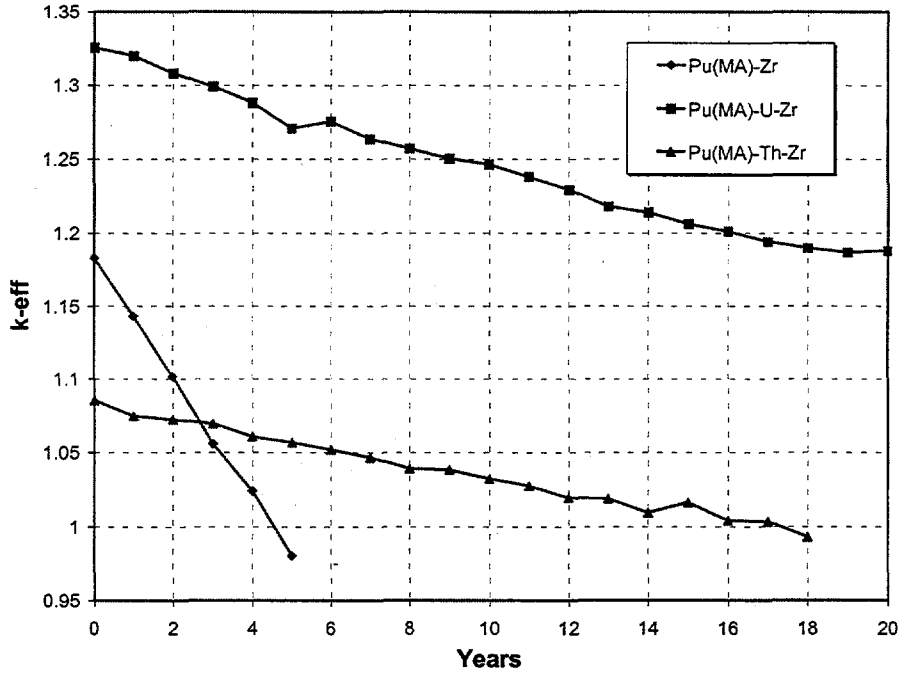

Figure 8. Comparison of metallic fuel types at 300 MWe.

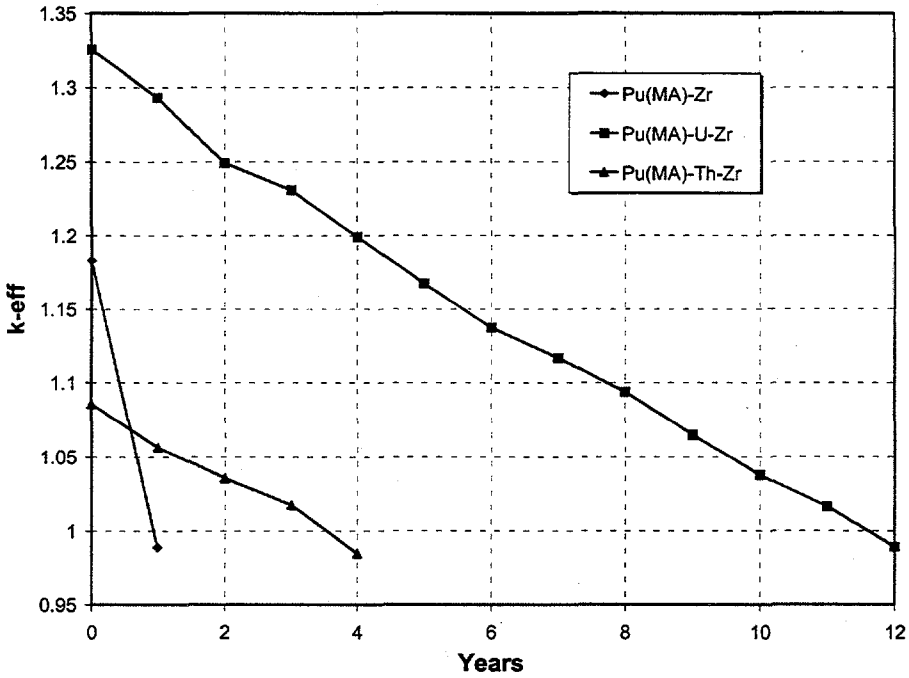

Figure 9. Comparison of metallic fuel types at 1800 MWe.

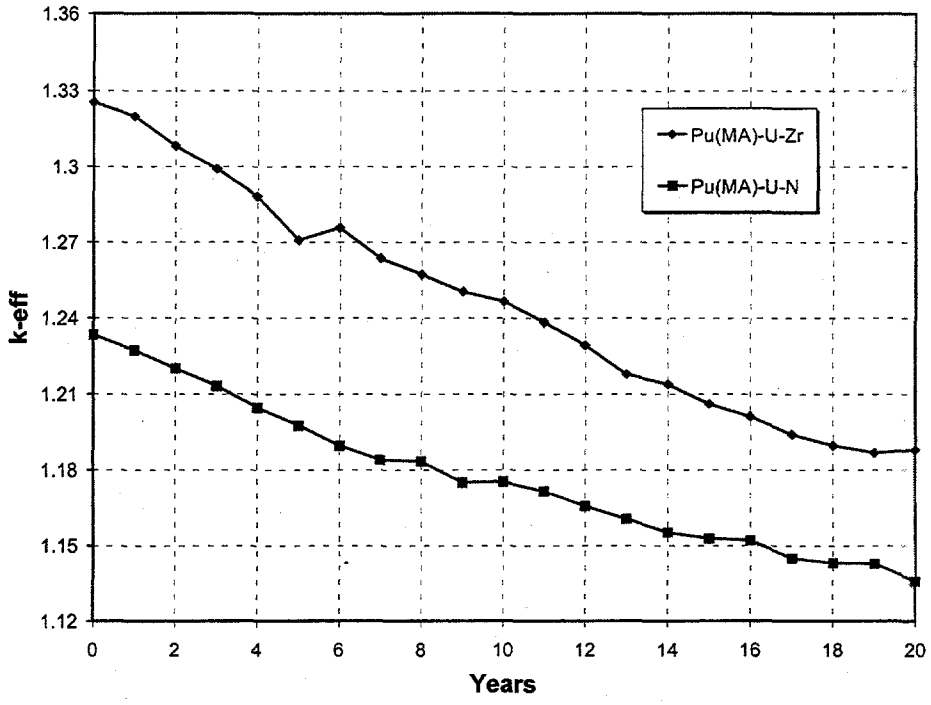

Figure 10. Comparison of metallic and nitride fuel at 300 MWe.

From the results thus far, the optimum mix appears to be the Pu-Th-Zr fuel, although calculations need to be performed on the Pu-Th-N fuel to verify the hypothesis. The Pu-Th-Zr fuel shows promise in that the reactivity swing is slightly smaller than the Pu-U-Zr fuel; the excess reactivity is lower than all other fuel types, therefore, more actinides can be burned in a given cycle; the actinide production (especially plutonium) is much lower; and a relatively long fuel life can be attainted. In other words, because the excess reactivity is low and the reactivity swing remains fairly constant throughout the life of the Pu-Th-Zr fuel, it is possible to add more actinides to increase the cycle length and thereby the burn rate. Note that because U-233 will be present in the Pu-Th-Zr fuel at the EOL, some uranium will need to be added to denature the fuel. Although some plutonium will be produced because of the addition of the uranium, the amount should be minimal.

3.3 Fuel and Materials Selection and Limits (MIT and INEEL)

The reference non-fertile fuel is made of zirconium, plutonium and minor actinides (the weight fraction is 74, 20.8 and 5.2%, respectively). Two different types of fuel form can be considered:

a) A metallic alloy of a zirconium, plutonium and minor actinides. Metallic alloys of zirconium, uranium and plutonium have been the subject of extended studies at Argonne National Laboratory (ANL) in the past two decades. However, in the ANL fuel the main component (i.e. the continuum phase) is always a heavy metal (i.e. uranium and/or plutonium). The presence of a significant amount of minor

actinides and the prevalence of zirconium over Pu, make the selected fuel relatively novel.

b) A dispersion type fuel with coated particles containing about 80 to 90 % TRU and the remainder zirconium in a pure zirconium matrix. The particle coating is yet to be determined, but could be zirconium carbide. Coated particle dispersion fuels are known to have better geometric stability under neutron irradiation and to retain more fission products, especially the gaseous fission products. Therefore, it is likely that a fuel design can be developed that does not require large fuel-to-cladding gaps and metal bonding in the gap. Some work has been done on this type of fuel for space reactor applications.

In the following sections only fuel a) is considered. Since the availability of experimental data for the thermo-physical characteristics of this fuel is scarce, a simplifying assumption is to be made: the properties of the minor actinides are set equal to those of plutonium.

3.3.1 Non-fertile Fuel Melting Temperature

Figure 11 illustrates the phase diagram of the binary alloy Zr-Pu, which provides the melting point and the crystalline configuration of the alloy fuel as a function of the zirconium weight fraction. It can be seen that at reference composition (74% Zr - 26% Pu) the fuel melting point is approximately 1600°C.

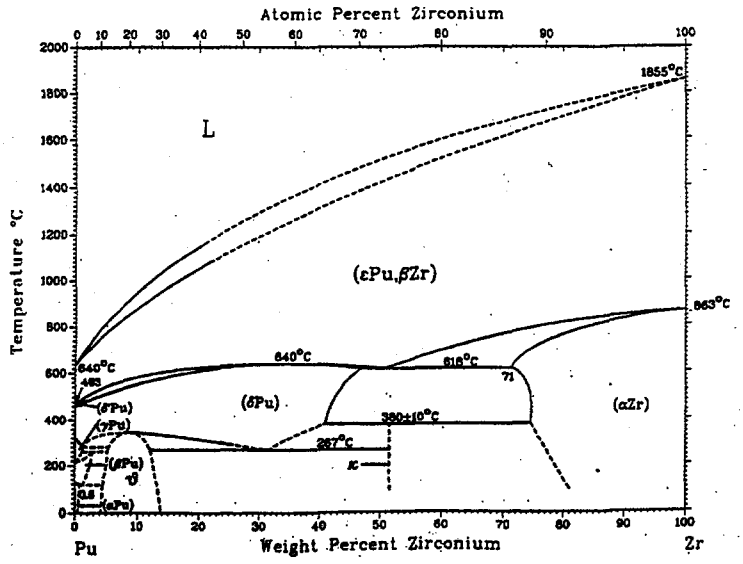

Figure 11. Phase diagram for the zirconium-plutonium binary alloy

3.3.2 Non-fertile Fuel Thermal Conductivity

To evaluate the thermal performance of the non-fertile metal fuel under steady state or transient conditions, it is essential to know its thermal conductivity as a function of temperature and burnup. An estimate of an alloy property can be in principle obtained by taking an appropriate average over its components. Here the thermal conductivity of pure βZr and εPu (see **Figure 11**) are averaged by means of the respective weight fraction to yield the thermal conductivity of the unirradiated fuel at any temperature above 973 K (i.e. 700°C):

$$k = 0.74k_{Zr} + 0.26k_{Pu} = a + b_1 T + b_2 T^2 + b_3 T^3$$

where T is in K and k is in W/cm K.

The value of the numerical coefficients is:

$a = 0.19856$
$b_1 = -1.04439 \times 10^{-4}$
$b_2 = 1.96148 \times 10^{-7}$
$b_3 = -5.00737 \times 10^{-11}$

A benchmark of this method against the experimental values of thermal conductivity of other Pu metallic binary alloys indicates a systematic tendency of this approach to over predict the thermal conductivity with an error of at most 30%.

When the fuel is irradiated, the fission gases fill the pre-existing fuel pores and create new ones. As a result the thermal conductivity decreases. When the number and size of gas filled pores become very large, most pores agglomerate into larger interconnected cavities, which are rapidly filled by the gap bond, leading to a sharp increase of the fuel heat transfer capability.

The pore effect on thermal conductivity (i.e. the burnup effect) can be estimated by means of the following equations [Bauer, 1993]. For gas filled pores only:

$$\frac{k_e}{k} = (1 - P_g)^{(3\varepsilon/2)}$$

where k_e is the thermal conductivity of the irradiated fuel and k is the thermal conductivity of the unirradiated fuel, P_g is the fuel porosity due to gas filled pores and ε is the pore shape factor (ε=1.0 if spherical pores are assumed). The maximum value of P_g (corresponding to the minimum thermal conductivity) was experimentally determined to be 0.25 for the ANL fuel [Bauer et al., 1993]. **Table 5** reports k_e versus the fuel temperature for five different values of P_g.

Table 5. Fuel Thermal Conductivity (W/cm K)

		ε = 1.0 (Spherical Shape)			
T (K)	P_g=0.0	0.05	0.1	0.18	0.25
600	0.195	0.181	0.167	0.145	0.127
700	0.204	0.189	0.183	0.152	0.132
800	0.215	0.199	0.183	0.168	0.139
900	0.226	0.210	0.179	0.178	0.147
1000	0.240	0.222	0.189	0.189	0.156
1100	0.254	0.235	0.201	0.159	0.165
1200	0.269	0.249	0.213	0.200	0.175
1300	0.284	0.263	0.224	0.211	0.184
1400	0.299	0.277	0.237	0.222	0.194
1500	0.314	0.291	0.268	0.233	0.204
1600	0.328	0.304	0.280	0.244	0.213
1700	0.341	0.316	0.270	0.254	0.222
1800	0.354	0.327	0.280	0.263	0.230

3.3.3 Evaluation of Structural and Cladding Materials

The objective of this evaluation is to identify the fuel cladding and structural materials that are compatible with a liquid lead-bismuth cooled, actinide burning fast reactor operated at temperatures from 450 to 650°C, with a cold leg temperature of 450°C, hot leg temperatures up to 600°C, and cladding temperatures of up to 650°C. The liquid lead-bismuth has a eutectic composition (44.5 wt% Pb) which results in a minimum melting temperature of about 125°C.

Iron is present in most structural materials. The solubility of iron in liquid lead-bismuth eutectic varies by a factor of about 20 within the temperature range of interest (450 to 650°C), 1 ppm at 450°C, and 19 ppm at 650°C (Weeks and Romano 1969). The large variation in the solubility introduces a solid-liquid metal interaction in which the principal process occurring is a continuous dissolution of the structural material in the hot regions of the test or reactor loop and precipitation of solid metal in the cold regions. This dissolution/precipitation process is also called liquid metal corrosion.

Past laboratory experiments with a liquid bismuth system (and, therefore, presumably in a liquid Pb-Bi system) have shown that the concentration of the dissolved solid metal in the solution reaches a steady state value throughout the system and is approximately equal to the *solubility of the solid metal (iron) at the lowest temperature* of the solid-liquid interface in the system. This is so because the rate of dissolution of iron into the liquid metal is slower than its precipitation rate. Therefore, the maximum corrosion rate corresponds to the solubility of the solid metal at the maximum interface temperature in the system. The maximum solution rate, R_{max}, can be approximated by

$$R_{max} = \alpha_{Tmax} (S_{oTmax} - S_{oTmin})$$

Where:

α_{Tmax} = solution rate constant at the maximum temperature
S_{oTmax} = solubility at the maximum temperature at the metal-coolant interface
S_{oTmin} = solubility at the minimum temperature at the metal-coolant interface

The above equation can be used to predict the maximum corrosion rate during a test if the value of α_{Tmax} can be determined from the available test results. For example, from the tests carried out at the Brookhaven National Laboratory with T_{max} = 650°C and T_{min} = 500°C, α_{Tmax} was equal to about 2.13 (Weeks and Romano 1969).

The main concerns in selecting the structural materials for a liquid lead-bismuth fast reactor are their resistance to liquid metal corrosion and their radiation stability. The corrosion tests performed during the last 50 years indicate that liquid metal corrosion can be minimized or eliminated by having a corrosion-resistant, self-healing, surface film (consisting of oxides, carbides, or nitrides) on the structural materials at the beginning of a test, and maintaining it during the test.

In addition to liquid metal corrosion, the structural and fuel cladding materials may be subject to liquid metal embrittlement, erosion-corrosion, low-cycle fatigue, and, depending upon the selected material, radiation-induced creep and swelling, and irradiation-assisted stress corrosion cracking. The following discussion presents a review of the technical literature related to liquid metal corrosion. Available information on the Russian experience in using materials resistant to liquid metal corrosion is also discussed.

Corrosion Tests. Corrosion Tests A progression of static and dynamic tests can be performed to evaluate liquid metal corrosion of structural and fuel cladding materials. The dynamic tests have included spinner tests, thermal convection loop tests, and forced convection loop tests (Branover et al. 1994, Asher et al. 1977, Vreeland et al. 1953). Static tests are generally performed under isothermal conditions and are easy to interpret. In addition, these tests are less expensive than dynamic tests, making them ideal for screening a large number of potential candidate materials. Generally, the materials that are corroded in the static tests are likely to corrode more in the dynamic tests.

There are four factors that may affect liquid-metal corrosion in static tests: temperature, the ratio of the specimen surface area to volume of the liquid, the purity of the liquid metal, and the surface condition of the candidate material, such as the presence of grain boundary precipitate (Manly 1958). It is essential that the isothermal condition of the test should be closely controlled.

Static test results may be evaluated using following methods:

- Visual and low-power microscopic examinations of specimen and corrodant.
- Weight change data on specimen.

- Metallographic examination of specimen and sections of specimen for type and extent of corrosion attack
- X-ray and spectrographic identification of surface layers that were formed before test or during test.
- Chemical analysis of the corrodant after the test for content of the specimen metal.

Spinner tests have been performed to evaluate the effects of high relative velocity on corrosion and erosion of various candidate materials. These tests may employ relatively high velocities (450 m/min) achieved by rotating specimens in a pot containing the liquid metal at isothermal conditions (Branover et al. 1994). Thermal convection loops have also been useful in mass-transfer studies. Flow of the molten metal is achieved by heating one leg of the loop, for example, to 600°C and other leg to lower temperature (450°C). Flow velocities of 6-10 ft/min are possible.

Asher et al.(1977) used a simple laboratory scale loop, based on the design of Cathcart and Manly (1954), for containing liquid lead. The loop was made from silica glass, which has proved to be very resistant to attack by molten lead, provided ingress of oxygen is prevented. Such a silica loop may be used if it is also found to be resistant to liquid lead-bismuth. Asher et al. had made some modifications to the loop design so that it can be used for static tests, spinner tests, and thermal-convection tests.

The most realistic type of dynamic test is the forced circulation loop that can be made to reproduce actual operating conditions. The heat is supplied by electric heaters and removed by means of air jets, if necessary, to obtain large temperature gradients. Pumping of liquid metal can be done by means of an electromagnetic pump. The flow velocity in operating reactors can be as high as about 2.5 to 3 m/s. The forced circulation loop should be designed to avoid cavitation at high velocities, where the velocity head may exceed the pressure head.

The factors that affect the liquid metal corrosion rate in dynamic tests, in addition to the ones mentioned for the static tests, include flow velocity and the temperature difference seen by the coolant (Manly 1954).

In addition to the evaluation methods for the static test results, the evaluation methods for the dynamic test results may include following:

- Examination and identification of the crystals that often form plugs in these tests.
- Evaluation of the effects of different velocities on the rate of erosion-corrosion.
- Evaluation of the effects of loop geometry and high flow velocities on cavitation-erosion.
- Study of the effects of high residual and applied stresses on the possibility of liquid metal embrittlement. Asher et al. (1977) accomplished this by applying high tensile stress to rotating specimen in their spinner tests.
- Studies of the effects of cyclic stresses on fatigue resistance

In summary, a complete evaluation of the structural and fuel cladding material for a liquid lead-bismuth cooled reactor might consist of first a static corrosion test; second,

spinner tests; third, thermal convection tests; and finally, forced circulation loop tests. This order of testing is the one in which the severity of the test, the difficulties in the experimental techniques, and cost of operation gradually increase. If a material fails in one of the tests, it has been found by experience that it is likely to fail in the more severe tests that follow.

Corrosion Protection with Surface Films. Corrosion Protection with Surface FilmsTo minimize liquid metal corrosion of structural steel, it is essential that the protective film is stable and easily repairable in the liquid lead-bismuth cooled reactor environment. Our review of the literature has identified the following five requirements for a film at the interface between a specimen and a lead-bismuth melt:

1. The film constituents have large, negative, Gibbs free energies in the operating temperature range.
2. The thermal expansion coefficient of the film matches well with that of the substrate.
3. The film thickness is small.
4. The film is readily reparable.
5. The film has adequate resistance to radiation and fission products.

The first criterion ensures that the film, once it is formed, is stable, whereas the fourth criterion ensures that the kinetics of film formation are fast enough so that the films can be readily repaired. The fourth criterion also implies that a sufficient supply of the constituents of the film (such as O_2, N_2, C, Fe, Si, Zr, Cr) are readily available at the interface. The second and third criteria ensure that the film will not crack easily during operation. The out-of pile corrosion tests discussed earlier can be used to evaluate the first four criteria. In-pile tests will be required to evaluate the fifth criteria.

A film satisfying the above criteria may consist of oxides, nitrides, or carbides of certain elements (such as Fe, Cr, Si) present in the structural materials (ferrous alloys and stainless steels), or of certain other elements (such as Zr, Ti) added externally. As discussed next, the film may be formed prior to the test by an oxidizing, carburizing, or nitriding process, or during the test provided a proper partial pressure of oxygen is present or appropriate inhibitors are added to liquid lead-bismuth pool.

Oxide Films. Iron, chromium, silicon, aluminum, and several other oxides have higher negative free energy of formation than those of the lead and bismuth oxides, as shown in **Table 6.** Therefore, lead and bismuth oxides can not reduce these oxides over the range of temperatures of interest. If an oxide film is present on a steel specimen, oxygen diffusing toward the specimen may strengthen this film; this process is strongly influenced by the temperature and the partial pressure of the oxygen. (If an oxide film is not present, then the oxygen from the liquid metal diffuses into the steel and causes internal oxidation. In addition, the steel may be damaged by liquid metal embrittlement.)

Table 6. Free Energies of Formation of Nitrides, Carbides, and Oxides at 600°C (-ΔG, Kcal/mol of N, C, O)[a] (DeHoff 1993)

Nitride	-ΔG	Carbide	-ΔG	Oxide	-ΔG
ZrN	66.91	Cr_7C_3	51.26	Al_2O_3	313.07
AlN	54.24	ZrC	45.78	ZrO_2	222.11
UN	46.23	TiC	41.28	Cr_2O_3	215.09
CrN	11.06	SiC	14.76	Fe_3O_4	195.94
Fe_4N	-7.83	WC	8.74	SiO_2	178.92
		Fe_3C	-1.76	$FeCr_2O_4$	?

[a] Free energy estimates do not include contributions from changes in heat capacities with temperatures. Some of the free energy estimates vary significantly from those provided by Taylor (1956).

The composition of the oxide film formed on a given steel depends on temperature and the oxygen partial pressure. For example, consider the oxidation of chromium-nickel steel similar to Type 304 stainless steel. At 600°C, both Fe_3O_4 and $FeCr_2O_4$ can form. But at a very low oxidizing potential (very low oxygen content), only $FeCr_2O_4$ (spinel) forms. The spinel provides a much better resistance to liquid metal corrosion and embrittlement as compared to Fe_3O_4 (Bichuya 1969). However, this low oxygen content might not be sufficient for ready repair of the damaged film as discussed next.

The oxide film can provide adequate protection against liquid metal corrosion provided it is readily reparable. Both an adequate oxygen supply and metal ions should be readily available to repair the damaged oxide film. To illustrate the point, we will describe the corrosion test results for Croloy 2 (Asher et al. 1977). A Croloy 2 specimen was oxidized in air at 450°C for 1 hour. The resulting oxide film thickness was about 3000 A. Then part of the surface was abraded to remove a portion of the oxide film. Then it was tested in liquid lead at 700°C for 29 days. Subsequent examination showed that the specimen was heavily pitted. The likely reason for this corrosion was that the lead had a very low oxidizing potential during the experiment since it was continuously being reduced by a hydrogen/argon blanket gas. Thus any flaw in the otherwise protective film would not be self-healing. This situation contrasts with that often found industrially where the molten lead is not completely protected from contact with air or other sources of oxygen. Hence, the lead has a high oxidizing potential and is able to cause healing of flaws in the oxide films on ferrous alloys used to contain it.

The addition of silicon to ferrous alloys may be beneficial because silica (silicon oxide) has a large, negative, free energy. However, sufficient amount of silicon should be present in the steel so that the damaged oxide film can be readily repaired. Asher et al. (1977) tested this idea on Tanitron, a cast iron containing 13.5 wt% Si. The specimen showed no sign of attack and no significant change in weight or in length after exposure to liquid lead at 720°C temperature for 56 days. The following test results suggest that we may not need silicon as high as 13.5 wt%, 1 wt% may be sufficient. Tsirlin et al. (1998) tested Type 316 L stainless steel (1 wt% Si, 10-14 wt% Ni) samples at 500°C in liquid lead covered with nitrogen at a certain partial pressure of oxygen. The oxygen partial pressure was sufficient to form PbO and make the liquid lead saturated with oxygen. This high oxygen content will facilitate rapid formation of an oxide film. Examination of samples at different stages of testing revealed that the concentration of Fe, Ni and Cr near the sample surface did not change during the test period (up to 800 h). At the same time, complex oxide phases (spinels and silicates) were found on the sample surfaces. The formation of these oxides was the reason for the high stability of Type 316 L SS samples in liquid lead and the absence of selective solution of nickel in lead. (Ni has a high solubility in liquid lead compared to Fe and Cr, but the composition of the steel substrate remained unchanged.) The effect of selective solution (leaching) of Ni is observed at a very low activity of oxygen in liquid lead, possibly because of slow kinetics of oxide formation and/or repair.

It appears that 1 wt% Si in a stainless steel composition may be sufficient for the formation and repair of silicate films. The silicone content in the Russian Fe-Cr steel (EP 823), successfully used as structural and cladding material for their liquid Pb-Bi submarine reactors (apparently operated at temperatures below 480°C), is in the range of 1 to 1.3 wt%. The chemical composition of EP 823 is given in **Table 7**. The Russian experience indicates that 10^{-7} to 10^{-9} wt% of oxygen in the liquid lead-bismuth coolant is sufficient to protect this material from liquid metal corrosion. It appears that the oxygen content is kept high so that the oxide film can be formed rapidly at the beginning of use and repaired readily as needed. However, it appears that the high oxygen content leads to formation of lead and bismuth oxides over time, and the reactor coolant has to be exposed to a strongly reducing environment (hydrogen injection) every one to three years to breakdown the lead and bismuth oxides (Chitaykin 1999). Also, proper distribution of the hydrogen requires a coolant flow rate of at least 1 meter/sec, and no flow stagnation at any location within the system. Therefore, the primary system must be a forced (pumped) flow system. Forced flow is also required to provide a uniform oxygen content everywhere in the primary system. Special filters are also needed for removing admixtures that cannot be reduced chemically.

Table 7. Chemical Composition of Potential Candidates for Structural and Fuel Cladding Materials

Material	C	Si	Mn	S	P	W	Cr	Ni	Mo	V	Nb
EP 823	0.14 0.18	1.0 1.3	0.5 0.6	NC	NC	0.5 0.8	10.0 12.0	0.5 0.8	0.6 0.9	0.2 0.4	0.2 0.4
HT-9	0.2	0.2	0.6	0.03	0.04	0.5	12.0	0.5	1.0	0.3	....
436 SS	0.12	1.0	1.0	0.03	0.04	....	16.0 18.0	....	0.75 1.25	...	**

NC - not clear; ** 5 x %C min Nb + Ta

Other candidates for the structural and fuel cladding material are HT-9 steel and ferritic stainless steels such as Type 436 stainless steel. The chemical compositions of these steels are also presented in **Table 7**. There is some reservation about HT-9 steel because its low silicon content (0.2 wt%) may not be adequate to form and maintain a silicate film on a specimen surface exposed to Pb-Bi melt at operating temperatures (450 to 650°C). However, spinel ($FeCr_2O_4$) can be formed and may be sufficient to provide corrosion protection. The HT-9 steel has low Ni, so radiation-induced swelling will not be a concern.

The chemical composition of Type 436 stainless steel is somewhat similar to that of EP 823. However, the absence of Ni raises some concern about its fracture toughness. This needs to be further investigated.

Nitride and Carbide Films. Several nitrides and carbides, especially zirconium nitride (ZrN) and carbide (ZrC), have a higher negative free energy of formation than those of chromium and iron nitrides, as shown in **Table 6**. Advantage of this fact is taken by adding zirconium as an inhibitor in the liquid metal at ppm levels. The added zirconium reacts with the small amount of nitrogen and carbon present in the steel and forms a protective, inert layer of ZrN and ZrC. Which of these films will form on the surface is determined by the relative activities of the C and N in the steel, and these in turn are functions of the steel composition and heat treatment. Thermodynamically, ZrN is appreciably more stable than ZrC; the free energies of formation at 550°C are estimated to be -67 Kcal/mole for ZrN and -46 Kcal/mole for ZrC, as calculated from the data given in DeHoff (1993). Electron diffraction examination of the surface films on uncorroded steel specimens from a large liquid-bismuth loop showed that the principal protective compound is ZrN (Horsley and Maskrey 1958).

Thermal convection loop tests have shown that inhibition cannot be maintained if the Zr concentration is less than 100 ppm. Several liquid-bismuth loop experiments at Harwell have indicated that about 200 ppm Zr was insufficient to repair (reform) the protective film. The maximum Zr concentration that can be used is determined by the solubility of Zr at the lowest temperature of operation.

The solubility of Zr in the liquid lead-bismuth eutectic under isothermal condition and at equilibrium is given by following equation:

$$\log_{10} (\text{ppm Zr}) = 6.15 - (3172/T)$$

over the temperature range of 350-750°C (Weeks and Romano 1969). The temperature, T, is in degree Kelvin. The solubility of Zr at 400 and 650°C is 27 and 516 ppm, respectively. This equation may not be applicable to a forced-flow liquid lead-bismuth cooled reactor system during operation because dynamic conditions rather than static equilibrium and isothermal conditions are present during plant operation. So the effect of plant operation on the solubility of the zirconium needs to be evaluated.

If oxygen is present near the ZrN film, it will react with the film and convert it to tetragonal zirconia, which readily breaks away from the steel surface. Oxygen can come from the melt or through a weld in a steel loop if the weld contains fine pores and cracks along which oxygen can travel to the liquid metal-steel interface. Magnesium may be added to liquid metal to act as a getter (scavenger) for oxygen.

A zirconium nitride film may provide only temporary protection to steel, it is liable to spall off and not reform. This results in localized corrosion attack on the underlying steel. This problem can be addressed by increasing the nitrogen content in the steel from typically 0.1 wt% to 1.0 wt%. This increase in the nitrogen content enhances the corrosion resistance of the film in two ways: (1) it reduces imperfections in the nitride film. The imperfections are due to reactions between Zr and phosphorus, sulfur, carbon, and other nonmetallic inclusions present in the steel. The reaction products reduce the ability of the ZrN film to adhere to the steel surface. (2) It provides a supply of nitrogen for film repair (Horsley and Maskrey 1958). However, nitrogen in the steel takes some time to diffuse to the interface, so it may not be readily available. As an alternate solution, the steel specimens may be nitrided prior to the test. A thin film of Zr is placed on the specimen surface, then the specimen is placed in dry ammonia atmosphere at elevated temperature. This nitriding process ensures that an ample amount of nitrogen is present at the interface and available for ready repair.

The thermal expansion coefficient of ZrN is approximately that of steel between 17 and 680°C; therefore, as the ZrN films become thicker their sensitivity to spalling upon temperature change is expected to increase. Therefore, a ZrN film should be as thin as possible so that it does not break easily, but it should be thick enough to provide adequate corrosion protection.

3.4 Core Thermal-Hydraulics (MIT)

3.4.1 Core Geometry

The thermal analysis of the fuel is based on the core geometry illustrated in **Figures 1** and **2**. Numerical values of the parameters are reported in **Table 8**. The selection of the core geometric characteristics and the fuel composition are mainly driven by neutronics requirements (see Section **3.1**).

Table 8. Core geometry including fuel dimensions, number of fuel assemblies, and composition.

Fuel pellet OD	8.64 mm
Fuel composition (wt%)	26% of Pu+actinides, 74% of Zr
Gap thickness (at BOL)	0.2 mm
Bond material - lead-bismuth-tin alloy (wt%)	33%Pb-33%Sn-33%Bi
Cladding thickness	0.63 mm
Cladding material	Stainless steel EP-823
Pin outer diameter	10.3 mm
Pitch	12.55 mm
pitch to diameter	1.2
Heated core length	1.3 m
Gas plenum height	1 m
Number of fuel assemblies	157
Number of fuel assemblies with control	53
Number of fuel rods per assembly	240
Core barrel inner diameter	4 m

3.4.2 Temperature Limits

The presence of a significant amount of minor actinides makes the selected fuel relatively novel. As discussed in Section 3.3, a simplifying assumption is made: to a first approximation the thermo-physical and thermodynamic properties of the minor actinides can be set equal to those of plutonium. This assumption enables the use of the ANL data on the characteristics of Pu-Zr binary alloys.

Figure 11 illustrates the phase diagram of the Pu-Zr alloy. It can be seen that the alloy melting point (along with its crystalline stability) significantly increases with the Zr weight fraction: at 74wt% Zr, the melting point is approximately 1600°C. Somewhat below this value is then the fuel temperature design limit. Since the fuel-clad gap is designed to allow for partial fuel restructuring, the phase transition at approximately 700°C does not set a serious limit on the fuel temperature.

Review of the existing literature on the stainless steel EP-823 corrosion by Pb-Bi indicates that the maximum allowable temperature is approximately 650°C. This is then the cladding outer surface temperature limit tentatively selected in this project. Corrosion

of EP-823 is inhibited by the formation and maintenance on the steel surface of an oxide layer whose integrity significantly depends on the Pb-Bi temperature, flow rate, and oxygen content as well as the minor alloy constituents in the cladding, especially the silicon and vanadium concentrations. The lack of Western experimental data and theoretical study on the physics of this oxide layer make the above temperature limit subject to uncertainty.

Moreover, because the thermo-physical properties of EP-823 are not readily available, they were set equal to those of HT-9, whose composition resembles it (but with less silicon, see **Table 7**).

3.4.3 Basic Assumptions

The variation of the fuel thermal conductivity with burnup displays a minimum at 25% porosity. The mechanism leading to a minimum is briefly described as follows: the thermal conductivity initially decreases with burnup as the released fission gases increase the fuel porosity. When the number and size of gas filled pores become very large (i.e. above 25%), most pores agglomerate into larger interconnected cavities, which are rapidly filled by the gap bond, leading to a sharp increase of the fuel heat transfer capability. To avoid computational complications, the thermal conductivity of the fuel is conservatively assumed to be constant with burnup and equal to this minimum value. For a theoretical estimate of the fuel thermal conductivity, refer to Section **3.3**. Convective effects in the gap are neglected hence heat is transferred through the thin liquid metal bond by conduction only.

Several correlations developed for liquid metal flow in rod bundles were considered for the calculation of the Pb-Bi heat transfer coefficient on the outer surface of the clad. The Westinghouse correlation [Kazimi and Carelli, 1976] consistently yields the most conservative values of the heat transfer coefficient and it was then chosen for this analysis.

Depending on the primary system configuration and on the reactor thermal power, the average coolant velocity in the core ranges from 0.5 to 2.5 m/s (see Section **3.5**). If the total core thermal power is assumed to range from 1500 to 2000 MWth, the average linear heat generation rate in the fuel pins ranges from 31.2 to 41.6 kW/m. The radial and axial power peaking factors are assumed to be 1.17 and 1.6, respectively (see Section **3.1**). Therefore the peak linear heat generation rate ranges between 58.4 and 77.9 kW/m, corresponding to 1500 and 2000 MWth, respectively. If the control rods are inserted from the core bottom[4], the axial power peak takes place at the core exit and the highest fuel and clad temperature are expected here. The coolant bulk temperature at the core exit is assumed to be 550°C because it will be demonstrated (see Section **3.4**) that core outlet temperatures about this value are necessary to obtain effective natural circulation in the primary system.

[4] This might be the preferred location due to the buoyancy of control materials in the heavier Pb-Bi coolant.

3.4.4 Results and Discussion

In **Figure 12** the fuel centerline temperature and the clad outer temperature of the hot channel are plotted as a function of the local linear heat generation rate for three different values of the coolant velocity.

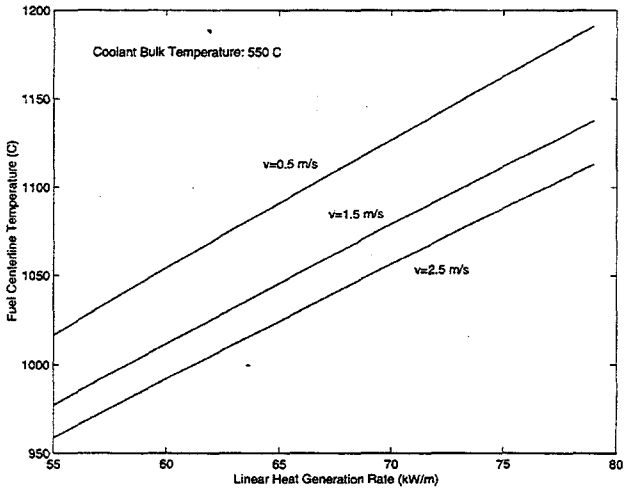

Figure 12a

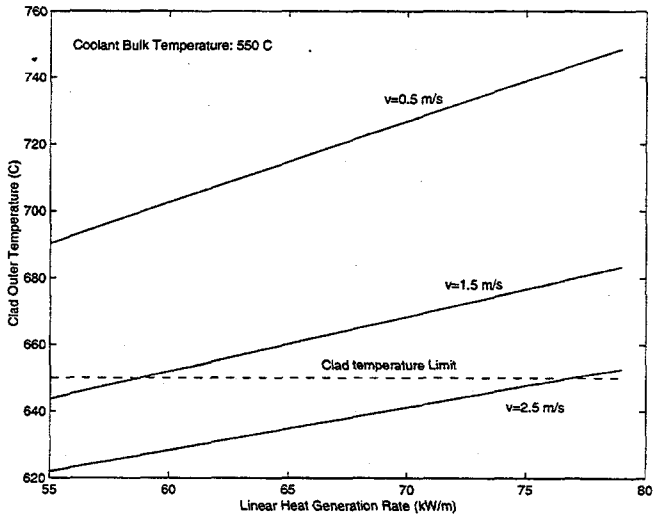

Figure 12b

Figure 12. Fuel centerline temperature (Figure 12a) and cladding surface temperature (Figure 12b) as a function of linear heat generation rate at three coolant velocities.

The maximum temperature within the fuel is considerably below the fuel melting point (i.e. 1600°C) under all circumstances. As expected, the cladding sets the most severe limit: the temperature on the outer surface of the clad appears to be unacceptably high for Pb-Bi velocities smaller than 2.5 m/s. This value of the coolant core velocity can be attained only if:

a) the primary coolant is actively pumped through the core, or

b) a gas-pump scheme is adopted (see Section **3.5.6**), which enables achievement of very effective passive circulation in the primary system.

In any case a factor of 1.6 axial power peaking (especially if occurring at the core outlet) is too large and means to reduce it should be explored.

3.4.5 Control Rod Cooling

The control rods are heavily irradiated by fast neutrons and gamma radiation, which will generate considerable heat. Additional heat is deposited as a consequence of the neutron absorption reactions. The magnitude of heating mainly depends on the material scattering and absorption cross sections (for the neutron interactions) and on its atomic number and density (for the gamma interactions). If adequate cooling is not provided, the control system may be unable to appropriately perform its function hence degrading the overall safety of the reactor.

Let us distinguish between two different sets of control rods.

1) The control rods for fine reactivity control have the main purpose of compensating for reactivity decrease with burnup in order to maintain criticality. At BOL these rods are well inserted in the fuel and they are subjected to large heat deposition. If they are immersed in the primary coolant, the heat is effectively removed by convection and conduction. However, a concern exists regarding the deposition of corrosion products and other debris (normally present in the primary coolant) on the control rod driving mechanisms that would decrease the system reliability. To mitigate this concern, it may be desirable to place these control rods in empty channels. In this case the dominant heat removal mechanism is radiation, which does not appear to be effective enough to keep the control rod temperature within acceptable limits.

2) The scram control rods are located above the core. To minimize the probability of scram failure, these rods are placed in empty channels that ensure rapid gravity insertion into the core. Under normal operating conditions, they are kept outside the core. As a result, the heat deposition is small.

Several neutron absorbing materials were explored as potential candidates for the control system. Special attention was given to B_4C because of its relatively large absorption cross section at high neutron energy and to hafnium because of its absorption

resonances that may be used to improve the doppler reactivity feedback. Gamma and neutron heating of control rods made of these two materials were computed by means of the MCNP code. The linear rate of heat deposition ranges from 4 kW/m to 10 kW/m depending on the material (i.e. higher gamma heating in Hf, higher neutron heating in B_4C) and on the location of the control rods. It would be extremely difficult for radiation cooling to remove such a high linear power. As a result, the fine reactivity control rods will likely have to be kept immersed in the primary coolant.

3.5 Primary System Concept and Vessel (MIT)

The reactor core and the Pb-Bi/He heat exchanger are integrated in a pool design. The pool configuration minimizes flow resistance, making natural circulation more feasible, and it offers a better thermal inertia that will moderate the temperature rise that occurs upon loss of the heat sink. Additionally, it facilitates maintenance of the coolant temperature above its freezing point (~125 °C) and helps achieve a more compact layout of the plant.

A reference schematic of the reactor pool is illustrated in **Figure 13.** The gravitational head H is a design variable mainly driven by natural circulation considerations. For the geometric characteristics of the heat exchanger, see Section 3.6.2.

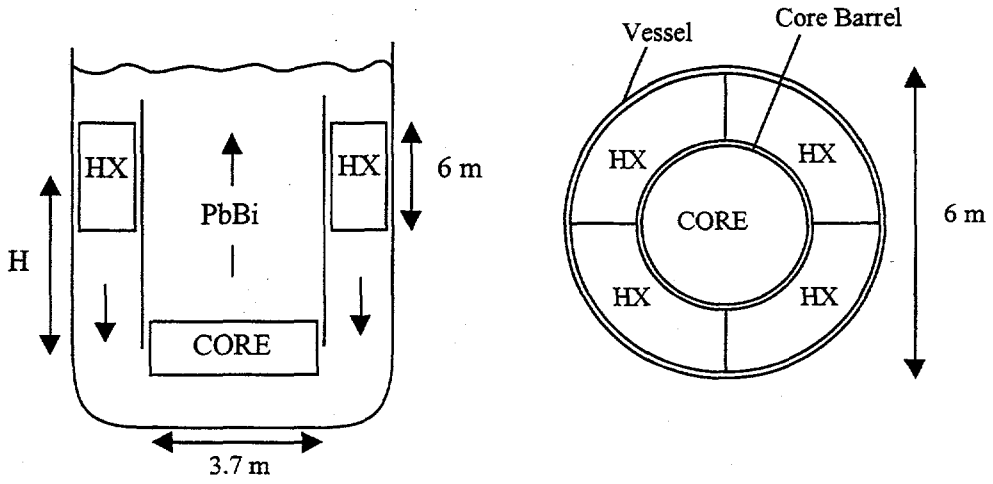

Figure 13. Sketch of the reactor pool and heat exchanger geometry.

3.5.1 Natural Circulation Analysis

The effect of the pitch-to-diameter ratio and gravitational head on the average power density removable by means of natural circulation are explored for the following reference conditions (consistent with the core geometry proposed in Section **3.1**):

- PbBi core inlet/outlet temperature: 250°C/550°C.
- Fuel pins: OD=10.3 mm, Active Length=1.3 m (+1.0 m gas plenum).
- Fuel pins arranged on a square lattice.
- Three spacing grids together with the stiffening provided by the voided streaming tubes shown in Figure 1 to ensure proper separation of the fuel pins.

Figure 14 illustrates the relevant results. The analysis shows that:

a) For relatively large p/d, average power density comparable to sodium cooled fast reactors (up to 50% of French Superphenix) can be achieved, which proves the excellent natural circulation potential of the lead-bismuth eutectic.
b) A core with diameter limited to 4 m can be designed (at 1,800 MWth).
c) Average coolant velocity in the core ranges from 0.5 to 1.0 m/s. In this velocity range the heat transfer coefficient varies between 10 and 12 kW/m^{2}K.
d) Average linear power up to 40 kW/m is obtained.

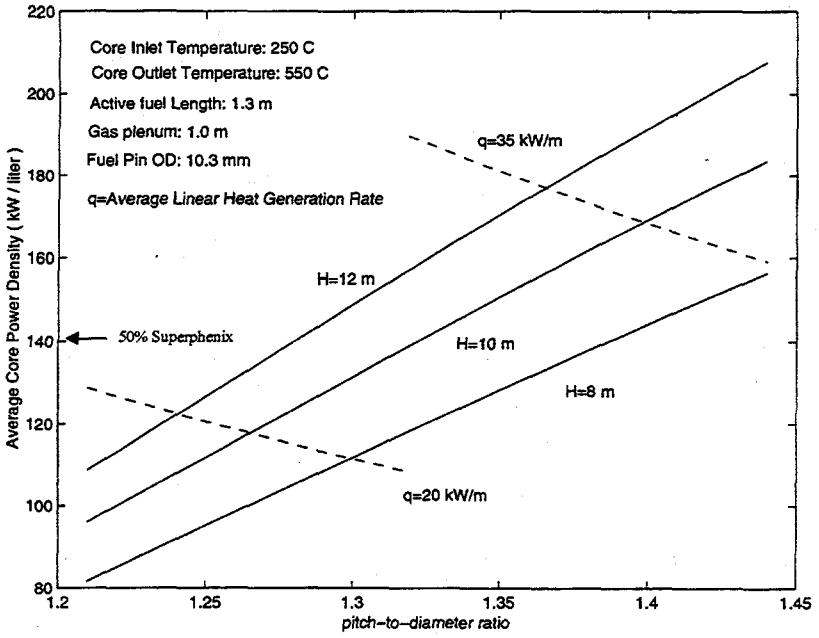

Figure 14. Average core power density as a function of pitch-to-diameter ratio and gravitational head for an active fuel length of 1.3 m.

On the other hand:

a) The neutronic stability of the core seems to favor a very tight lattice (p/d around 1.2), which would deteriorate the natural circulation capability of the system.

b) Given the large axial power peaking, an average linear power above 40 kW/m implies peak linear powers above 60 kW/m. It was shown in Section 3.4.4 that a high Pb-Bi velocity (well above 1.0 m/s) is needed to remove this magnitude of thermal load.

3.5.2 The Effect of Fuel Length on Natural Circulation

To achieve a large negative void reactivity coefficient, the fuel length can be reduced to increase the axial neutron leakage upon voiding. Alternatively, the core diameter can be greatly reduced to increase radial neutron leakage upon voiding. In this case, if the total thermal power is to be kept constant, the fuel active length must be increased leading to a core shape that significantly differs from the reference one (i.e. cigar shape core vs. pancake shape core). **Figure 15** illustrates the power density that can be removed by natural circulation if the fuel length is increased to 3 m. **Figure 16** illustrates the simple geometrical relation between the power density and the core diameter for a 1,800 MWthh total power.

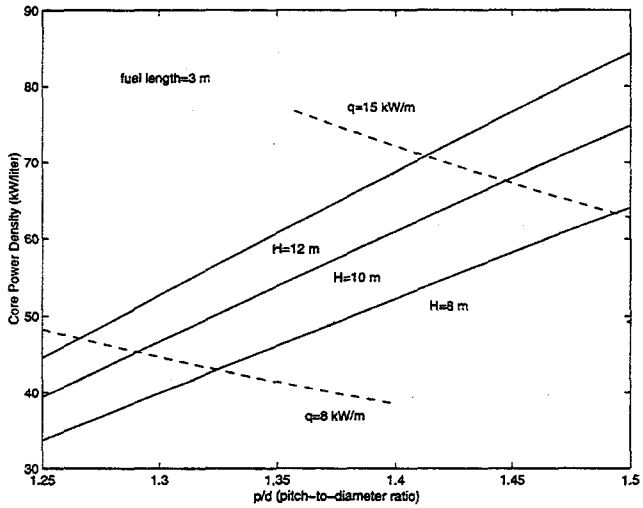

Figure 15. Average core power density as a function of pitch-to-diameter ratio and gravitational head for an active fuel length of 3 m.

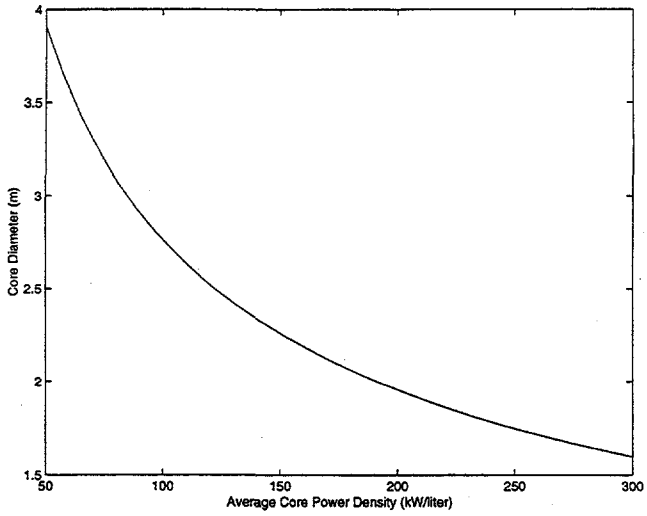

Figure 16. Relationship between power density and core diameter for a 1,800 MWth core.

The results clearly show that the cigar shape core cannot be designed to operate on natural circulation only. The achievable power density is rather small and requires a large core diameter, which defeats the initial purpose of increasing the radial neutron leakage.

3.5.3 Core Voiding as a Result of Secondary Coolant Entrainment upon Rupture of a Heat Exchanger Tube

In Section 3.1 it was shown that the proposed core configuration achieves a negative reactivity coefficient when the coolant is totally or partially displaced from the core. Although this feature greatly mitigates the consequences of coolant voiding, it is also of interest to assess the likelihood of occurrence of this event. When a tube of the heat exchanger fails, the secondary coolant is discharged at high velocity into the reactor pool. A simple (although very conservative) way to evaluate the maximum vertical penetration of this fluid within the Pb-Bi and therefore to assess whether gas can reach the core inlet, is to set the velocity head of the discharged secondary coolant equal to the static pressure of the primary coolant as illustrated in **Figure 17**.

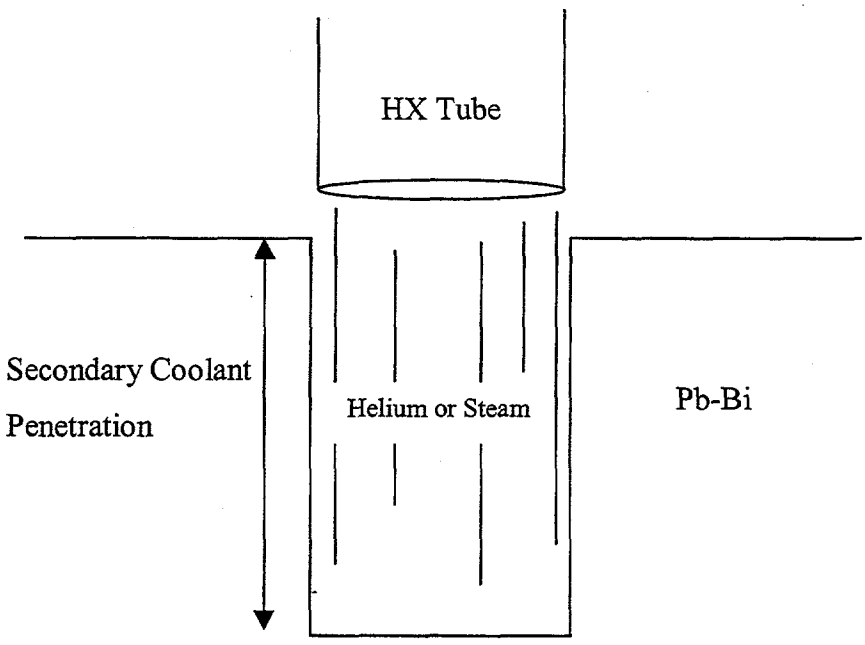

Figure 17. Depth of secondary coolant penetration.

For helium at 7.0 MPa and for steam at 7.0 MPa, the penetration in the Pb-Bi pool is estimated to be 25 and 18 m, respectively, which is greater than the downcomer length of approximately 15 m.

Clearly the possibility of core voiding cannot be ruled out on the basis of this overly simple model that ignores the effects of viscosity and the two dimensional nature of a gas jet injected into a stagnant liquid. It may be possible to rule out this scenario using a more realistic estimate of penetration distance from a more sophisticated fluid dynamic model.

3.5.4 Vessel Structural Analysis

The reactor vessel constitutes one of the main barriers to the release of radioactivity. It is essential to ensure that the design temperature limit of the reactor vessel is not exceeded under normal or accident conditions. The vessel material is assumed to be EP-823 stainless steel, whose mechanical characteristics are inferred from those of HT-9. The structural analysis satisfies the requirements of the ASME code case N-47. **Figure 18** shows the HT-9 allowable design stress intensity vs. temperature curve. The dependence of the design stress intensity on the operating time is a consequence of the importance of thermal creep at high temperature (above 450°C).

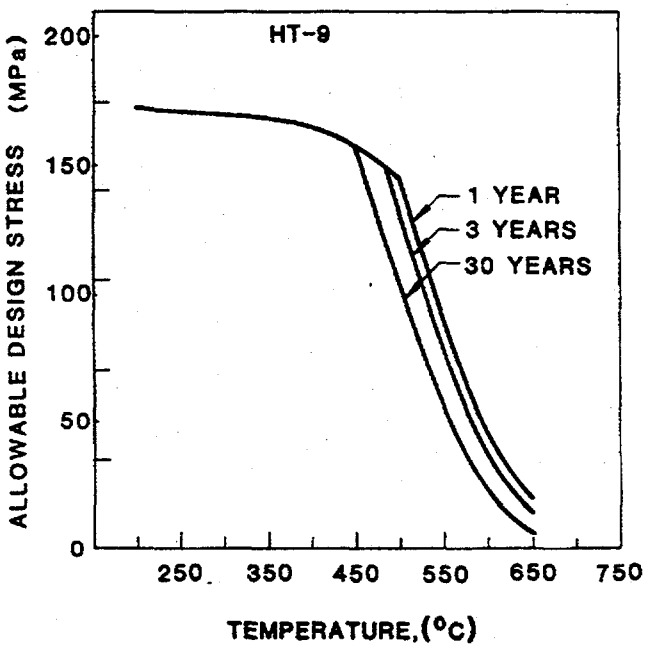

Figure 18. Allowable design stress versus temperature for HT-9 stainless steel.

The vessel is subjected to the following loads:

a) Operating pressure. Due to the high density of PbBi, the operating pressure varies linearly from 0.1 MPa (i.e. atmospheric pressure) at the vessel top to approximately 1.8 MPa at the bottom of the lower head.
b) PbBi and vessel weight.
c) Radial temperature gradient across the vessel thickness (especially when the decay heat is discharged through the vessel, see Section **3.5**). It should be noted that in this structure the radial temperature gradient induces hoop and axial thermal stresses whose average over the vessel thickness is zero. These stresses have to be taken into account in a fatigue analysis only. Since the expected frequency of use of the Decay Heat Removal System is very low, fatigue considerations are relatively unimportant and to a first approximation the thermal load can be neglected.
d) Fast neutron flux. Depending on the downcomer thickness (from 50 to 100 cm), the fast neutron flux (E_n >1MeV) at the vessel surface can range up to 1.8×10^{12} n/cm^{2}s, leading to a total fluence over 30 years of vessel lifetime up to 1.7×10^{21} n/cm^2. The corresponding dose effect is 1.7 dpa.

The vessel beltline thickness is varied between 3 and 10 cm (with a reference value of 5 cm). The thickness of the lower head is selected to be 2/3 of the beltline thickness to minimize the discontinuity stresses. Thin shell theory is applied to find the primary membrane stresses, the discontinuity membrane and bending stresses, the elastic, thermal and irradiation creep strains and displacements. The results of the analysis show that irradiation creep and swelling are negligible. Irradiation embrittlement is expected to shift the Ductile-To-Brittle-Temperature by at most 10°C.

As expected, the mechanical stresses (induced by the operating pressure and the structure weight) decrease as the vessel thickness increases. Since the strength of the material is a decreasing function of temperature (as shown in **Figure 18**), lower stresses enable operation of the vessel at higher temperature: the vessel temperature limit as a function of the vessel thickness is illustrated in **Figure 19**. The design point tentatively selected is 5 cm, corresponding to a maximum allowable temperature of approximately 530°C.

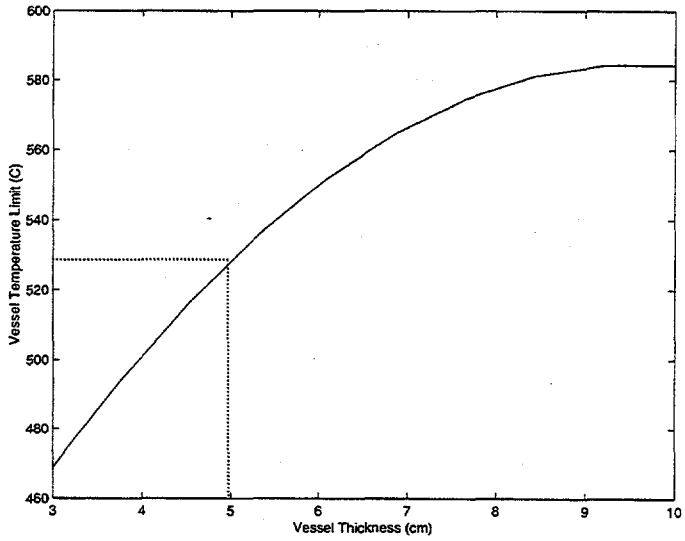

Figure 19. Vessel temperature limit as a function of vessel thickness.

3.5.5 Primary Coolant Capital Cost

Bismuth is a relatively expensive material. Its impact on the cost of the electricity produced by this reactor was evaluated.

Assuming a reactor pool of 15 m height and 6 m diameter, 75% filled with primary coolant, the required Pb-Bi inventory is approximately 3,030 metric tons. Assuming a plant net electric output of 600 MWe, the coolant mass per unit electric power is approximately 5 ton/MWe. The market cost of bismuth is around 7 $/kg and that of lead is around 1 $/kg. The lifetime of the coolant is assumed to be 25 years. The after-tax averaged cost of capital is 8%. Disposal cost or salvage value are not considered.

Table 9 compares the capital cost of lead-bismuth coolant and pure lead coolant for two different values of the capacity factor f.

Table 9 Coolant levelized capital cost (mills/kWh)

	f=0.6	f=0.9
Pb-Bi	0.53	0.35
Pure Pb	0.12	0.08

As can be seen, the cost of PbBi is not negligible. Nevertheless, it is approximately a factor of 5 less than the cost of the D_2O inventory for a CANDU reactor.

3.5.6 An Innovative Primary System Configuration

In this section a possible alternate scheme of the reactor pool is presented that relies on a direct steam cycle produced by direct contact heat exchange between water and lead-bismuth. **Figure 20** illustrates a schematic of the proposed concept.

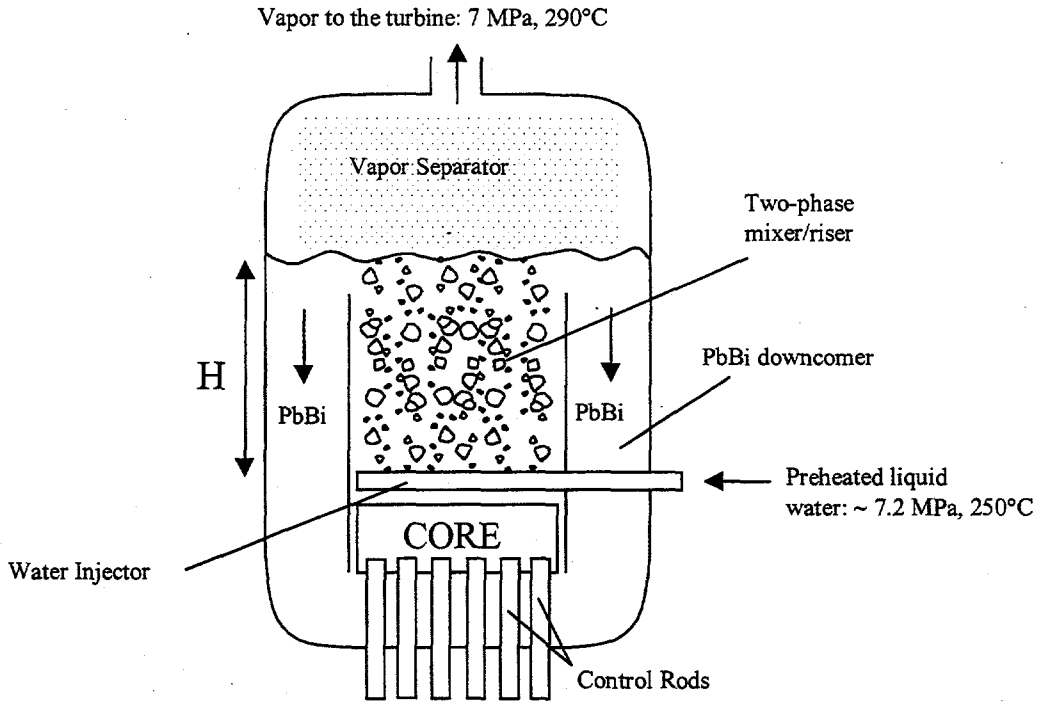

Figure 20. Direct steam cycle produced by injecting water into molten lead-bismuth.

Subcooled water is injected in the hot lead-bismuth above the core. The direct contact between the fluids causes water to rapidly vaporize leading to the formation of voids in the hot leg. The vapor is separated in the upper part of the reactor vessel and sent to the turbine. The large density difference between the cold and the hot leg drives the natural circulation of Pb-Bi in the vessel. The secondary coolant operates a Rankine cycle identical to a BWR that can achieve thermal efficiencies above 30%. The required vessel is within the design envelope of the ESBWR and operates at 290°C.

The direct heat transfer of Pb-Bi with water has been investigated in Israel for over a decade: an extensive experimental database [Branover, 1999] and the computational tools to predict the behavior of PbBi-water systems are available.

Some advantages of this innovative reactor design over the more traditional one introduced earlier are identified:

a) Preliminary calculations show that full power natural circulation (even at very low core p/d) can be easily achieved in this system with the following benefits:

   - A significantly shorter reactor pool (10 m vs. 20 m) can be designed. A shorter pool makes the reactor overall size smaller and requires a substantially lower inventory of Bi hence cutting the capital costs.

   - The maximum Pb-Bi temperature can be kept relatively low (<500°C vs. >550°C); see **Figure 21**.

   - An average Pb-Bi core velocity above 2 m/s is achieved, which enables removal of high linear heat generation rates in the fuel (see 3.4.2).

b) Elimination of the pumps and heat exchangers makes the operation more reliable.

c) Preliminary MCNP simulations indicate that the thermal-hydraulics of the two-phase hot leg and the core neutronics are decoupled, which may considerably decrease the concern of flow and neutronic instabilities.

d) Preliminary MCNP simulations indicate that accidental flooding of the core with liquid water does not cause reactivity to increase.

On the other hand new challenges arise:

a) The primary system needs to be pressurized, which increases its cost.

b) The consequences of mixing Pb-Bi and water in terms of chemistry control of the primary system are not clear.

c) A small Pb-Bi inventory may be undesirable if a large amount of decay heat is to be initially stored in the reactor pool upon loss of the normal heat sink (see 3.5.2).

d) The inclusion of separators into the primary vessel will complicate reactor refueling.

e) Direct contact of Pb-Bi and steam significantly aggravates the issue of polonium contamination.

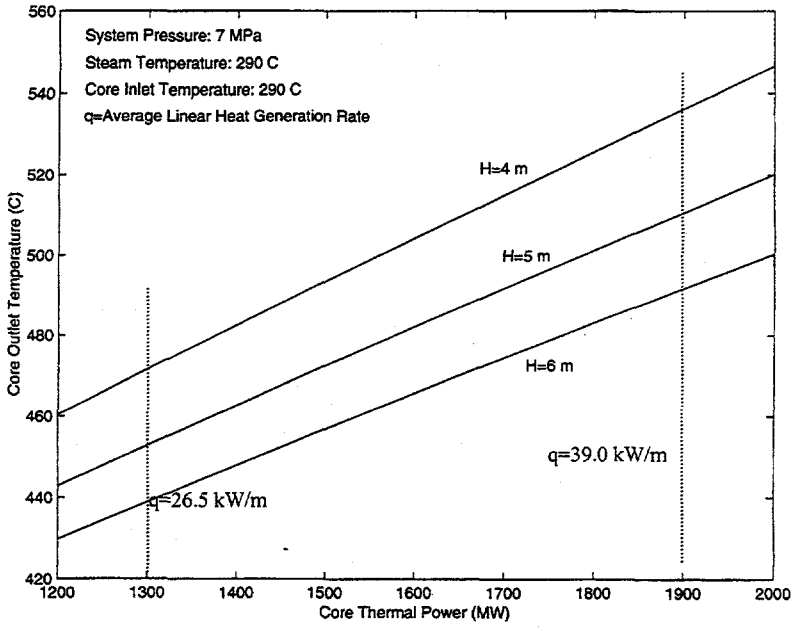

Figure 21. Core outlet temperature versus core power and gravitational head in a direct steam cycle reactor.

The latter problem is felt to be the most serious limitation of this design. The rate of formation and release of Po and its compounds (i.e. mainly PbPo and H_2Po) in steam is directly proportional to their concentration in the Pb-Bi coolant.

A first evaluation of the magnitude of Pb-Bi activation shows that the amount of Po in the reactor can be substantially reduced by continuous purification of a small fraction of the primary coolant inventory (see **Figure 22**). When created by neutron activation, polonium rapidly forms a stable compound with lead (i.e. PbPo). Upon contact of this compound with water a very volatile polonium hydride (i.e. H_2Po) is formed, which entrains steam hence contaminating it. A parallel mechanism of Po release into steam is direct evaporation of PbPo. The magnitude of Po release associated with both mechanisms will be assessed.

Some small leakages of contaminated steam out of the turbine are inevitable and will be a radioactive hazard. It must be demonstrated that the polonium concentration of the leaking steam is within prescribed limits (10 Bq/m^3).

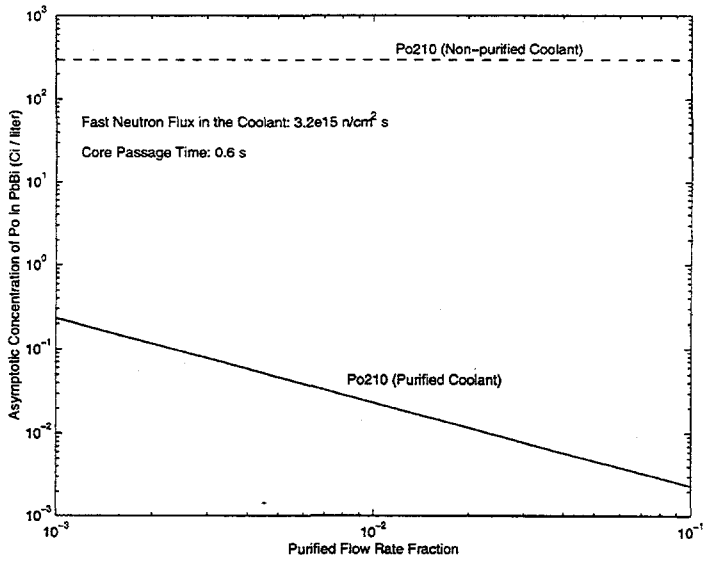

Figure 22. Concentration of polonium in the lead-bismuth coolant as a function of purification flow rate

3.6 Decay Heat Removal System (MIT)

3.6.1 Decay Heat after Shut-Down

Accurate knowledge of the decay power in the first few hours after shut down is essential to design the decay heat removal system (DHRS) and to predict the consequences of most accidents. In **Figure 23** the decay heat rate of the actinide-burning reactor (indicated as "trans") is compared to that of a typical PWR for several values of the burnup. As expected, the difference appears to be within the inherent uncertainties of the model used to calculate the decay power.

3.6.2 Steady-State Analysis

After the reactor has been shut down, the heat produced by the radioactive decay of the residual fission fragments poses a serious threat to the integrity of the nuclear fuel. Normally this heat is removed from the reactor pool through the primary-to-secondary-coolant heat exchanger, but means must be provided to ensure continuous cooling of the nuclear fuel should that heat removal path fail. To increase the reliability of the emergency decay heat removal, it is desirable to select a system that operates passively.

The heat can be removed by means of dedicated heat exchangers located in the reactor pool (as in most sodium cooled fast reactors) or it can be discharged to the environment through the reactor vessel.

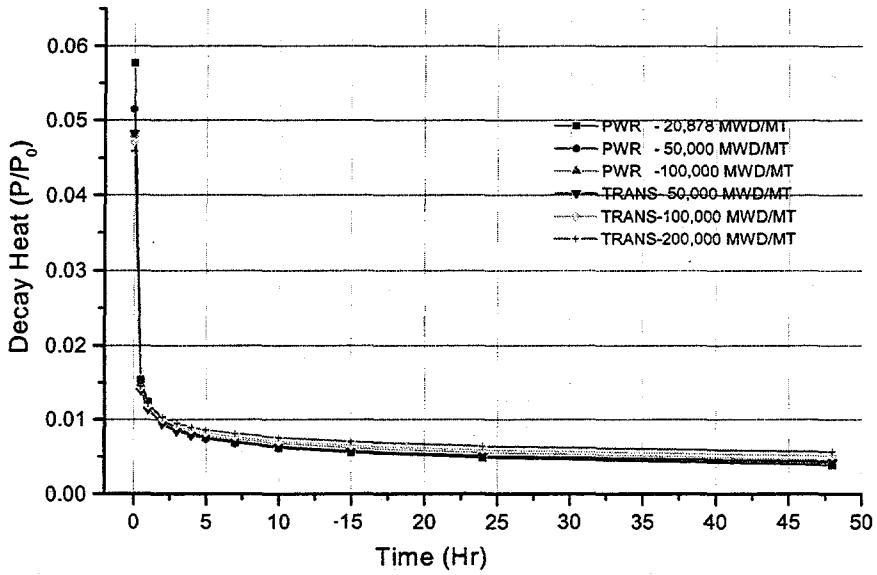

Figure 23. Comparison of of the decay heat in a pressurized water reactor at various burnup levels with the decay heat in an actinide-burning reactor.

Only the latter case has been so far investigated in this project. Two basic configurations are shown in **Figure 24** and were considered:

a) The decay heat is removed by natural circulation of air on the outer surface of the reactor containment.

b) The decay heat is removed by boiling water at the outer surface of the reactor containment.

Design a) is supported by the previous experience of the advanced sodium-cooled reactor project undertaken by GE [GE, 1991], to which we refer for its general technical characteristics. However, it should be noted that the GE system was conceived for a reactor of rather small power and some changes are then needed to make it suitable to our larger reactor design.

The heat produced in the core is conveyed by naturally circulating Pb-Bi to the surface of the reactor vessel. The gap between the vessel and the containment vessel can be filled either with an inert gas (e.g. nitrogen) or with a liquid metal (e.g. Pb-Bi or Pb). In the first case the heat is transferred through the gap mainly by radiation and large temperature gradients are expected. On the other hand, if a metal bond is employed, conduction and convection make the gap thermal resistance very small.

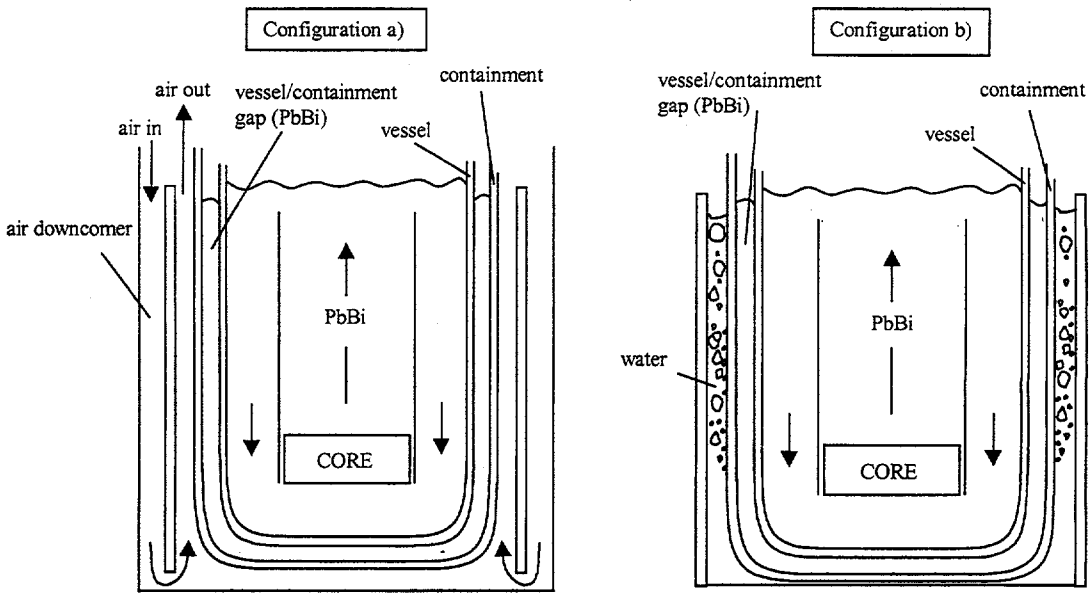

Figure 24. Reactor schematics for two approaches for removing decay heat; natural circulation of air on the outside surface and use of boiling water on the outside surface.

In configuration a) air enters the reactor building through two inlets and flows through the downcomer to the containment bottom. From here it flows upward in the riser where it is heated and it is finally discharged to the atmosphere through four stacks that supply the gravitation head needed to passively drive the air through the circuit[5].

In configuration b) the annular region around the containment is flooded with water. If the heat flux at the outer surface of the containment is large enough, boiling occurs and the produced vapor must be vented.

Tables 10 and 11 report the geometric characteristics of the vessel, the containment and the air system.

Table 10. Reactor Vessel and Containment

Vessel Outer Diameter (m)	6.0
Vessel Thickness (cm)	5.0
Vessel Height (m)	15
Vessel/Containment Gap (cm)	5
Containment Thickness (cm)	2.5

Table 11. Air System

Inlet Temperature (°C)	37
Stack Height (m)	30
Inlets	2
Outlets	4
Cold Air Tube Length (m)	30
Cold Air Tube Diameter (m)	2
Hot Air Tube Length (m)	40
Hot Air Tube Diameter (m)	3
Riser Thickness (cm)	12

[5] In Figure 24 the air inlets and exhaust stacks are not shown.

In **Figure 25** the performance of configurations a) and b) are compared for the case of vessel/containment gap filled with Pb-Bi. The primary coolant maximum temperature is plotted versus the decay power expressed as a fraction of the nominal thermal power of the core (assumed to be 1,800 MWthh).

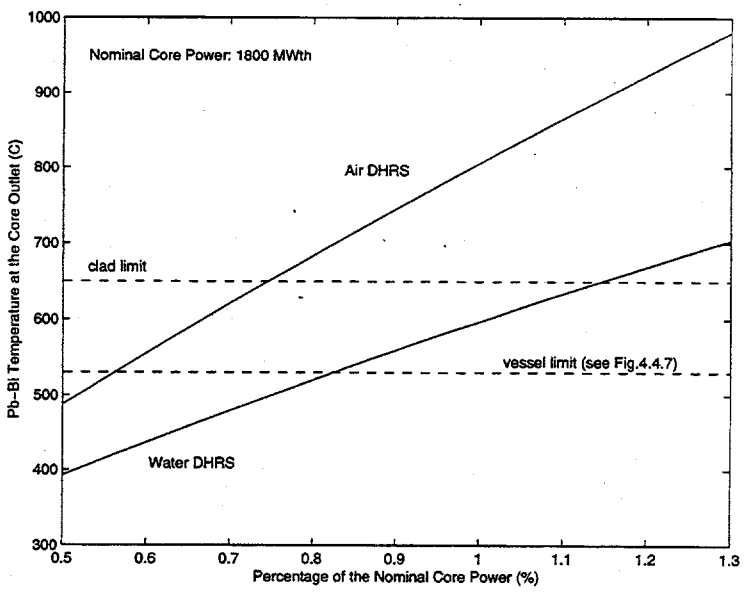

Figure 25. Maximum primary coolant temperature versus the decay power for two decay-heat-removal configurations.

As expected, the DHRS based on boiling water consistently provides better heat removal. The reactor vessel[6] temperature limits (see Section 3.5.4) are exceeded at 0.55% of the nominal core power for configuration a) and at 0.85% for configuration b). However, it can be shown that if the vessel/containment gap is filled with nitrogen, it is impossible to keep the Pb-Bi maximum temperature within acceptable limits at any power level of practical interest. Thus the vessel/containment gap must be filled with a thermal bond (i.e. Pb-Bi or Pb).

These results also show that, when the reactor is normally operating, it is possible to let the DHRS run without dissipating too high a power. In case of sudden loss of the normal heat sink, this ensures immediate removal of the decay heat and reduces the risk of the reactor coolant overheating. Nevertheless, it may prove preferable to operate with a dry containment vessel pit and only flood with water when the occasion demands.

[6] Here the vessel thickness is assumed to be 5 cm.

3.6.3 Reactor Pool Thermal Inertia

Depending on the primary system configuration the average Pb-Bi temperature under normal operating conditions can range between 300 to 400°C. Assuming an average Pb-Bi density of 10,200 kg/m^3 the coolant inventory in the reactor pool is approximately 3,030 tons. The coolant specific heat is 146.5 J/kg°C.

In Section 3.6.1 it was shown that the maximum decay power removable at steady state by the DHRS is approximately 15.3 MW (i.e. 0.85% of the nominal reactor power). It takes about 3.5 hours before the decay power falls to this level. Even though the DHRS is activated immediately, during this time the decay heat release rate is not matched by the rate of heat removal and the average temperature of the primary coolant in the reactor pool increases hence reaching a maximum at 3.5 hrs after shutdown. It is important to ensure that this maximum temperature does not exceed the vessel temperature limit (i.e. ~ 525°C for a 5-cm thick vessel). An energy balance over the reactor pool shows that the Pb-Bi temperature rise at 3.5 hrs is approximately 125°C, hence leading to a maximum temperature of 525°C.

On the other hand, if the DHRS activation is not instantaneous (e.g. DHRS is activated 3.5 hours after shutdown), all the decay heat is accumulated in the primary coolant causing a clearly unacceptable temperature rise of about 635°C !

One way to reduce this figure is to fill the vessel/containment gap with solid Pb and to allow it to melt when the reactor is shut down hence absorbing a significant fraction of the decay energy released in the core. Assuming even a 30 cm vessel/containment gap entirely filled with pure solid lead, the primary coolant temperature rise after 3.5 hrs is approximately 445°C, which is still too high.

Therefore it seems crucial to guarantee that the DHRS can be activated almost immediately after shut down (i.e. within few minutes). In this case 15.3 MW are constantly removed and it appears to be possible to maintain the primary coolant temperature within acceptable limits.

3.7 Power Cycle (MIT)

3.7.1 Temperature Constraints

The thermal efficiency of the power cycle strongly depends on the temperature at which the heat is supplied by the primary to the secondary coolant. It is clear that in principle a high cycle maximum temperature is desirable. In practice the mechanical and corrosion characteristics of the core, vessel and heat exchanger materials set the limit (see Section **3.3**). The interim selected cladding material (i.e. the Russian steel EP-823) exhibits acceptable corrosion resistance to the lead-bismuth eutectic for temperatures up to 650°C[7]. If a typical 100°C temperature drop across the Pb-Bi boundary layer is

[7] However, the mechanical strength of this material rapidly decreases with temperature above 500°C.

assumed (see Section **3.3**), the primary coolant bulk temperature limit is around 550°C. Assuming a further 50°C temperature drop across the heat exchanger, the maximum temperature achievable in the power cycle seems to be approximately 500°C.

Also the choice of the secondary coolant temperature at the inlet of the heat exchanger requires some attention. It must be sufficiently above 125°C (i.e. the freezing point of Pb-Bi at atmospheric pressure) to provide margin to undesirable freezing of the primary coolant during transients. At the same time, it must be smaller than the minimum Pb-Bi temperature in the primary system (which in turn is determined by natural circulation requirements, see Section **3.4**). In practice, the available design window for this temperature is 125 to 250°C, which requires preheating of the working fluid: this is done by means of the regenerator in a gas cycle and by means of the feedwater heaters in a steam cycle.

Finally the cycle minimum temperature is limited by the conditions of the ultimate heat sink (i.e. typically seawater or air at 20 to 30°C).

3.7.2 Gas Turbine Cycle

Gas cycles are generally simpler, more compact than steam cycles and offer potential for more automated operation. In our case helium is chosen as the working fluid for its chemical inertness and its relatively high thermal conductivity. However, while significant experience has been accumulated on air gas turbine cycles, only little relatively has been accumulated on helium cycles.

Figure 26 illustrates the schematic and the T-s diagram of the Brayton cycle of choice. Regenerative heat transfer between the turbine outlet stream and the high pressure compressor outlet stream increases the cycle thermal efficiency and helps achieve higher temperatures at the inlet of the PbBi/He heat exchanger hence complying with the requirement discussed in **Section 3.7.1**. The use of inter-cooling increases the net work per unit mass and widens the regeneration temperature window. An alternate configuration with two inter-coolers is also explored.

Re-heating of the hot helium was discarded to keep the design of the PbBi/He HX simple.

The turbine and compressor efficiencies and the regenerator effectiveness are assumed to be 92, 92 and 65%, respectively. The cycle minimum temperature is set equal to 50°C. The cycle maximum pressure is set equal to 7.0 MPa. Several values of the cycle compression ratio are explored with the goal of maximizing the thermal efficiency.

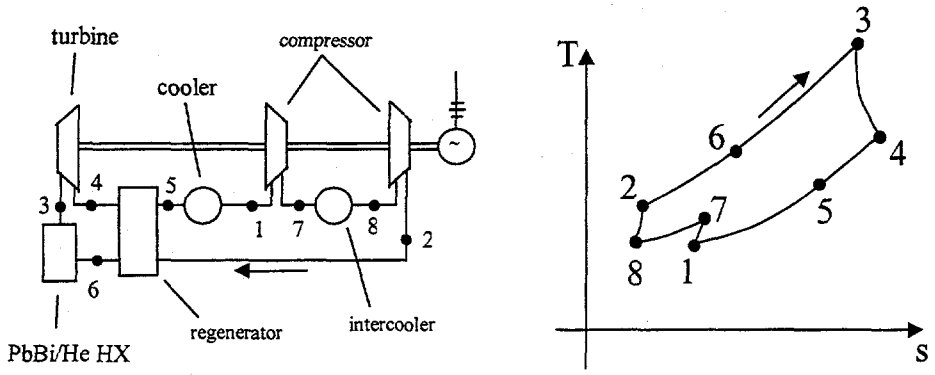

Figure 26. Brayton cycle temperature versus enthalpy diagram.

Figure 27 illustrates the thermal efficiency as a function of the compression ratio for the configuration with one and two inter-coolers and for a 500°C cycle maximum temperature. The net work per unit mass ranges from 300 to 450 kJ/kg. Values of T_6 (i.e. the heat exchanger inlet temperature) are within the required 125 to 250°C range. It should be noted that the addition of a second inter-cooler significantly improves the thermal efficiency, which, however, remains below 30%.

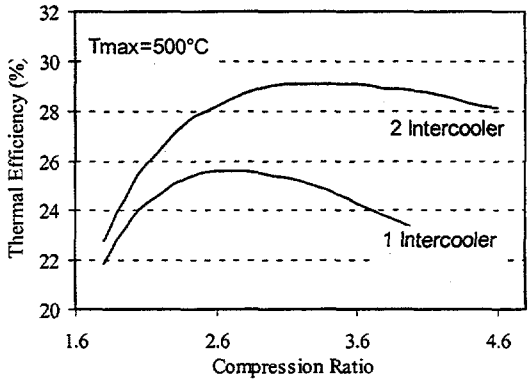

Figure 27. Thermal efficiency as a function of compression ratio.

Values of the thermal efficiency comparable to current LWR plants (~33%) may be achieved at the expense of cycle complexity and cost by adding a third inter-cooler and/or by re-heating the turbine outlet stream helium and/or by increasing the regenerator size and hence its effectivness.

A reference design of the PbBi/He HX was established with the following characteristics:

- Exchanged power: 1,800 MWth
- Counter-flow shell-and-tubes: PbBi in the shell/He in the tubes
- 180,000 circular smooth tubes made of EP-823 stainless steel; OD 8 mm; wall thickness 1 mm; length 6 m
- Triangular lattice (p/d=1.45)
- PbBi inlet/outlet temperature: 250°C/550°C
- He inlet/outlet temperature: 200°C/500°C

The heat exchanger occupies the entire annular region between the core barrel and the reactor vessel (see **Figure 13**)

3.7.3 Steam Cycle

The steam turbine option has not been thoroughly investigated so far; thus we limit this section to some preliminary considerations.

The characteristics that make water an attractive secondary coolant are:

a) its low cost,
b) the long and successful experience with operating steam cycle power plants,
c) the possibility of achieving relatively high thermal efficiency (>30%) at low maximum temperature (<400°C),
d) the possibility of designing a compact PbBi/water HX due to the high heat transfer coefficient of boiling water.

On the other hand:

a) water is corrosive and requires careful chemistry control,
b) Rankine cycles are generally rather complicated and costly.

If an indirect scheme is chosen (see section 3.5.1), it is possible to employ a supercritical Rankine cycle (i.e. a cycle operating above the critical pressure of water) and to obtain thermal efficiencies as high as 42% [Adamov et al. 1992].

If a direct scheme is chosen (see Section 3.5.6), the limit on the operating pressure is set by the design of the reactor vessel. The BWR experience indicates that:

a) a large vessel at approximately 7.0 MPa can be designed and safely operated,
b) thermal efficiencies above 30% can be achieved if saturated steam at 7.0 MPa is produced.

3.8 Development of a RELAP5 Version for Lead-Bismuth Cooled Reactors (INEEL)

The RELAP5 computer code (LMITCO 1995) was developed to calculate the thermal-hydraulic response of nuclear reactors and related experimental systems during various transients, such as loss of forced flow, loss of heat sink, and loss of coolant. The code models the hydraulic response of the working fluid, including both single- and two-phase effects, and the thermal response of heat structures such as fuel pins and steam generator tubes. Core power is calculated with a reactor kinetics model that accounts for thermal-hydraulic feedback and decay heat. The code is extremely flexible, allowing it to simulate a wide range of thermal-hydraulic systems. Although RELAP5 was primarily developed for analysis of commercial, light water reactors, it has been applied and assessed extensively for a variety of reactor designs.

The ATHENA computer program (Carlson et al. 1986) is based on RELAP5. ATHENA shares a common source code with RELAP5, but includes extensions to allow the use of other working fluids, including liquid metals, but not the lead-bismuth eutectic. Thus, the fluid properties of the lead-bismuth eutectic were added to ATHENA so that it could be used as a tool in the analysis of lead-bismuth reactor designs. The development of lead-bismuth fluid properties is described in Section 3.8.1.

Since the ATHENA code is based on RELAP5, which was primarily developed for analysis of light water reactors, it is appropriate to question the applicability of the code for analysis of lead-bismuth reactors. An evaluation of the applicability of the code for analysis of lead-bismuth reactors is presented in Section 3.8.2.

3.8.1 Lead-Bismuth Fluid Properties

The equation of state for the lead-bismuth eutectic was developed (Shieh 1999) as follows. First, the specific heats for both lead and bismuth up to the melting point of the alloy were obtained from standard tables. The specific heat of the alloy was computed as a weighted average of the lead and bismuth values according to their mass fractions (Young 1999). A simple integration technique was then used to find the enthalpy of the alloy in the solid state up to the melting point. This was then added to the latent heat of fusion data (Young 1999) to obtain the enthalpy of the liquid alloy at the melting point. The density and temperature at the melting point were also obtained from the tables.

A soft sphere model (Young 1977) was then used to find the equation of state of the alloy. The soft sphere model is based on the generalized Van der Waal's equation that has five adjustable parameters. Two of these parameters plus two constants used in the model are the same for both lead and bismuth and therefore the same for the alloy. The procedure to find the remaining three parameters is to solve the two simultaneous nonlinear equations, i.e.

Pressure at the melting point = 0, and

Energy at the melting point = enthalpy at melt.

These two equations were successfully solved for the three parameters using a standard numerical technique. Details of this procedure are given in the Software Design and Implementation Document (Shieh 1999).

ATHENA uses a simplified Clausis-Clapeyron formulation for the saturation pressure, P_{sat}:

$$P_{sat} = ce^{\gamma T}$$

where T is the temperature and c and γ are constants that were obtained from a least-squares curve fit of data (Nesmeyanov 1963).

ATHENA also requires equations for the transport properties of dynamic viscosity, thermal conductivity, and surface tension. These equations are described below. The dynamic viscosity, μ, is calculated as:

$$\mu = \psi \rho$$

where ψ is the kinematic viscosity and ρ is the density. The kinematic viscosity and density have been correlated (Touloukian et al. 1970) as:

$$\psi \ (m^2/s) = 61.423 \ (T - 273.15)^{-0.61106} * 10^{-7}$$
$$\rho \ (kg/m^3) = 10728.0 - 1.2159 \ (T - 273.15)$$

where T is the temperature in degrees K. The thermal conductivity, k, is calculated as a piecewise linear function (Touloukian et al. 1970):

$$k \ (W/m\text{-}K) = 9.408 - 0.00318 \ (T - 437.321) \text{ if } T < 437.321 \text{ K}$$
$$= 9.78 + 0.00973 \ (T - 475.554) \text{ if } 437.321 \text{ K} < T < 475.554 \text{ K}$$
$$= 9.78 + 0.0131 \ (T - 475.554) \text{ if } T > 475.554 \text{ K}$$

The surface tension, σ, is calculated based on data (Lyon 1952) as:

$$\sigma \ (N/m) = -5.5 \times 10^{-5} \ (T - 1073.15) + 0.367$$

3.8.2 Code Applicability Evaluation

Because of its high boiling point, the lead-bismuth eutectic will be a single-phase liquid during normal reactor operation and most transients. This greatly simplifies the applicability evaluation since ATHENA's two-phase models, which constitute much of the coding, are relatively unimportant. The most important phenomena that the code needs to be able to model for lead-bismuth are heat transfer and wall friction.

ATHENA contains many heat transfer correlations to simulate several heat transfer regimes, including forced and natural convection, nucleate boiling, critical heat flux, transition boiling, film boiling, and condensation. However, forced convection is the only regime expected to occur with lead-bismuth. For lead-bismuth and other liquid metals, the code calculates the forced convection Nusselt number, Nu, as:

$$Nu = 5 + 0.025\ Pe^{0.8}$$

where Pe is the Peclet number. This correlation is applicable for fully developed flow of a liquid metal in a tube with constant wall temperature (Bird et al. 1960). A separate correlation for natural convection is not required because the Nusselt number approaches five, rather than zero, as the Reynolds number goes to zero.

ATHENA allows the user to apply correlations developed for rod bundles rather than tubes for specific heat structures. However, this option was not available for liquid metals in the original code. Thus, a correlation developed by Westinghouse for rod bundles (Todreas and Kazimi 1990) was added to the code. The correlation is:

$$Nu = 4.0 + 0.33(P/D)^{3.8}(Pe/100)^{0.86} + 0.16(P/D)^{5}$$

where P/D is the pitch-to-diameter ratio of the rods. The correlation was developed for a range of $1.1 < P/D < 1.4$ and $10 < Pe < 5000$.

ATHENA calculates single-phase wall friction using the Darcy-Weisbach friction factor for laminar and turbulent flow. Turbulent flow is of most interest for lead-bismuth reactors. For turbulent flow, the code uses the Zigrang-Sylvester approximation (Zigrang and Sylvester 1985) to the Colebrook-White correlation (Colebrook 1939) to compute the friction factor. The Zigrang-Sylvester approximation is accurate to within 0.5% of the Colebrook-White correlation (LMITCO 1995). The SSC code (Guppy et al. 1983), which was developed for analysis of liquid metal fast breeder reactors, used an explicit approximation to the Colebrook-White correlation that was accurate to within 5%. The Colebrook-White correlation also is in good agreement with a correlation used in previous analyses (Greenspan et al. 1998) of lead cooled reactors. Thus, it is judged that the code's model is applicable for the calculation of single-phase wall friction in lead-bismuth reactors.

Phenomena that are not important in water, and thus were justifiably neglected in RELAP5, could be more important in liquid metals. Two such phenomena, axial heat conduction in the fluid and thermal entry length, were evaluated to determine their potential importance in a lead-bismuth reactor.

The thermal-conductivity in lead-bismuth is more than an order of magnitude greater than that of water. Thus, axial conduction within the fluid has the potential to be more important in lead-bismuth than in water. Axial conduction effects can generally be neglected if the Peclet number is greater than 100 (Kays and Crawford 1980). At normal operation, the core and steam generator tube design described in Section 3.9 have Peclet

numbers greater than 3000. The effects of axial conduction could begin to be become significant only at very low flow rates, such as might occur during natural circulation. The Peclet number will be checked after transients involving natural circulation are simulated. Until then, it seems appropriate to neglect axial conduction.

The forced convection heat transfer correlations used by the code for lead-bismuth are based on fully developed flow. The thermal entry length depends on the Prandtl number, and is longer for fluids with low Prandtl number fluids, such as liquid metals, than for water. Based on results for a circular tube with constant heat flux (Kays and Crawford 1980), the average heat transfer coefficient over the length of the core described in Section 3.9.1 during full-power operation is about 25% greater than the fully developed heat transfer coefficient. The average heat transfer coefficient for the steam generator tubes described in Section 3.9.2 is less than 10% greater than the fully developed value. Since the code does not model thermal entry effects, it is expected to provide somewhat conservative calculations of fuel temperature and heat exchanger efficiency.

ATHENA has a fundamental limitation for cases in which water enters the reactor system, such as would occur in the direct steam cycle described in Section 3.4.5 or in the case of a steam generator tube rupture. The limitation arises because the code solves continuity, energy, and momentum equations for two phases of a single fluid, lead-bismuth in this case. If water enters the system, four phases will be present: liquid lead-bismuth, a tiny amount of lead-bismuth vapor, liquid water, and steam. The code cannot handle this situation mechanistically as it represents only two fluid fields, one for the liquid and one for the gas. A major revision to the code would be required to model all four phases and the interactions between them. However, since the liquid water will boil to steam, acceptable results can probably be obtained by representing the steam as an ideal, non-condensable gas. A control system would be required to inject an appropriate amount of non-condensable gas and to remove from the lead-bismuth coolant the amount of energy needed to boil the water to steam.

The addition of water into the primary coolant will result in a two-phase flow. The predicted void fraction in the two-phase region will be governed by the code's bubble size and interphase drag correlations. These correlations are important because they will affect the natural circulation flow rate and determine whether or not steam flows into the core, where any voiding may significantly affect the reactivity. The applicability of these correlations for a lead-bismuth reactor has not yet been evaluated but has been deferred until the need arises.

In summary, the evaluation indicates that the ATHENA code is generally applicable for analysis of lead-bismuth reactors. The important code models of single-phase forced convection heat transfer and wall friction were judged to be applicable. The code will predict average heat transfer coefficients that are up to 25% lower than would be expected in the reactor because it assumes fully developed flow. The neglected effects of axial conduction were judged to be not important. The code cannot mechanistically represent the effects of adding water into the reactor coolant, but it is believed that the most important of these effects can be modeled adequately for design calculations. The

applicability of the code's interphase drag correlations for lead-bismuth/steam mixtures will be determined as the need arises.

3.9 Lead-Bismuth Cooled Reactor Thermal-Hydraulic Design (INEEL)

The INEEL design of the lead-bismuth-cooled reactor is based on the design illustrated in **Figure 13**. The primary differences between this design and the one described in Section 3.5 are that the heat exchangers are steam generators, with water on the shell side, and reactor coolant pumps are located in the downcomer below the steam generators to provide forced circulation. Details of the core and steam generator designs are provided in Sections 3.9.1 and 3.9.2, respectively. Note that the thermal-hydraulic design has not been optimized and thus is considered preliminary.

3.9.1 Core Design

The basic core design is consistent with the one analyzed in Section 3.2 and is summarized in **Table 12**. The core lattice is relatively open with a pitch-to-diameter ratio of 1.6. The large pitch-to-diameter ratio was chosen to reduce the frictional pressure drop across the core and thus enhance natural circulation, while still providing acceptable neutronic characteristics. The lattice is square, which results in a subchannel that is bounded by four adjacent fuel rods. The flow area fraction within each subchannel is 0.69, with the remainder of the area occupied by the fuel rods. When this fraction is applied to the entire area within the core barrel, the flow area is approximately 6.7 m^2, which is the value currently assumed in the analysis.

Russian experience (MacDonald 1999) with lead-bismuth indicates that the oxygen content of the fluid must be tightly controlled to prevent slag formation and that hydrogen will have to be added to the system periodically to breakdown the slag. The fluid velocity must be at least 1 m/s to properly distribute the hydrogen. However, erosion problems can occur if the fluid velocity exceeds 3 m/s. The design fluid velocity in the core was chosen to be at an intermediate value, 2 m/s, to avoid these problems.

Table 12. Core design.

Parameter	Value
Fuel pellet outer diameter	8.64 mm
Fuel composition (wt%)	20% of Pu+actinides, 80% of Zr
Gap thickness (at BOL)	0.2 mm
Bond material - lead-bismuth-tin alloy (wt%)	33%Pb-33%Sn-33%Bi
Cladding thickness	0.63 mm
Composition of cladding material	SS 304
Fuel pin outer diameter	10.3 mm
Pitch	16.48 mm
Pitch to diameter	1.6
Heated core length	1.2 m
Gas plenum height	0.9 m
Core barrel inner diameter	3.5 m
Pool vessel inner diameter	5.0 m
Core power	1000 MW_t
Fluid velocity	2.0 m/s

3.9.2 Steam Generator Design

Four basic criteria were used in the evaluation of steam generator designs. First, the designs evaluated were similar to those of commercial pressurized water reactors (PWRs) to reduce development costs. Second, the lead-bismuth fluid velocity was selected to be 2 m/s for the reasons described in the previous section. Third, the lead-bismuth was assumed on the tube side and the water on the shell side. This arrangement was selected to minimize potential problems with materials. However, it is recognized that reversing the arrangement and placing the lead-bismuth on the shell side would reduce the frictional pressure drop through the tubes by about 40%, thus enhancing natural circulation. Finally, the length of the steam generator tubes was selected so that the exit temperature of the lead-bismuth would be within about 5 C of the saturation temperature on the secondary side. In commercial PWRs, the cold leg temperature is generally about 10 C higher than the saturation temperature on the secondary side.

Three separate steam generator designs have been considered. The first design was a vertical once-through steam generator (once through steam generator) similar to a Babcock and Wilcox PWR. This steam generator contains an annular downcomer region that is separated from the boiler region, which contains the tubes, by a thin shell. Feedwater enters the downcomer, flows downwards and enters the bottom of the boiler, which contains the tubes. The flow is upwards through the boiler which is counter to the downwards flow of lead-bismuth inside the tubes. As the water flows up the boiler, it is heated until reaching dry superheated steam, which then flows into the steam lines.

The second design is a horizontal steam generator similar to those used in Russian VVERs except that the tubes are once through rather than bent. In this design, the tubes are in the lower half of the steam generator vessel and the mixture level is maintained

above the top row of tubes. Due to the large flow area and relatively small velocities, the steam rises to the top of the horizontal vessel and the liquid naturally separates due to the effects of gravity. Dryers can be added to the top of the steam generator if necessary.

A vertical U-tube design was considered briefly, but this design was dropped because the tube length required to cool the lead-bismuth to within 5 C of the secondary saturation temperature was less than the length of an average 180° U-bend.

The first two steam generator designs are summarized in **Table 13**. The values in the table are for a single steam generator. Four steam generators are required to remove the core power. The steam generator tube lengths are much shorter than in commercial PWRs, but the vessel diameters are comparable. The vessel diameters are too large to allow the steam generators to fit into the annular gap between the core barrel and the pool vessel (see **Table 12**). Consequently, the steam generators can not be immersed in the pool unless the pool vessel inner diameter is increased significantly.

Table 13. Steam generator designs.

Parameter	Vertical once through steam generator	Horizontal once through steam generator
Lattice	Triangular	Triangular
Number of tubes	13,255	13,288
Tube outer diameter	0.0147 m	0.0147 m
Tube thickness	0.001 m	0.001 m
Tube length	3.8 m	2.8 m
Pitch to diameter	1.5	1.5
Secondary hydraulic diameter	0.022 m	0.022 m
Vessel inner diameter	2.85	3.8

The following section compares the two steam generators during full-power operation.

3.10 System Modeling Studies (INEEL)

An ATHENA model of the lead-bismuth cooled reactor system has been developed. The system model uses the horizontal once through steam generator described in Section 3.9.2. A stand-alone model of the vertical once through steam generator described in Section 3.9.2. has also been developed. Both input models are described in Section 3.10.1. Steady-state results obtained with both models are described in Section 3.10.2.

3.10.1 Input Models

The ATHENA model of the lead-bismuth cooled reactor system is shown in **Figure 28**. The model represents the lead-bismuth pool as two regions that are separated by the core barrel. The region inside the core barrel, which contains the core and upper plenum, is represented with Components 510 and 520. The upper pool region, which is located

above the core barrel, is modeled with Components 530, 535, and 540. The liquid level normally resides in Component 535. Component 540 contains a cover gas at near atmospheric pressure that sets the pressure of the system. The region outside the core barrel contains the steam generators, reactor coolant pumps, and down-comer. In the design, the down-comer is separated into four quadrants, each connected to one steam generator and reactor coolant pump. For simplicity, the four steam generators, pumps,

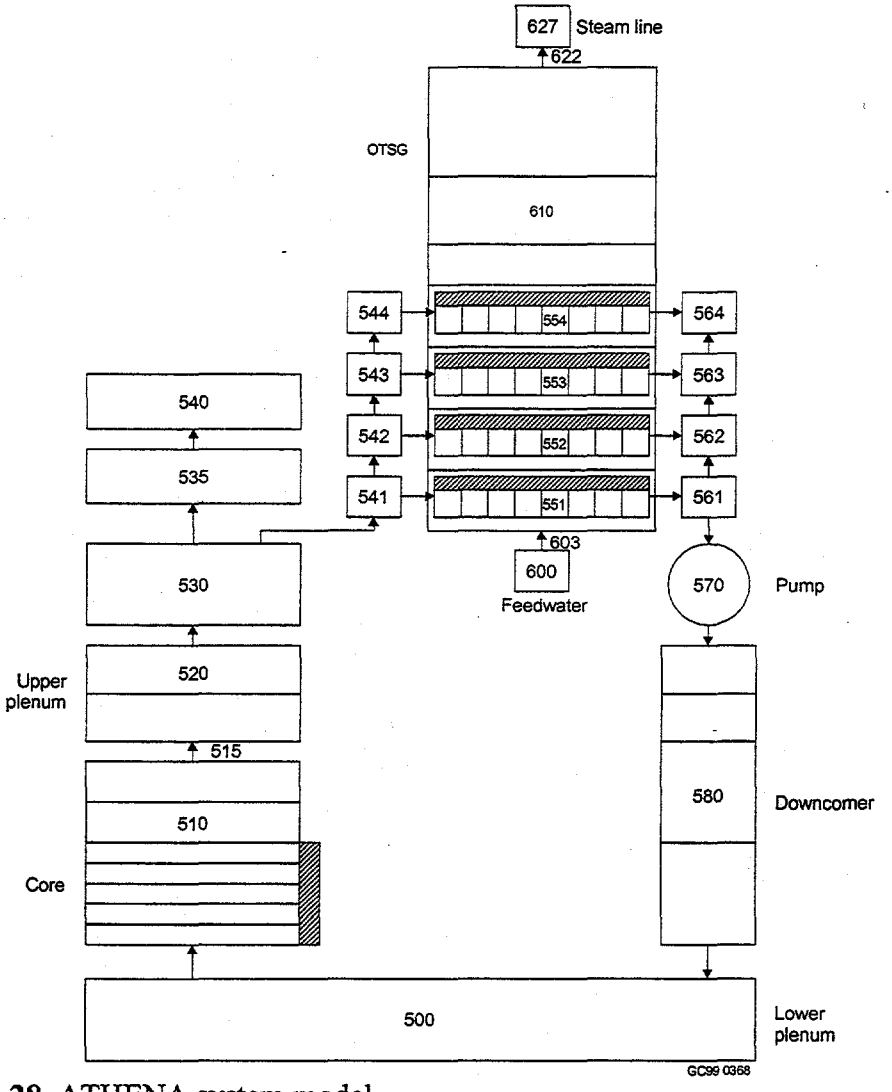

Figure 28. ATHENA system model.

and down-comer quadrants are combined into a single flow path in the model. As the design matures, it is relatively simple to expand the model so that it represents each quadrant separately. Component 500 represents the lower plenum region, which connects the down-comer and core.

The horizontal once through steam generator is represented as an inlet plenum (Components 541 through 544), steam generator tubes (Components 551 through 554), and outlet plenum (Components 561 through 564). The steam generator is divided into four layers of tubes so that the effects of a decreasing secondary liquid level in loss-of-heat sink events can be simulated. The lowest layer contains approximately 10% of the tubes while each of the other layers contains about 30%.

The secondary side of the steam generator is modeled in two halves. The lower half contains the tubes and is divided into four levels consistent with the primary side nodalization. The upper half is divided into three levels to allow separation of steam. Boundary conditions are used to set the feedwater flow rate and steam line pressure. The steam line pressure was set at 6.89 MPa, which is representative of the secondary pressure in commercial PWRs.

The bottom of the steam generator tubes is currently about 4 m above the top of the active core. However, this distance may be changed to improve the performance of the system during transients.

The ATHENA system model contains 67 volumes and 69 junctions.

The stand-alone model of the vertical once through steam generator is shown in **Figure 29**. The primary side of the tubes is represented with Component 550. Normally flow is down through the tubes. The secondary side of the steam generator is modeled with three components representing the downcomer (Component 605), boiler (Component 610), and upper annulus (Component 615) regions. Boundary conditions are used to set the hot leg temperature, cold leg pressure, and flow rate on the primary side and the feedwater flow rate and steam line pressure on the secondary side.

3.10.2 Steady-State Results

The ATHENA system model with the horizontal once through steam generator was used to perform a full-power, steady state calculation. The results of the steady-state calculation are summarized in **Table 14**.

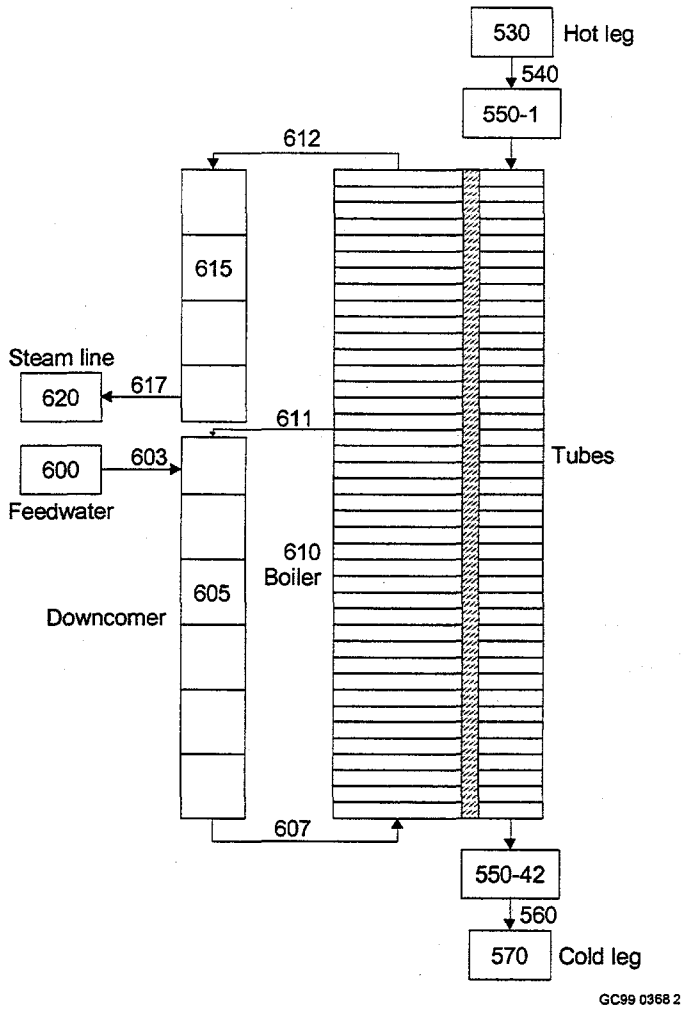

Figure 29. ATHENA model of the vertical once through steam generator.

Table 14. Steady-state results with the ATHENA system model.

Parameter	Value
Core power	1000 MW_t
Lead-bismuth flow rate	1.38×10^5 kg/s
Hot leg temperature	340.9 C
Cold leg temperature	292.8 C
Secondary fluid temperature	284.8 C
Core fluid velocity	2.0 m/s
Once through steam generator tube velocity	2.0 m/s
Pump differential pressure	140 kPa
Lead-bismuth fluid mass	2.1×10^6 kg

The core fluid velocity was chosen to be 2.0 m/s to avoid materials problems as described in Section 3.9.1. Choosing the fluid velocity essentially determines the system mass flow rate since the core flow area is known from the geometry and the density of the lead-bismuth is only a weak function of temperature. The additional specification of the core power then determines the fluid temperature rise across the core. The cold leg temperature is governed by the secondary temperature, which is set by the assumed steam line pressure, and the length of the steam generator tubes. The resulting cold leg temperature of the current design is similar to those of commercial PWRs. Coincidentally, the temperature rise across the core is also quite similar to commercial PWRs.

The pressure rise generated by the pump for this design is about 25% of that in commercial PWRs. Assuming the same pump efficiency, the ratio of pump power to core power in this design is only about half that of existing PWRs. Thus, the pumping power requirements for this design should be acceptable. About 60% of the pressure rise across the pump is dissipated in the once through steam generator tubes and most of the remainder is dissipated in the core.

The temperature difference between the upper plenum and downcomer generates a gravitational head that is about 3% of that produced by the pump. The temperature rise across the core during natural circulation should be less than that shown in **Table 12** once the core power drops to 3%. Thus, the design is expected to have adequate natural circulation characteristics. The natural circulation flow rate in this design is much less than the 100% value for the design described in Section 3.5.1. The primary reasons for the different natural circulation flow rates are that the temperature rise is much smaller in this design (48 C versus 300 C), the fluid velocity is between 2 and 4 times higher, and the lead-bismuth is assumed to be inside rather than outside of the tubes.

A steady-state calculation using the vertical once through steam generator model was also performed. **Table 15** presents a comparison of the results with the horizontal and vertical once through steam generator models. The values shown represent the total of four steam generators.

Table 15. Steam generator results.

Parameter	**Horizontal once through steam generator**	**Vertical once through steam generator**
Power, MW_t	1002	994
Secondary pressure, MPa	6.89	6.89
Feedwater flow rate, kg/s	611	552
Feedwater temperature, K	533	533
Steam temperature, K	558	597
Collapsed liquid level, m	1.10	1.57
Total fluid mass, kg	18,600	20,800

The power removed by the horizontal once through steam generator is the total of the core power and the pump heat. The power removed by the vertical once through steam

generator is about 1% too low and can be increased by adding more feedwater. One of the most significant differences between steam generators is that the vertical once through steam generator produces steam that is 39 C superheated whereas the horizontal once through steam generator produces saturated steam. Consequently, the vertical once through steam generator will be more efficient. However, the vertical once through steam generator also requires a longer tube because of the poorer heat transfer characteristics in the post-dryout regimes. The longer tube causes an increase in the frictional pressure drop and poorer natural circulation characteristics.

One of the advantages of both steam generators compared to the lead-bismuth-helium heat exchangers described in Section 3.7.2 is that the initial inventory of water can be used to remove core decay heat and lessen the demands on the DHRS. For example, the amount of liquid initially present in the horizontal once through steam generator can remove at least 7 minutes of decay power before drying out. It would be relatively easy to increase the initial liquid level in the horizontal once through steam generator and thus provide more decay heat removal capability. For example, the steam generators in some PWRs are able to remove decay heat for more than an hour after a total loss of feedwater. Similar results should be possible with this design.

4. MAJOR TECHNICAL ACCOMPLISHMENTS

4.1 Neutronics

A reference core geometry and fuel composition were established that:

- yield negative reactivity feedback upon total or partial coolant voiding, and
- provide enough excess reactivity for the non-fertile fuel at BOL to operate on an 18 month refueling cycle.

B_4C was chosen as the control material.

Preliminary data on reactivity and fuel life for fertile fuel was obtained, with the fuel choice being a Pu-Th-Zr (+depleted U).

4.2 Materials

The fuel thermal conductivity variation with temperature and burnup was calculated based on models developed by ANL for the IFR metallic fuel.

4.3 Plant Engineering

A design envelope was established for the following items:

- full power natural circulation in the primary system;
- linear power limits in the fuel;
- vessel temperature limits;
- decay heat removal system; and
- gas turbine power cycle.

The following tasks were performed to develop a system analysis capability:

- lead-bismuth fluid properties were added to a version of the ATHENA code;
- the ATHENA code was shown to be generally applicable for the analysis of lead-bismuth cooled reactors;
- ATHENA models of preliminary designs were created; and
- the ATHENA models were used to perform steady-state calculations at full power.

5. MAJOR REMAINING TECHNICAL CHALLENGES AND FUTURE WORK

5.1 Neutronics

(a) Verify the core reactivity stability throughout the burnup cycle.
(b) Achieve a suitable doppler feedback.
(c) Reduce the axial power peaking due to control rod insertion.
(d) Investigate the implications of the two different approaches to actinide burning (i.e. critical reactor vs. ATW) in terms of neutronic safety and fuel cycle.
(e) Achieve a core lifetime of ~15 years for fertile fuel.
(f) Keep the U-233 <12wt% of the total uranium content for fertile fuel, while maintaining core life and non-proliferation concerns.
(g) Keep the power density less than that of Na cooled reactors (~80%).
(h) Do an economic comparison of non-fertile and fertile fuel cycles.

5.1.1 Planned Papers and Conferences for Neutronics Work

Listed are currently proposed papers/conferences and submittal dates for the work that has been done.

- "Computational Results of Metallic and Nitride Fuels for Advanced Lead-Bismuth Cooled Fast Reactors", *Eighth International Conference On Nuclear Engineering* – ICONE-8. Abstract to be submitted September 15, 1999.
- "Comparative Study of Reactivity Trends of Lead/Lead-Bismuth Cooled Small Long-Life Nuclear Reactors". Manuscript to be submitted to *Nuclear Technology*, September 30, 1999.
- "Design Criteria and Computational Results for Optimal Fuel Composition of a Long Life Lead-Bismuth Cooled Nuclear Reactor". Manuscript to be submitted to *Nuclear Engineering Design*, October 30, 1999.

Other papers/conferences will be submitted as work progresses. These will include, but are not limited to, optimal core designs, safety and non-proliferation issues, and economic estimation/analysis/comparison of lead-bismuth cooled reactors.

5.2 Materials

(a) Advance the knowledge and understanding of Pb-Bi interaction with reactor materials with the goal of accurately establishing the temperature limits.

(b) Select a set of suitable materials for the fuel cladding, the gap thermal bond, the core internals and the reactor vessel.

Los Alamos National Laboratory (LANL) has already acquired some information on the Russian experience with Pb-Bi: an effort will be made to bring this knowledge to the INEEL and MIT.

The possibility of setting up an experimental Pb-Bi loop at MIT and/or the INEEL will be considered with the following goals:

- validate the Russian approach to the corrosion issue,
- explore alternative approaches,
- establish a US Pb-Bi database.

5.3 Plant Engineering

(a) Establish in quantitative terms the trade-off between natural circulation capability, reactor size, reactor power and costs.
(b) Explore innovative primary system configurations.
(c) Assess the economic and technical impact of the polonium issue.
(d) Select a suitable power conversion cycle.

No matter what primary system configuration is selected, the problem of managing the coolant activation to form Po will be a key question. Information on the chemical characteristics of Polonium and its compounds is rare and incomplete. MIT will evaluate the possibility of undertaking an experimental program with the goal of assessing:

- the basic thermodynamic properties of Po and its compounds;
- a suitable way to separate Po from Pb-Bi; and
- rate of release of Po and its compounds in air and water.

The technical challenges related to the development of system analysis capability and future work are:

- develop ATHENA input for reactor kinetics to simulate feedback between the thermal-hydraulics and neutronics;
- develop models of the DHRS; and
- simulate various transients, such as loss of flow and loss of heat sink, to investigate the safety characteristics of the design.

5.3.1 Planned Papers and Conferences for Plant Engineering Work (INEEL)

Two papers are tentatively planned for submittal to a journal or conference. The topics for these papers include the addition of lead-bismuth fluid properties into the ATHENA code and the simulation of various transients to investigate the safety characteristics of the design.

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FULL PROJECT TITLE	**Regulatory Excellence: Performance-Based Regulation for DOE Facilities**
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Regulatory Excellence: Performance-Based Regulation for DOE Facilities

1. Scientific or Technical Purpose

Performance-based regulation (PBR) focuses on results as the primary objective of regulatory oversight. PBR can potentially reduce operating costs by avoiding unnecessarily burdensome requirements and preventing needless interruptions in production and/or processing. Building upon existing PBR activities in the commercial power industry, this three-year research program was initiated to develop a framework for formulating, selecting, and demonstrating performance-based goals for licensing and regulating U.S. Department of Energy (DOE) nuclear facilities. Using innovative approaches, we are developing a methodology, supported by software tools, that can be used to define goals and a systematic process for demonstrating compliance with these goals. Our ultimate objective is to demonstrate the feasibility of this regulatory approach using three DOE facilities as case studies. Many of the methods developed from our research will be applicable to the commercial power industry and industries manufacturing/handling hazardous materials.

2. Progress to Date

Major Issues and Tasks to Address Issues

DOE owns and operates approximately 34 individual sites across 13 states. Facilities located at these sites include nuclear research and production reactors, nuclear weapons assembly and disassembly facilities, chemical processing facilities, nuclear material storage vaults, reactor fuel fabrication facilities, tritium recovery facilities, particle accelerators, and research laboratories. DOE typically regulates all radiological, chemical, and physical hazards at its nuclear facilities. Most DOE regulations were derived from existing regulations developed by the U.S. Nuclear Regulatory Commission (NRC) or other agencies.

In order to develop a PBR system and apply it to specific DOE facilities, several key issues must be addressed. These issues range from understanding key concepts, such as Defense in Depth (DID), Safety Margin (SM), and As Low as Reasonably Achievable (ALARA), embedded in existing regulation to developing an appropriate framework for defining performance goals and requirements. Prior to embarking on this research program, we identified the key issues listed in Table 1 that will be addressed.

Table 1. Key issues associated with PBR application.

Issue 1	What are the key concepts, such as "defense in depth" and "safety margins", embedded in current NRC regulations that must be considered in this framework?
Issue 2	Can a framework be developed that will encompass most types of DOE facilities?
Issue 3	What are appropriate DOE facility performance requirements?
Issue 4	To what extent may risk information be used to select performance goals and monitor the facility's ability to meet these performance goals?
Issue 5	What proof must be offered to demonstrate compliance with performance-based goals?
Issue 6	What happens if a facility does not meet the performance goals?
Issue 7	How should uncertainties be handled in defining the goals and demonstrating compliance?

Having identified these issues, we next defined a programmatic approach for addressing the specific problem at hand. In this approach, we defined the five tasks listed in Table 2. Note that some of the original tasks proposed in Reference 1 were modified to incorporate Reference 2 peer review comments. Program tasks, which will be completed within three years, will be performed in parallel because results from some tasks impact activities in other tasks. To facilitate communication and monitor our progress in this program, results from each task are periodically documented in informal letter reports. Two of these reports are included as attachments to this document. Student theses, technical conference presentations, and journal

articles will also discuss research results. Highlights from research conducted in this program are discussed in this report. Attachments A and B to this report summarize more detailed technical results from the first two tasks of this project.

Table 2. Regulatory Excellence Project Task Descriptions

Task	Focus/Description	Initial Task Products
1	Review current regulation to identify key concepts for inclusion in our proposed framework (MIT)[1]	-Preliminary Letter Report (May 1999) -Final Letter Report (June 1999)
2	Classify DOE facilities in order to select representative case studies (INEEL)	-Preliminary Letter Report (May 1999) -Final Letter Report (June 1999)
3	Utilize expert input to develop objectives of performance requirements (MIT)	-Draft letter report summarizing approach and workshop results (*August 1999*)[2]
4	Develop preliminary regulatory framework (MIT)	-Letter report describing proposed framework (*September 1999*)
5	Apply and refine regulatory framework (INEEL)	-Letter report documenting application of framework to first case study (*March 2000*)
6	Interact with recognized experts to review and gain acceptance for our proposed approach (MIT)	-Draft letter report summarizing results from first workshop (*September 1999*)

Technical Accomplishments

Although we are only in the first year of this three-year program, preliminary results were obtained from several of the tasks listed in Table 2. Detailed technical results may be found in letter reports documenting progress on completed tasks.[3,4] Highlights from various tasks are summarized below.

Task 1: Focussed Review of DOE and NRC Regulatory Guidance and Activities (MIT)[1]

During the first year of this project, INEEL and MIT conducted a focused review of DOE and NRC regulatory guidance and activities to gain perspective about the intent of regulations currently applied by each organization. Led by MIT, this joint MIT/INEEL effort included a review of recent NRC interactions with the Nuclear Energy Institute (NEI) to implement PBR and risk-informed PBR (RIPBR) in commercial power plants and a review of current DOE/NRC pilot project interactions.

Key results from this task, which are documented in Reference 3 (Attachment A to this report), include identifying key concerns that must be addressed in implementing PBR, "cornerstones" that must be included in PBR, lessons learned from implementing PBR and RIPBR into commercial power plants, and insights from DOE/NRC pilot plant interactions.

- Current NRC Regulatory Philosophy and Critical Concerns. In our review, it is obvious that principles, such as ALARA, DID, and SM have been the basis for the treatment of uncertainties by the current regulatory system. This approach has ensured public health and safety, but it also has caused undue regulatory burdens. A major pitfall in the current framework is that qualitative evaluation of risk does not permit the most effective allocation of resources because prior to PRAs, uncertainties were not quantified. Traditional engineering analysis integrated with PRA, however, has revealed a potentially successful means of addressing and quantifying uncertainties.

[1] Although both organizations contributed to each project task, a lead organization was assigned to each task.
[2] Dates in italics are proposed completion dates.

In moving to a PBR or RIPBR structure, two conflicting concerns have been raised:

- Can DID be undermined by the introduction of PBR?
- Will the benefits of the risk-informed regulation be restricted by DID?

The critical question is how to take advantage of PBR and reduce unnecessary burden associated with DID and SM.

- NEI/NRC interactions.

An important topic currently under debate by the NRC is revising aspects of the regulatory oversight process. In spite of successful improvement in plant performance of the last 10 years, the NRC and the nuclear industry recognized that the current inspection, assessment and enforcement processes show some deficiencies. Redundant actions and outcomes, non safety-focused inspections, and subjectivity were cited as major obstacles to process efficiency.

The issuance of SECY-98-045,[5] marked a decisive step in NRC implementation of a new integrated assessment process. In September 1998, the previously used Safety Assessment of Licensee Performance (SALP) process was officially suspended, and it is currently debated whether it will be indefinitely terminated.

In parallel with the NRC-staff's development of the Integrated Review of the NRC Assessment Process (IRAP) proposal, NEI developed an independent proposal for improving the assessment process.[6] The proposed NEI approach conceptually focused on: maintaining the barriers to radionuclide release, minimizing events that could challenge the barriers, and ensuring that systems can perform their intended functions. Performance would be measured through reliance on high-level, objective indicators with thresholds set for each indicator to form a utility response band, a regulator response band, and a band of unacceptable performance (performance tiers).

In response to the NEI proposal, a public 60 days comment period (ended October 6th 1998) was issued by the NRC and after a 4-day public workshop (September 28 – October 1, 1998), consensus was reached on the overall philosophy for regulatory oversight.

The objectives of the process were defined as follows:
- Increase objectivity
- Improve scrutability
- Reduce redundancy
- Risk-inform the process

To increase objectivity, the staff will rely on performance indicators (19 performance indicators have been currently defined). Performance indicators, in conjunction with risk-informed baseline inspections, are aimed at providing a wide range of data to assess licensee's performance in risk significant areas. They are not intended to provide a thorough coverage of all plant design and operational aspects.

- NRC/NEI Interaction on Special Nuclear Materials

Upon encouragement from NEI, the NRC staff revised 10 CFR Part 70, Domestic Licensing of Special Nuclear Materials (SNM), with the intention of providing a RIPBR approach.[7] The proposed amended rule was released last year, after suggestions from the NEI, and iterative discussions between the NRC staff and the NEI.[8] The major provisions of the revision include: (1) the licensing basis for the safety program would be a formal ISA (Integrated Safety Assessment); (2) adverse consequences limits would be established against which licensees must protect; (3) safety bases would be included in the license application (i.e., identification of potential accidents, items in prevention and mitigation systems for these accidents, and measures needed to ensure continuous availability and reliability of these items); (4) licensees would be able to make certain changes without NRC prior approval, based

on ISA results; and (5) the Commission may employ a qualitative backfitting mechanism (to enhance regulatory stability), after initial conduct and implementation of the ISA by the licensees.[9]

- DOE/NRC Pilot Plant Interactions.

  Several DOE facilities were evaluated for regulation by NRC. The DOE/NRC pilot plants included the Lawrence Berkeley National Laboratory (LBNL), the Radiochemical Engineering Development Center (REDC) at Oak Ridge National Laboratory (ORNL), and the Receiving Basin for Offsite Fuel (RBOF) at the Savannah River Site. No significant safety issues were observed at these pilot plants. Other pilot plant interactions were never completed.

  DOE/NRC pilot plant interactions provided several key insights about DOE facility regulation for our project. Results suggest our PBR framework should encompass a broad range of issues (waste management and treatment, emergency preparedness, environmental and personnel monitoring, radioactive materials control, etc.). In addition, key NRC regulations (10 CFR Parts 19, 20, 30, 70, 73, 74) and principles (ALARA, DID, and SM) referenced in pilot plant interactions must be considered in our PBR framework. Finally, DOE/NRC pilot plant interactions demonstrated a need to select more complex facilities.

  To better understand interactions occurring during the DOE/NRC pilot plant interactions, MIT held a workshop featuring Ms. Roxanne Summers, who served as an NRC representative on the DOE/NRC pilot plant task force. Ms. Summers discussed her perspective about DOE's current method for regulating its facilities, how external regulation of DOE facilities could be accomplished by NRC, and the current status of the pilot plant interactions. Representatives from INEEL, MIT, and JAC attended this meeting.

Task 2: DOE Facility Review and Case Study Selection (INEEL)

To ensure that the proposed framework is sufficiently broad-based and applicable to most DOE nuclear facilities, we reviewed various types of DOE facilities, developed key DOE facility groups, determined a prioritization of facility groups for evaluation as case studies, and selected representative facilities as case studies from the three higher-priority DOE facility groups.

To assist us in our review, we developed the NUclear FACility (NUFAC) searchable database into which information about various DOE facilities could be entered. Figure 1 contains a window from this newly-developed software tool for a representative reactor, the INEEL Advanced Test Reactor (ATR). As indicated in this screen, NUFAC allows comparisons between the facilities selected as case studies and other facilities in each group. NUFAC includes key facility information, such as design features, current and potential missions, co-located facilities, licensing status, regulatory issues affecting operation, point-of-contacts, and pertinent licensing documentation. A pilot version of this software was presented to Mr. Bill Magwood, Director, DOE Office of Nuclear Energy, Science, and Technology (DOE-NE). DOE-NE has agreed to our proposal to refine NUFAC and launch it as an internet application. DOE-NE plans to use this tool to identify facilities required to support DOE missions for the next 20 years.

Results from our review, which are summarized in Reference 4 (included as Attachment B), emphasize the uniqueness of DOE facilities. For example, currently operating DOE reactors differ in their fuel composition, moderator, and coolant. This uniqueness requires that our proposed framework be sufficiently general to encompass various facility groups and most designs within each group. To cover the diversity of DOE facilities, we developed the seven key DOE facility groups listed in Table 3. Although boundaries between each group may not always be clearly defined, each group has sufficiently different characteristics and licensing issues to warrant separate evaluation.

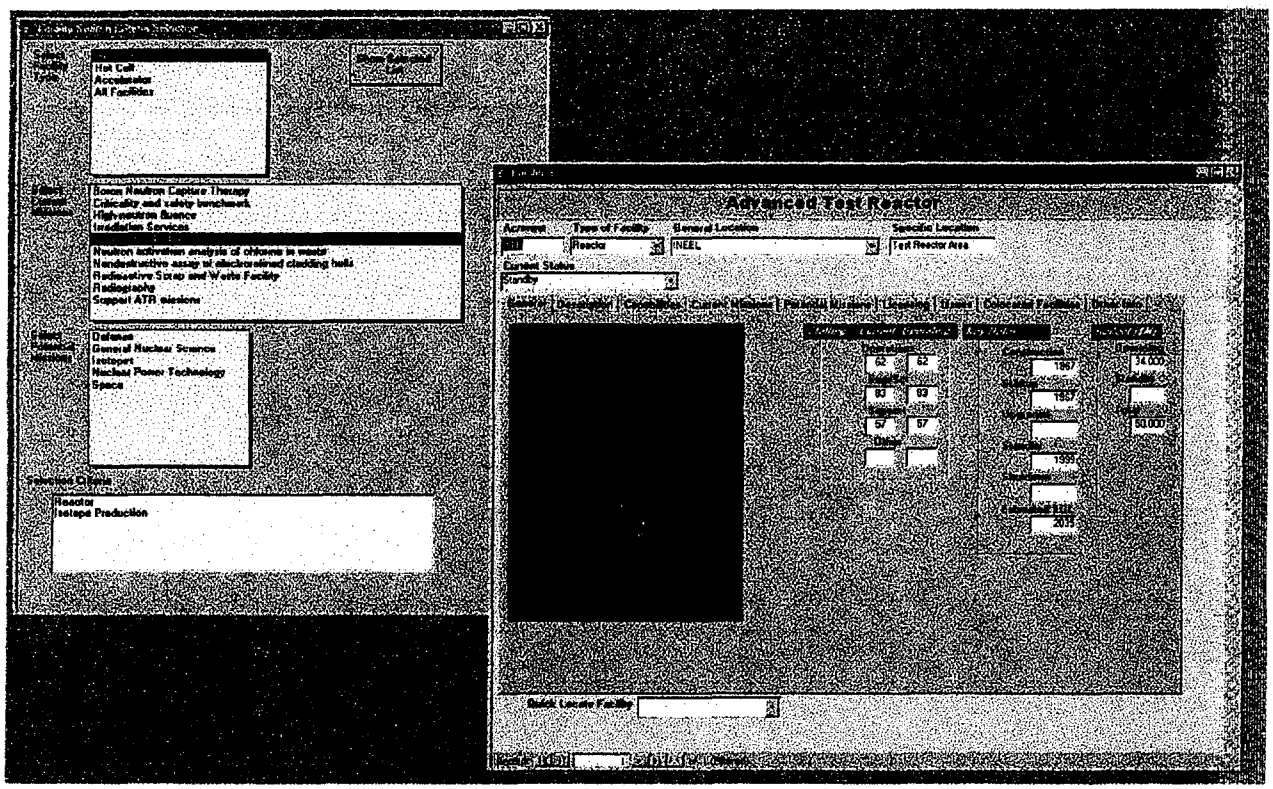

Figure 1. ATR screen in pilot version of NUFAC database.

Table 3. Proposed DOE facility groups.

Group	Description
Reactors	Wide range of unique reactors and critical facilities used for code validation and irradiation services (fuel testing, materials irradiation, isotope production, etc.).
Hot Cells	Structures housing one or more hot cells used for fuel characterization and testing, materials evaluation and testing, fuel development, waste characterization and testing, etc.
Waste Storage Sites	Facilities designed to house, monitor, and retrieve radioactive waste.
Nuclear Materials End-Use Facilities	Facilities utilizing low-level radioactive materials for a range of applications (e.g., radiopharmaceuticals, advanced medical imaging, nuclear magnetic resonance imaging, etc.).
Contaminated Sites	Buildings and grounds where radioactive materials were previously utilized and/or where radioactive materials are temporarily housed.
Manufacturing/Processing Facilities	Facilities that handle, examine, process, and perform research and development (R&D) on radioactive materials.
Accelerators	Systems capable of producing high energy particles (electrons, protons, neutrons, positrons, heavy ions, etc.) for isotope production, materials testing and irradiation, etc.

Facility groups are listed in Table 3 according to their priority for evaluation in our project. The reactor group was assigned the highest priority for consideration because of our desire to include examples that are fairly complex, whose results could be directly transferred to the commercial power industry. The hot cell group was also considered as a higher priority group because results from this example can be compared with

group was also considered as a higher priority group because results from this example can be compared with results from the recent DOE-NRC pilot plant evaluation of the ORNL REDC Hot Cells. The third group that will be considered is a Waste Storage Site. This particular group was considered at this time because of a recently-constructed INEEL facility that is subject to both NRC and DOE regulation. It is anticipated that evaluation of this facility will yield useful insights about contrasts and similarities in requirements for the two agencies.

Case studies for higher priority facility groups (reactors, hot cells, and waste storage sites) were selected based on location and organizational support. Because of financial constraints, we limited this study to facilities located in Idaho. Furthermore, the selection procedure required that the organization operating each facility would be willing to provide staff to interact with INEEL and MIT. Within these constraints, we selected the following facilities as case studies:

- Advanced Test Reactor (ATR) from Reactor Group. This 250 MWt light water moderated and cooled reactor is located at the Test Reactor Area of INEEL. It performs a range of irradiation services for government, industry, and foreign organizations.

- Hot Fuel Examination Facility (HFEF) from Hot Cell Group. This facility includes an air-filled decontamination cell and an argon-filled main cell. Located at the Argonne National Laboratory-West (ANL-W) site of INEEL, it primarily provides services to DOE fuels and waste programs.

- Three Mile Island Unit 2 (TMI-2) Independent Spent Fuel Storage Installation (ISFSI) from Waste Storage Facility Group. This recently-commissioned facility provides horizontal dry storage for the TMI-2 core debris. Located at the Idaho Nuclear Technology and Engineering Center (INTEC) operating area at INEEL, the facility is subject to DOE and NRC regulations through an agreement between the state of Idaho and DOE.

Information about the design, missions, and the current regulatory status for each of these facilities is summarized in Reference 4 (Attachment B). We are currently reviewing appropriate facility documents, such as Final Safety Analysis Reports (FSARs), Technical Safety Requirements (TSRs), DOE contracts with M&O contractors, facility-specific authorization agreements, PRAs, events reported in the DOE Occurrence Reporting and Processing System (ORPS) and the Nonconformance Tracking System (NTS), and applicable DOE Rules, Orders, and Regulations to identify areas for improvement in the current DOE regulatory system and to select performance goals, methods for demonstrating compliance, and appropriate noncompliance measures. We are also interviewing personnel associated with case study facilities.

Task 3: Expert Input to Develop PBR Objectives (MIT)

In developing a more efficient PBR system, we must ensure that performance requirements are derived from both a systems analysis approach (e.g., PRA or ISA), as well as recognizing requirements from a broad range of regulatory and facility experts (i.e., NRC, DOE-HQ, Regional DOE Offices, Laboratory Contractor Management, Facility Management, Facility Employees, Facility Customers, State Government, Federal Government, Regional Government, Occupational Safety and Health Administration [OSHA], Environmental Protection Agency [EPA], etc.). By incorporating experts from the beginning of the PBR development process, we seek to avoid many of the problems experienced in the past where a broad range of facility stakeholders weren't consulted. The expert elicitation process will help us formulate regulatory performance goals for the DOE that will encompass their concerns and priorities concerns.

Our first expert group elicitation will be a 1 1/2 day workshop at MIT on August 3 and 4, 1999. The purpose of the workshop is to gain insights from the experts about essential issues to consider in a PBR

approach, potential objectives of PBR, and potential performance measures to be used in PBR. More specifically, we hope to find out what attributes experts are looking for in a regulatory system, understand what is acceptable to them, and why. We plan to represent part of this information in one or more value trees. Figure 2 shows the basic structure of the value influence diagram, and some examples of impact categories and objectives that may come out of the workshop. Because this is work in progress and the workshop has not yet happened, the value trees we will use may look very different from this example.

Reaching consensus among the experts is not the goal of the workshop. Rather, we wish to gain insight into the possible performance requirements for any DOE regulatory system. Ideally, we would be able to identify a set of "safety cornerstones" (see discussion under Task 4) for PBR of DOE facilities based on the value diagram(s) elicited from the experts.

Confirmed workshop participants include: Gary Zeman, DOE Contractor, Lawrence Berkeley National Laboratory (LBNL), Participant in DOE/NRC pilot plant interaction for LBNL; Raymond Furstenau, DOE/ID Facility Manager for the INEEL ATR; Andy Marchese, Consultant and former Deputy Director, DOE Office of Operating Experience Analysis; Tom Kress, NRC Advisory Committee on Reactor Safety (ACRS) Member and Former Oak Ridge National Laboratory (ORNL) scientist; Kim Thompson, Assistant Professor at the Harvard School of Public Health; and Gary Marshall, DOE Contractor, Argonne National Laboratory –West (ANL-W), HFEF ES&H Program Control Manager.

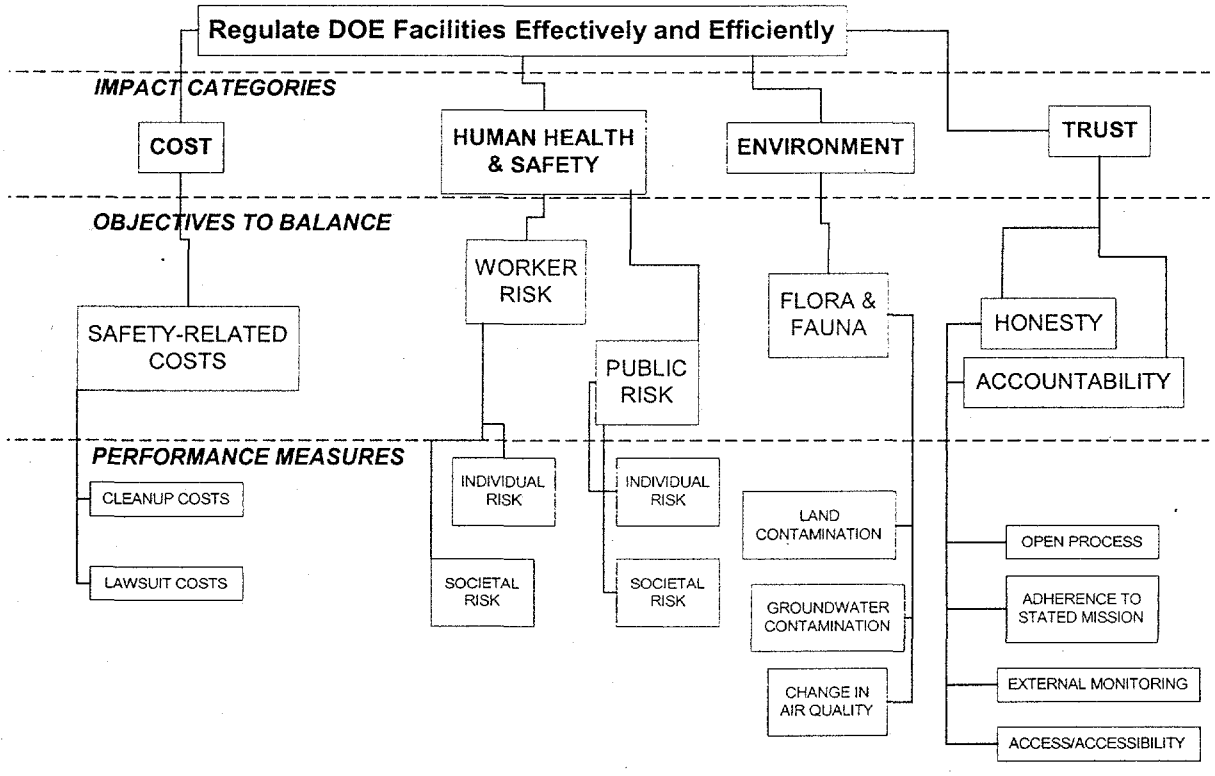

Figure 2. Example value tree for DOE regulatory system.

Task 4: Develop Preliminary Framework (MIT)

Figure 3 illustrates a preliminary PBR framework that was developed based on Reference 10. This broad-based framework was constructed so that it is applicable to DOE reactor facilities. However, similar diagrams were constructed for other key facility groups.

The proposed framework includes several high-level objective(s) or "cornerstones" (Issues 1 and 3), such as ensuring the public health and the safe operation of the facility, with numerical values to define "acceptable" levels. In the future, the framework will be further refined to include lower-order objectives, such as adequate SM and plant safety performance. In order to ensure that the cost of achieving such objective(s) is reasonable, it is anticipated that "acceptable" values of these lower order objectives will include appropriate cost/benefit considerations. These lower order objectives must be defined in a way to ensure that the selected criteria are consistent and complete. In Figure 3, it is proposed that these lower order objectives be organized according to the various components found in plant risk assessments.

The framework will consider how compliance with these lower order objectives should be demonstrated and what types of corrective actions are appropriate when performance goals are not met. The framework will identify appropriate databases reflecting experience feedback, such as building leakage rates and failure data for systems, structures, and components, which could be used to demonstrate compliance with the proposed performance goals.

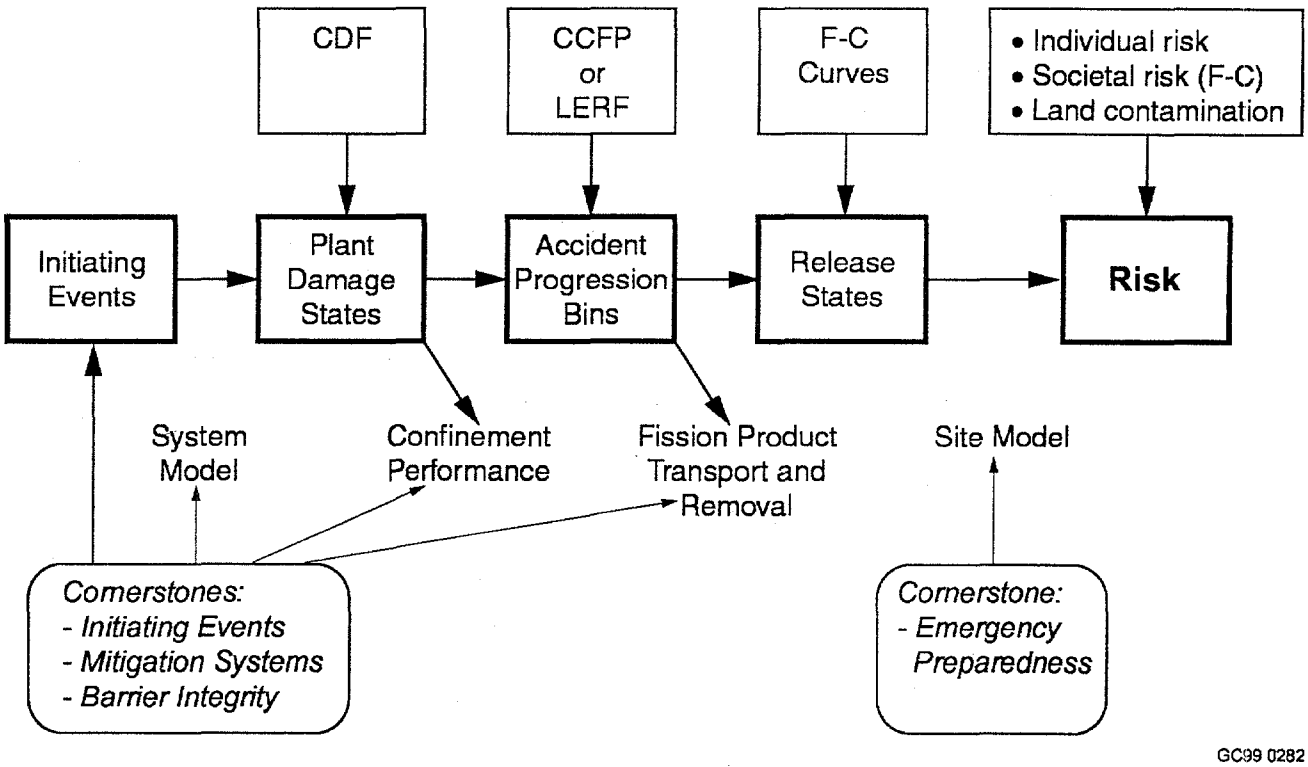

Figure 3. Proposed Framework for DOE Reactor Facilities

When considering the merit of a new PBR framework, we must consider the cost of transition to and implementation of the new framework as well. The benefits of the new framework would have to outweigh these costs in order to be lucrative for the regulated entity (the DOE). It is anticipated that this framework will be revised as case studies are performed.

In formulating our PBR framework for well-characterized complex DOE facilities (e.g., the ATR or ISFSI), we can draw from the substantial progress in PBR studies for commercial nuclear power plants and long-term High Level Waste/Spent Nuclear Fuel (HLW/SNF) repositories. There is less guidance available for PBR of less complex, lower potential hazard facilities, for which risk assessments may exist in a less formal format. Furthermore, it will be challenging to formulate a PBR framework for those inherited DOE facilities which keep/kept few operational records, and may contain poorly characterized materials (e.g., the Hanford HLW tanks).

Measures of Success

Several measures of success indicate that this project will accomplish its planned objectives. Project goals and status information are listed below .

- The first goal for this year was to plan tasks for this three-year research program that could accomplish our project objectives. This work was completed during the first part of this fiscal year and is documented in a status report that was submitted to the LDRD office. All research tasks defined in this status report are on schedule (although some modifications were made to reflect changes discussed during the April program review meeting).
- A second goal for this year is to increase visibility of this project. This goal is being accomplished using several mechanisms. First, we are documenting results from tasks in technical reports. Two letter reports have already been issued. A third report will be issued during FY99. Second, we are participating in technical conferences. Two project papers have been accepted for presentation and publication at international technical conferences (the 1999 International Conference on Probabilistic Safety Assessment and the International Conference on Public Participation). Furthermore, a session will be devoted to discussing results from this research at the Eighth International Conference on Nuclear Engineering (ICONE8) in Baltimore, Maryland, April 2000. Third, research results will be discussed with experts during workshops. The first project workshop was held during June 1999. A second is planned for August 1999.
- A third goal for this project is to receive direct funding. During this year, we contacted individuals at DOE-NE about this project and tools and methods that are being developed as part of this project. As a result of a presentation to Mr. Bill Magwood, Director, DOE-NE, DOE has agreed to fund additional development of the NUFAC database to assist the DOE-NE Roadmapping effort. INEEL hopes to expand this database to assist other areas of DOE in their Roadmapping efforts (DOE-EM). During FY2000, INEEL will brief DOE about project results with example PBR that could be implemented to improve the safety, availability, and economics of facility operation.
- Another performance goal is to conduct research worthy of MIT graduate level degrees. The MIT research team includes four graduate students pursuing doctoral degrees. Thesis topics and student names are:
  - A Risk-informed Performance-based (RIPB) Regulatory framework that can be Applied to the Diverse Range of Radioactive Waste (i.e., HLW, LLW, TRU) Storage Systems at DOE Facilities – S. Tina Ghosh
  - From a Deterministic to a Performance-based Regulation: Application to Test and Research Reactors - Emanuele Borgonovo
  - Reliability of Electric Power Systems under Generation Deregulation – Frank Felder
  - Development of a Rationale of Safety Goals for Nuclear Power Plants – Yu Sui

Any Problems

None. As discussed above (see Table 2), project tasks are progressing on or ahead of schedule.

3. Brief Summary of Path Forward

During our first fiscal year, project funds helped us complete necessary groundwork. The project tasks were identified. Background information pertaining to NRC and DOE regulation and DOE facilities was reviewed. Finally, we selected DOE facilities for case studies, obtained concurrence from facility personnel to participate in this project, and conducted preliminary interviews with facility staff to understand the process currently used by DOE to regulate their facilities. A framework for applying PBR to these case studies is being developed that identifies "cornerstones" required for facility regulation.

During the remainder of this project, the INEEL/MIT team will concentrate on applying this preliminary framework to the case studies. This application will require interactions with facility personnel and with other regulatory experts to ensure that it is reasonable, complete, and more efficient (less costly) than the existing regulatory process.

4. References

1. T. Leahy, et al., Idaho National Engineering and Environmental Laboratory and Massachusetts Institute of Technology Strategic Nuclear Research Collaboration Program Plan, July 1998.
2. Letter from Dr. Malcolm A. Weiss, MIT, to Dr. James A. Lake, Peer Review Comment Transmittal, September 21, 1998.
3. G. Apostolakis, et al. Review of Applicable U.S. Department of Energy and U.S. Nuclear Regulatory Commission Activities (Project Task 1), MIT/NE-316, June 1999.
4. J. L. Rempe, et al., Case Studies to Investigate Performance-Based Regulation at DOE Facilities (Project Task 2), INEEL Letter Report, INEEL/EXT-99-00535, June 1999.
5. "Status of the Integrated Review of the NRC Assessment Process (IRAP) for Operating Commercial Nuclear Power Plants, SECY-98-045, March 1998.
6. NEI, "A New Regulatory Oversight Process: Towards Risk-Informed Performance-Based Assessment, Inspection and Enforcement," September 10, 1998.
7. NRC Staff, "Risk-informed, Performance-based and Risk-informed, Less-prescriptive Regulation in the Office of Nuclear Material Safety and Safeguards," NRC SECY-98-138, June 11, 1998.
8. NRC Staff, "Proposed Resolution to Petition for Rulemaking Filed by the Nuclear Energy Institute," NRC SECY-97-137, June 30, 1997.
9. NRC Staff, "Proposed Rulemaking - Revised Requirements For The Domestic Licensing of Special Nuclear Material," NRC SECY-98-185, July 30, 1998.
10. J. N. Sorensen, G. E. Apostolakis et al, "On the role of Defense in Depth in Risk-Informed Regulation," *to be presented at PSA'99*, Baltimore, Maryland, August 1999.

Publication/Presentation Information

Title	Journal/ Conference	Status/Date	Authors
Developing a Performance-Based Regulatory System for DOE Facilities	1999 International Conference on Probabilistic Safety Assessment (PSA '99), Washington, D.C., August 1999	Accepted paper submitted for publication. Will be presented August 1999. (Conference requires peer-reviewed paper summary).	J. R. Rempe, et al., INEEL, G. Apostolakis, et al., MIT, and A. Stevens, JAC
Adding Rigor to Your Public Involvement Methods	1999 International Association for Public Participation (IAP2), Banff, Canada	Summary submitted.	G. Apostolakis, MIT, A. Stevens, JAC, and T. Ghosh, MIT
Performance–based Regulation for DOE Facilities, Session entirely composed of papers on this topic.	Eighth International Conference on Nuclear Engineering (ICONE8), Baltimore, Maryland, April 2000.	Session, chaired by Tim Leahy and George Apostolakis, approved by ICONE8 Technical Program Committee (Conference requires that full papers be peer-reviewed according to ASME standards)	At least 5 papers describing various aspects of the program will be submitted.
Review of Applicable U.S. Department of Energy and U.S. Nuclear Regulatory Commission Activities (Project Task 1)	MIT Letter Report, MITNE-316	June 1999.	G. Apostolakis, et al., MIT, and T. Leahy, et al., INEEL.
Case Studies to Investigate Performance-Based Regulation at DOE Facilities (Project Task 2)	INEEL Letter Report, INEEL/EXT-99-00535	June 1999.	J. L. Rempe, T. J. Leahy, and D. L. Knudson; INEEL; G. Apostolakis, et al., MIT.
Upcoming Expert Workshop at MIT with: -Andrew Marchese, former DOE and NRC Manager -Ray Furstenau, DOE-ID, ATR Facility Manager -Tom Kress, former ACRS member -Gary Zeman, LBNL, DOE/NRC Pilot Plant Participant -Kim Thompson, Assistant Professor, Harvard School of Public Health -Gary Marshall, ANL-W, ES&H Coordinator. -Additional experts from NRC, ATR, and ISFSI.		August 3-4, 1999	MIT/INEEL/JAC

Participant Information

Participants	Educational Status (professor, post-doctorate, grad student, etc.)	Fraction of time spent on project (per fiscal year Oct 1-Sept 30)	(Students only) Degree Program	(Students only) Starting date	(Students only) estimated finishing date
G. Apostolakis, MIT	Professor	1/3			
M. Golay, MIT	Professor	1/5			
M. Chaniotakis, MIT	Research Scientist	1/3			
T. Ghosh, MIT	Graduate Student	Full	Doctoral	January 1999	May 2002
F. Felder, MIT	Graduate Student	1/3	Doctoral	September 1997	May 2000
E. Borgonova, MIT	Graduate Student	Full	Doctoral	January 1999	May 2002
Y. Sui, MIT	Graduate Student	1/3	Doctoral	June 1998	June 2000
J. Rempe, INEEL	Consulting Engineer	1/3			
T. Leahy, INEEL	Department Manager	1/8			
D. Knudson, INEEL	Advisory Engineer	1/3			
T. Wierman, INEEL	Engineering Specialist	1/8			
A. Stevens, JAC	Public Participation Specialist	1/6			

Personnel Exchange Information

Name of University Personnel who spent time at INEEL Laboratories – and dates attended

T. Ghosh, F. Felder, E. Borgonova, G. Aposotolakis, and M. Golay, MIT, visiting INEEL and ANL-W facilities and discussing questions with facility employees, July 1999.

Name of INEEL personnel who spent time at your University Laboratory - and dates attended

T. Leahy, J. Rempe, and D. Knudson, INEEL, and A. Stevens, Jason Associates Corporation, attended MIT/INEEL April Program Review Meeting, April 1999.

J. Rempe and D. Knudson, INEEL, and A. Stevens, Jason Associates Corporation, attended MIT workshop featuring Roxanne Summers, former NRC participant in NRC/DOE Pilot Projects, June 1999.

J. Rempe, T. Leahy, and D. Knudson, INEEL, and A. Stevens, Jason Associates Corporation, participation at expert workshop at MIT, August 1999

"Follow-On" Research Proposals

Title	Agency	Proposers	Status
NUclear FACility (NUFAC) Database for DOE-NE Roadmapping Efforts	Department of Energy Office of Nuclear Science and Technology (DOE-NE)	J. Rempe, D. Knudson, and T. Wierman	Proposal for multi-year project accepted. DOE submitted reprogramming proposal to Congress for FY99 funding.

Attachment A

Review of Applicable U.S. Department of Energy and U.S. Nuclear Regulatory Commission Activities (Project Task 1)

Regulatory Excellence Project:
Performance-Based Regulatory Framework for U.S. Department of Energy Facilities

Review of Applicable U.S. Department of Energy and U.S. Nuclear Regulatory Commission Activities (Project Task 1)

June 1999

MITNE-316

MIT

George Apostolakis (Principal Investigator)
Michael Golay (Principal Investigator)
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Regulatory Excellence Project:
Performance-Based Regulatory Framework for U.S. Department of Energy Facilities

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Table of Contents

List of Acronyms

ACRS: Advisory Committee on Reactor Safeguards (to the U.S. Nuclear Regulatory Commission)
AEC: Atomic Energy Commission
ALARA: As Low As Reasonably Achievable
AOT: Allowed Outage Times
ATR: Advance Test Reactor
CCFP: Conditional Containment Failure Probability
CCDF: Complementary Cumulative Distribution Function
CCFP: Core Coolant Fission Products
CDF: Core Damage Frequency
CDP: Core Damage Probability
CFR: Code of Federal Regulations
DID: Defense-in-depth
GDC: General Design Criteria
HLW: High-level Waste
IPE: Independent Plant Evaluation
ISA: Integrated Safety Assessment
LB: Licensing Basis
LERF: Large Early Release Frequency
LLW: Low-level Waste
LWR: Light-water Reactor
MITR: MIT Reactor
MOW: Model of the World
NEI: Nuclear Energy Institute
NMSS: Nuclear Material Safety and Safeguards
NPP: Nuclear Power Plant
NRC: United States Nuclear Regulatory Commission
PA: Performance Assessment
PBR: Performance-based Regulation
PI: Performance Indicator
PSA: Probabilistic Safety Assessment
PSHA: Probabilistic Seismic Hazard Analysis
PRA: Probability Risk Assessment
PWR: Pressurized Water Reactor
RG: Regulatory Guide
RI: Risk-informed
RIPBR: Risk-informed, Performance-based Regulation
RIR: Risk-informed Regulation
RI-ISI: Risk-Informed In-Service Inspection
SAR: Safety Analysis Report
SM: Safety Margins
SNF: Spent Nuclear Fuel
SNM: Special Nuclear Material
SRM: Staff Requirements Memorandum
STI: Surveillance Test Intervals

TFI: Technical Facilitator Integrator
TI: Technical Integrator
TMI-2: Three Mile Island Unit 2
TRU: Transuranic Waste
USDOE or DOE: United States Department of Energy
U/TH: Uranium/Thorium
VLLW: Very Low-level Waste

List of Figures

List of Tables

1.0 Overview of the Project

Performance-based regulation (PBR) can potentially enhance safety, minimize costs by avoiding unnecessarily burdensome requirements, and prevent unwarranted interruptions in production or processing. Although some have suggested that the United States Nuclear Regulatory Commission (NRC) should regulate U.S. Department of Energy (DOE) facilities, PBR has significant potential benefits regardless of the regulator's identity. In recognition of the importance of PBR, the Idaho National Engineering and Environmental Laboratory (INEEL) and the Massachusetts Institute of Technology (MIT) have initiated a project to develop a framework for selecting and applying PBR criteria for regulating DOE facilities.

The overall goal of regulation is to ensure safe operation of potentially hazardous facilities. Ensuring safety entails managing risk effectively. While the current regulatory framework for the DOE manages risk effectively, there may be more efficient ways to do so. As was found in the analogous case of NRC regulation of commercial reactors, the DOE's current regulatory system may place undue burden on itself through instances of regulatory requirements that do not improve safety significantly, or whose results could be achieved more efficiently.

Traditionally, regulations have been primarily prescriptive and deterministic, where safety is achieved through conservatism. Design requirements through technical specifications of facility components and other deterministic criteria are set at target levels much higher than necessary to achieve an acceptable level of safety. These conservative criteria come from the concepts of "defense-in-depth" and "safety margins," which were developed to ensure that regulatory compliance is achieved in an uncertain world (see Section 3 for discussion).

Using a performance-based regulatory (PBR) framework instead is one way to regulate effectively and more efficiently. A performance-based approach to regulation relies on "measurable (or calculable) outcomes (i.e., performance results) to be met, but provides more flexibility to the licensee as to the means of meeting those outcomes" [NRC White Paper]. This focus on outcomes and flexibility leads to a more efficient regulatory system, which in turn should lead to cost savings and better allocation of safety-related resources within the regulated organization.

The use of risk information can help allocate resources in accordance with safety significance. "A 'risk-informed' approach to regulatory decision-making represents a philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety" [NRC White Paper].

A Risk-Informed Performance-Based approach (RIPB) would combine the key elements of both risk-informed regulation (RIR) and PBR (as listed above). The use of risk information in PBR should result in additional efficiency in the regulatory system. PBR does not have to be risk-informed. RIR typically implies the use of quantitative safety analyses, such as Probabilistic Risk Assessment (PRA), Performance Assessment (PA), or Integrated Safety Analysis (ISA). Such safety analyses are available for some, but not all, DOE facilities. In the cases where the

utility of quantitative safety analysis is low,[1] PBR framework can be developed without such assessments. Thus, the extent to which risk information will be used in our proposed PBR framework may vary from one group of DOE facilities to another.

The first task of this project is to review applicable DOE and NRC regulatory requirements to gain perspective about the intent and scope of existing regulations. In addition, on-going regulatory initiatives, including NRC and industry interactions, are examined. The purpose of this report is to codify efforts to date. Section 2 reviews the current regulatory framework for nuclear facilities highlighting the use of defense-in-depth (DID) and safety margins (SM) in nuclear facilities, primarily commercial nuclear power reactors. In Section 3, we provide more formal definitions of RIR and PBR, and describe on-going NRC and industry initiatives in these areas. Section 4 reviews pilot programs conducted by the NRC and the DOE to determine how new and existing DOE facilities and operations might be regulated to better ensure nuclear safety. Section 5 reports on lessons learned from this review that can be applied when developing a PBR framework for DOE facilities.

[1] This occurs when the benefits of results from such an analysis do not outweigh the costs of performing the analysis.

2.0 Existing Regulations for Nuclear Facilities

2.1 The Management of Uncertainty

The purpose of regulating nuclear facilities is to protect public health and safety. In striving towards this goal, nuclear regulation has always acknowledged the uncertainty inherent in anticipating and guarding against accidents. In the past, the NRC had exclusively used qualitative evaluations of risks, based on engineering judgement and experience, to carry out its mission. The response to uncertainty that cannot be quantified is to use the concepts of defense-in-depth (DID) and safety margins (SM).[2]

The definition of DID has evolved over time. At the start of the nuclear industry (1953-55), DID was commonly defined as the principle that no single element or barrier would be emphasized to the exclusion of others. This definition implies that multiple barriers should exist to prevent release of radioactive material. For light water reactor nuclear power plants, these barriers are the fuel matrix, metal cladding, the reactor coolant pressure boundary, and the containment. More recently, DID has been thought of as an overall safety strategy [Sorenson, *et al*]. It emphasizes the importance of the balance between mitigation and prevention.

Safety margins (SM) ensure the adequate performance of systems structures and components by over designing equipment and systems to account for usage outside of normal operating parameters. This can occur due to abnormal operating conditions or due to the uncertainties associated with measuring parameters.

DID and SM have been the historical approach to the treatment of uncertainty. The completion of the *Reactor Safety Study* [WASH-1400] enabled uncertainty to be quantified and incorporated explicitly into safety analysis, although difficulties remain regarding how to treat model uncertainty. With the ability to treat uncertainty more formally than in the past, the logical question is what should be the role of DID and SM?

2.2 Nuclear Power Plant Safety Goals

The Atomic Energy Act made possible the civilian use of nuclear energy. The act defined responsibilities for ensuring the safe use of nuclear technology in qualitative terms. It instructed the Atomic Energy Commission (AEC) to "provide adequate protection to the health and safety of the public" from radiological hazards but did not specify what was meant by "adequate protection." As a result, the AEC and its successor, the NRC, relied on imprecise criteria such as "adequate protection," or "reasonable assurance of no undue risk" to evaluate applications for plant licenses.[3] The technical staff of the AEC and NRC were left with the task of writing rules and supporting regulatory guides that defined the engineering requirements to be met by applicants to receive licenses to construct and operate a reactor.[4]

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[2] See, for example, [ACRS Letter 1998] and [ACRS Letter 1997].

[3] "Risk" is defined as the risk related to the "release of radioactive materials from the reactor to the environment during normal as well as accidental situations" [FR, 1986].

[4] The NRC, pursuant to the Atomic Energy Act, promulgates the current regulation concerning civilian nuclear activities. The regulations are largely prescriptive.

Prior to the Three Mile Island Unit 2 (TMI-2) accident in March 1979, there had been a steady increase of interest in the United States in the use of quantitative safety goals to define safety requirements for nuclear power plants. It was not until after the accident, however, that the NRC undertook a large-scale effort to develop safety goals. The accident greatly increased the impetus to determine quantitatively what level of safety was safe enough. Following the accident, the President's Commission on the TMI-2 Accident, the NRC's Advisory Committee on Reactor Safeguards (ACRS), and the NRC's Special Inquiry Group all strongly recommended that the NRC should spell out more clearly its reactor safety objectives by establishing quantitative safety goals. Those recommendations, along with the agency's own recognition of the need to rethink past assumptions and policies in light of the experiences at TMI-2, motivated the NRC to develop safety goals.

The Safety Goal Policy Statement [NRC Aug. 1986] includes two qualitative goals and two quantitative goals. The qualitative goals are the following:

1. Individuals should "bear no significant additional risk to life and health" from nuclear power operation; and
2. Societal risks "should be comparable to or less than the risks of ... viable competing technologies and should not be a significant addition to other societal risks"

The quantitative objectives are the following:

1. The risk to an average individual living near a nuclear plant should not increase the risk of fatality from an accident of more than one-tenth of one percent of the sum of "prompt fatality risks resulting from other accidents"; and
2. The risk to the population within ten miles of a nuclear plant of dying from cancer should not increase by more than one-tenth of one percent beyond the sum of cancer fatality risks from all other causes. It also included a general performance guideline that "the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation."

With the experience gained in the application of these safety goals and the advances in Probabilistic Risk Assessment (PRA), the major tool for showing compliance with the safety goals, additional insights about the safety goals and their implementation has been gained. As a result, some modifications have been made including the following:

1. The general performance guideline (the large release frequency guideline) has been removed from the final policy statement because it was found to be much more stringent than the quantitative goals; and
2. Two subsidiary goals that the industry and the NRC staff have agreed to are being used in lieu of the quantitative health objectives. They are a Core Damage Frequency (CDF) goal of 1 in 10,000 per year of reactor operation and a Large Early Release Frequency (LERF) goal of 1 in 100,000 per year of reactor operation.

Some other modifications that the ACRS recommended be given consideration include the following:

1. Elevation of the CDF goal to the status of a fundamental goal;
2. Modification of the quantitative goal treating societal risk;
3. Addition of goals for land contamination and interdiction; and
4. Addition of goals for temporary risk increases (e.g., as may arise from particular plant configurations).

The safety goals presented the Commission's judgment on acceptable risk from nuclear power generation. They gave a definition to "how safe is safe enough" and provided a yardstick for nuclear safety. The safety goals help to identify the systems and activities that are most important with respect to risk, to allocate more efficiently resources both for the regulators and the industry, and to maintain a coherent and consistent regulatory system. It was a major step toward the use of risk-informed insights in making regulatory decisions and an important milestone in the evolution of the NRC's approach to regulation. Despite all the advantages the safety goals may offer, it is generally recognized that safety goals should complement but not replace traditional safety analyses and reliance on DID, which is mainly due to the large uncertainties in PRA analysis and in demonstrating compliance with the goals.

2.3 Types of Facilities Examined

In order to understand better the regulatory problems inherent in the types of facilities used in the DOE's activities, we examined the current regulatory practices at several such facilities.

2.3.1 Commercial Power Reactors

The NRC regulatory policy is based on three basic "lines of defense" for nuclear reactors. These lines of defense are the following [Sorenson, *et al*]:

1. Prevention of accident initiators through superior quality of design, construction and operation;
2. Prevention of accident escalation through engineered safety systems; and
3. Minimization of fission products release through consequence-limiting safety systems.

Most of the current regulations have been developed for commercial light water reactors power plants. They are design-based, that is, they are formulated in terms of required systems and plant features to either prevent or mitigate possible accidents. "Structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed" [10 CFR 50]. This statement clearly reveals the DID philosophy. In fact, if the stated conditions are satisfied, adequate balance between prevention and mitigation is presumed to have been achieved.

Prevention of accidents is the main idea underlying the first line of defense. Accident initiators are to be minimized. In accordance to the DID philosophy, the concept of safety limits

is introduced. Safety limits are defined as "limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radiation" [10 CFR 50.36].

To understand better the underlying philosophy, it is useful to look at some of these criteria in detail. 10 CFR 50, Appendix A General Design Criterion (GDC) Criterion 10, Reactor Design, states that "The reactor core and associated coolant shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition or normal operation, including the effects of anticipated operational occurrences". Criterion GDC 36, Inspection of Emergency Core Cooling System, states that "The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles and piping, to assure the integrity and capability of the system".

Regarding the level of radioactivity in the primary coolant: "...the application shall also identify the design objectives and the means to be employed for keeping levels of radioactive materials...as low as reasonably achievable". "Reasonably achievable" means taking into account the current technology and the economics of improvements in relation to the benefits to society and to the use of nuclear energy.

The second line of defense is intended to guarantee that reactors are equipped with safety systems adequate to inhibit possible accident sequences. These systems are again considered under the DID and SM Principles. Especially after the TMI-2 accident, additional attention has been placed on safety related systems. For example, safety related electric equipment is defined as the equipment that is relied upon to remain functional during and following design basis events to ensure:

1. The integrity of the reactor coolant pressure boundary;
2. The capability to shut down the reactor and maintain it in a safe shutdown condition; and
3. The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure.

The electric equipment qualification is based on physical and technological properties, and sufficient margins are to be applied in dealing with uncertainties. "Margins must be applied to account for unquantified uncertainty, such as the effects of production variations and inaccuracies in test instruments. These margins are in addition to any conservatism applied during the derivation of local environmental conditions of the equipment unless these conservatisms can be quantified and shown to contain appropriate margins" [10CFR50]. In assessing accident progression, several uncertainties appear. The current regulations require uncertainties to be dealt with using a conservative approach. That is, no matter what its probability is, the worst possible scenario will be always examined and designed against. In this conservative approach, design margin is the prescribed means by which safety is reasonably assured.

The third line of defense aims at minimizing fission product release through consequence-limiting safety systems. Reactor containment is the final barrier against radiation

release after the core has been damaged. The licensee is required to provide containment isolation. The licensee is also asked to develop a safety study to demonstrate that containment integrity will be maintained during an accident under "worst case scenarios."

2.3.2 Research Reactors and Hot Cells

To provide specific examples of how the NRC regulates non-DOE research reactors and hot cells, the regulatory requirements that apply to MIT's research reactor (MITR), were reviewed.[5] These two types of facilities should be considered together because it is common to have hot cells with a research reactor. Hot cells vary in size and uses but can be simply thought of as compartments consisting of thick walls, usually containing large amount of lead, with thick lead-based glass and remote operating devices to manipulate and process radioactive material. Hot cells may be used for research purposes, such as testing the strength of various materials that have been exposed to significant amounts of radiation in a research reactor, or for industrial uses such as the manufacturing of particular radiochemical materials.

In the case of university reactors, the NRC regulates their design, operation, and use fundamentally similar to the way it regulates power reactors. The regulations are prescriptive, detailed, and require audit and verification by the NRC in order to ensure compliance on a routine basis.

NRC's regulation of university research reactors does, however, acknowledge some of the differences between research and power reactors. One fundamental difference is that research reactors contain significantly less radioactive material than power reactors and, therefore, pose less of a hazard. Research reactors also operate at lower temperatures and pressures than power reactors so there is less stored energy to push radioactive material out in the event of an accident. In addition, their fuel types are different. University research reactors typically use aluminum clad cermet fuels whereas power plants have zircaloy rods containing UO_2 fuel pellets. If a power plant fuel rod ruptures, there is an instantaneous release of fission product gas. If the cladding of a university research reactor element, however, fails, then the fission product gases have to diffuse through the cermet, which is a slower process than what would occur in a power reactor.[6]

As a result of these differences, the corresponding NRC regulatory requirements are different. For example, regional evacuation plans are not required for university research reactors because the amount of fissionable material that can be released to a large area is significantly less than that of a power reactor and therefore does not present a significant hazard to the nearby population. The principal hazard from a research reactor is in radiation of the people who use them for experiments (e.g., as a result of radiation from the beam ports or radiation from the samples, if the samples are used improperly). NRC-regulated research reactors do not have to have a PRA. Such a requirement, it is believed, would be too costly and not necessary given the relatively low amount of radioactive material that exists in the core.

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[5] This information is based on interviews of various MIT personnel involved in the safety and operation of MIT nuclear facilities.

[6] Other differences that may exist are that research reactors may not produce electricity and may have beam ports whereas commercial reactors do not.

The NRC usually conducts three inspections a year at these MIT facilities. One inspection deals with reactor operations, another addresses health physics and emergency planning, and the third deals with some other issue such as security, special nuclear material, or a topic of interest to the NRC. MIT is obligated to operate its facilities in accordance with NRC approved Technical Specifications and must report any violation to the NRC within 24 hours. The NRC also licenses the reactor operators and administers the associated examinations.

2.3.3 Storage Facilities

The nature and magnitude of risks posed by a radioactive waste storage facility can be very different from those posed by a reactor for several reasons:

- The threat from radio-toxic materials may not be as concentrated as in a reactor core;
- Reactors are active units in operation, while storage units are often passive facilities that are not serviced regularly;
- Waste storage units encompass a very wide range of hazards from very low-level waste (VLLW) all the way to high-level waste (HLW) and spent nuclear fuel (SNF); and
- Waste storage units at DOE sites exist in a diversity of conditions from recently designed and well-characterized storage units in good condition to poorly characterized inherited waste from the cold war years.

Current commercial LLW[7] storage facilities are licensed by either the NRC or Agreement States. Regulations dictate that (1) the LLW be stored in a manner appropriate to its level of hazard, (2) radiation doses to workers and members of the public must be kept below NRC-specified levels, and furthermore, as low as detected by the reasonably achievable (the as low as reasonably achievable (ALARA) principle). Regulatory criteria include administrative details such as requirements for clear markers and postings in areas where LLW is stored, to prevent inadvertent radiation exposure to workers or the public. Regulatory criteria for LLW disposal address such topics as siting, design, and operation requirements, such as mandatory maintenance and monitoring activities and restricted access to the site [NUREG/BR-0216]. A safety analysis in the form of a Performance Assessment (PA) is not required for a LLW storage or disposal facility (see Section 3.6 for discussion on Performance Assessments).

Commercial HLW/SNF storage is regulated by the NRC. NRC licenses both spent fuel pools for wet storage, and metal or concrete casks for dry storage and transportation. Safety analyses are required for HLW/SNF storage systems, but not PAs. No HLW/SNF has been disposed yet in the US, but regulatory criteria (including PA) for a potential HLW/SNF repository, and criteria that the EPA used to license the Waste Isolation Pilot Plant for transuranic waste in May 1998, are discussed in Section 3.6.

Current NRC and some EPA regulations that could be applicable to DOE storage facilities were reviewed. These regulations are identified and brief insights gained are described in the remainder of this section.

[7] e.g., from nuclear power plants or hospitals.

NRC's 10 CFR PART 20 contains the Standards For Protection Against Radiation, which reflects NRC's basic protection principles. This rule suggested the idea that some existing regulatory requirements may be utilized in a PBR framework and might enable a smoother transition to a PBR framework. For example, data-keeping and performance prediction requirements in 10 CFR 20 require that whenever a worker is likely to be exposed to radiation above a particular dose level, a pre-activity assessment must be made of the dose that is anticipated to be received. This predicted dose, as well as the actual measured dose received through the activity, must be recorded. Recording the actually measured dose is necessary to track the cumulative yearly dose of each worker. The additional requirement of predicting the dose suggests that we might find such instances of potential in the current regulatory system, e.g., information that is required to be collected anyway, which may be used for more efficient regulation and help smooth the transition to RIPBR.

NRC's 10 CFR PART 72—Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste would be relevant for high risk waste storage units, and was applied to the INEEL TMI-2 Independent Storage Facility Safety Installation. This rule governs the license application, issuance and conditions; records, reports, inspections and enforcements; siting evaluation factors; general design criteria; and quality assurance requirements. These are largely prescriptive criteria in the form of "minimum requirements". However, the licensee has some flexibility in showing how the system to be licensed meets those requirements.

NRC's 10 CFR PART 70—Domestic Licensing of Special Nuclear Materials[8] could be relevant to medium risk waste storage facilities. NRC's PBR-directed revision of this rule is discussed in Section 3.6.

EPA's 40 CFR Part 192, Health and Environmental Protection Standards for Uranium and Thorium Mill Tailings, could be relevant to low to medium risk waste storage facilities as well as contaminated sites. Section 40 CFR 192 suggests cost-benefit analysis for those activities that do not pose high risks to workers and the public. For example, there are primary standards that must be met at Uranium/Thorium (U/Th) mill tailings sites. A licensee does not have to take any action if an assessment shows that there is a reasonable expectation that contamination (e.g., from radon-222) will not exceed harmful levels for 200 to 1,000 years. However, if soil or groundwater sampling or analysis show that the standard will be exceeded, then corrective action must be taken. There are, however, supplemental standards for special situations. For example, the implementing agency may apply different standards if "the estimated cost of remedial action... is unreasonably high relative to the long-term benefits, and the residual radioactive materials do not pose a clear present or future hazard." Different standards may be used if "the restoration of groundwater quality at any designated processing site... is technically impracticable from an engineering perspective." This clearly shows not

[8] Special Nuclear Materials means "(1) plutonium, uranium 233, uranium enriched in the isotope 233 or in the isotope 235, and any other material which the Commission pursuant to the provisions of section 51 of the act, determines to be special nuclear material, but does not include source material; or (2) any material artificially enriched by any of the foregoing but does not include source material" [10 CFR 70.4]. Source Material means "(1) Uranium or thorium, or any combination thereof, in any physical or chemical form or (2) Ores that contain by weight one twentieth of one percent (0.05%) or more of (i) Uranium, (ii) Thorium, or (iii) any combination thereof. Source material does not include special nuclear material" [10 CFR 72.3].

only a risk-informed framework, but also a regulatory framework that takes cost-benefit tradeoffs into account.

40 CFR 191, 40 CFR 194 and 10 CFR 60 could all be relevant to high-risk storage facilities or contaminated sites. EPA's 40 CFR 191 describes the Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes. EPA's 40 CFR 194 describes the Criteria for the Certification and Re-certification of the Waste Isolation Pilot Plant's Compliance with the 40 CFR Part 191 disposal regulations. NRC's 10 CFR 60 and 10 CFR 63 are applicable for SNF/HLW disposal at Yucca Mountain. All of these use performance-based standards to some extent, and are discussed in Section 3.4 on Performance Assessments.

2.3.4 Other Low Hazard Facilities and End-Use Facilities

How does performance-based regulation mean in the context of very low hazard or other end-use facilities? The subset of DOE facilities that pose very little hazard may be the category for which it is most difficult to construct a PBR scheme. If the risk is very low, how can PBR be applied? Insights derived from PRA are the basis for choosing performance indicators for the proposed PBR of commercial reactors. However, a full-scale PRA (or equivalent analysis) should not be required of facilities that pose little threat to its workers and the communities that surround them. The resource expenditure to complete such a detailed safety analysis would not be justifiable. So performance indicators must be chosen on a different basis.

The category of low hazard facilities suggests the need to develop a systematic way to classify the various DOE facilities. For example, Table 1 is adapted from NUREG/CR-6372 on earthquake hazard analysis for reactors. There are three degrees of complexity of issues:

1. Non-controversial; and/or insignificant hazard;
2. Significant uncertainty and diversity; controversial; and complex; and
3. Highly contentious; significant to hazard; and highly complex.

For each of these issues, decision factors are identified to help decide the level of study needed for the issue. A similar classification scheme for DOE facilities could be developed.

Table 1: Degrees of Probabilistic Seismic Hazard Analysis (PSHA) Issues and Levels of Study [NUREG/CR-6372]

ISSUE DEGREE	DECISION FACTORS	STUDY LEVEL
A: Non-controversial; and/or insignificant to hazard	Regulatory Concern	**1.** Technical Integrator (TI) evaluates/weights models based on literature review and experience; estimates community distribution
B: Significant uncertainty and diversity; controversial; and complex	Resources Available	**2.** TI interacts with proponents and resource experts to identify issues and interpretations; estimates community distribution
C: Highly contentious; significant to hazard; and highly complex	Public Perception	**3.** TI brings together proponents & resource experts for debate and interaction; TI focuses debate and evaluates alternative interpretations; estimates community distribution
		4. Technical Facilitator Integrator (TFI) organizes panel of experts to interpret and evaluate; focuses discussions; avoids inappropriate behavior on the part of evaluators; draws picture of evaluators' estimate of the community's composite distribution; has ultimate responsibility for project

The idea of classification of risks as a gauge of importance is common. It is similar to classifying various components in a system according to their contribution to safety. For example, licensees of spent fuel storage systems are expected to classify the systems structures, and components (SSCs) into broad categories according to their importance to safety. The NRC guidance on this is contained in NUREG/CR-6407. Table 2 lists the SSC classification categories used. This is similar to SSC classifications that the DOE already uses (e.g., in Safety Analysis Reports (SAR)).

Taking a holistic view of the entire set of DOE facilities, these facilities can be classified according to a similar scheme. Low hazard facilities would fall into the C category, since a failure or mishap at these facilities would not be likely to create a situation adversely affecting public health and safety. One level (presumably a low level) of regulations would apply to these. High hazard facilities would fall into category A, requiring a PRA, or equivalent, along with demonstration of DID. In addition to the pure safety-related classification in Table 2, classification criteria could include non-safety-related factors. One example is the decision factor in Table 1 labeled "public perception." This could be an additional classification criterion or decision factor that leads to a higher level of regulation of a low hazard facility than would be suggested by the nature and level of hazard alone. The question of whether factors other than

safety should be considered in a safety regulatory system is ultimately a policy issue that must be resolved by the agency of interest.

Table 2: Description of Classification Categories for Components of Dry Spent Nuclear Fuel Storage Systems [NUREG/CR-6407]

CLASSIFICATION CATEGORY	IMPORTANCE TO SAFETY	DESCRIPTION
A	Critical to operation	Category A items include structures, components, and systems whose failure could directly result in a condition adversely affecting public health and safety. The failure of a single item could cause loss of primary containment leading to release of radioactive material, loss of shielding or unsafe geometry compromising criticality control.
B	Major impact safety	Category B items include structures, components, and systems whose failure or malfunction could indirectly result in a condition adversely affecting public health and safety. The failure of a Category B item, in conjunction with the failure an additional item, could result in an unsafe condition.
C	Minor impact on safety	Category C items include structures, components, and systems whose failure or malfunction would not significantly reduce the packaging effectiveness and would not be likely to create a situation adversely affecting public health and safety.

The issues faced in PBR of low hazard facilities are similar to the issues faced by the NRC Nuclear Material Safety and Safeguards (NMSS) office in regulating nuclear materials. The NMSS office is currently studying how RIPBR can be adapted to nuclear materials regulation, e.g., how and which RIPB regulations can be adapted for nuclear materials, and which uses of NMSS regulatory activities are amenable to RIPBR. (See Section 3.6.)

3.0 Current Regulatory Initiatives

3.1 Basic Concepts and Definitions

In this section we will report the most commonly encountered definitions of PBR and RIR and will discuss how they can be combined with the concepts of DID and SM in the search of a coherent regulatory framework.

The NRC Staff defines PBR [NRC White Paper] as "A regulation can be either prescriptive or performance-based. A prescriptive requirement specifies particular features, actions, or programmatic elements to be included in the design or process, as the means for achieving a desired objective. A performance-based requirement relies upon measurable (or calculable) outcomes (i.e., performance results) to be met, provides more flexibility to the licensee as to the means of meeting those outcomes. A PBR approach is one that establishes performance and results as the primary basis for regulatory decision-making, and incorporates the following attributes:

(1) measurable (or calculable) parameters (i.e., direct measurement of the physical parameter of interest or of related parameters that can be used to calculate the parameter of interest) exist to monitor system, including licensee, performance against clearly defined, objective criteria;

(2) licensee have flexibility to determine how to meet the established performance criteria in ways that will encourage and reward improved outcomes; and

(3) a framework exists in which the failure to meet a performance criterion, while undesirable, will not in and of itself constitute or result in an immediate safety concern."

According to the same document, "a risk-informed approach to decision-making represents a philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to health and safety. Risk Information A risk-informed approach enhances the traditional approach by:

(a) allowing explicit consideration of a broader set of potential challenges to safety,

(b) providing a logical means for prioritizing these challenges based on risk significance, operating experience, and/or engineering judgement,

(c) facilitating consideration of a broader set of resources to defend against these challenges,

(d) explicitly identifying and quantifying sources of uncertainty in the analysis, and

(e) leading to better decision-making by providing a means to test the sensitivity of the results to key assumptions.

Where appropriate, a risk-informed regulatory approach can also be used to reduce unnecessary conservatism in deterministic approaches, or can be used to identify areas with insufficient conservatism and provide the bases for additional requirements or regulatory actions" [NRC White Paper, p. 4].

RI and PB regulatory approaches can be used separately. For instance, RI techniques can identify which systems and components should receive most of the regulator's attention but use prescriptive approaches to regulating the maintenance and operation of those components. Similarly, PB approaches can be used without formal RI approaches.

It is worth mentioning how DID fits into RIPBR. According to the NRC Staff, the concept of DID will continue to be a fundamental tenet of its regulatory practice. Risk insights can make the elements of DID clearer by quantifying them to the extent practicable. Although the uncertainties associated with the importance of some elements of DID may be substantial, the fact that these elements and uncertainties have been quantified can aid in determining how much DID makes sense. Furthermore, decisions on the adequacy of or the necessity for elements of DID should reflect risk insights [NRC White Paper, p. 4].

As an alternative to prescriptive regulation, RIPBR uses a new approach for achieving the desired level of nuclear safety performance. It concentrates upon satisfying performance goals rather than upon specific methods. RIPBR uses mutually negotiated performance goals and incentives for judging and rewarding licensee behavior. In the past, the USNRC has used system performance goals in regulation to a limited extent. Important examples include using test-based reliability standards for emergency diesel-generator starting (Brattle and Campbell, 1983), and use of required reactor survival durations in judging the acceptability of systems for withstanding station blackout conditions (Baranowsky, 1985).

RIPBR often, but not exclusively, includes expected risks among the measures of expected safety performance. Analysts estimate these risks using PRAs to evaluate changes in Technical Specifications such as increasing the allowed outage times (AOT) of subsystems or equipment and surveillance test intervals (STI) (i.e., the time between maintenance surveillances). This treatment differs from the existing, prescriptive, regulatory approach, in which regulators are concerned with ensuring that proper hardware, skilled personnel, and comprehensively specified procedures are used in regulated activities. Regulators can apply both the prescriptive and performance-based approaches in all areas of nuclear safety regulation, such as nuclear medicine and nuclear waste disposal.

The effort to introduce RIPBR is progressing. The NRC has stated a commitment to add RIPBR to deterministic analyses, expert judgement, and defense-in-depth to the analytic bases and principles upon which the agency will base future regulatory decisions. See Figure 1.

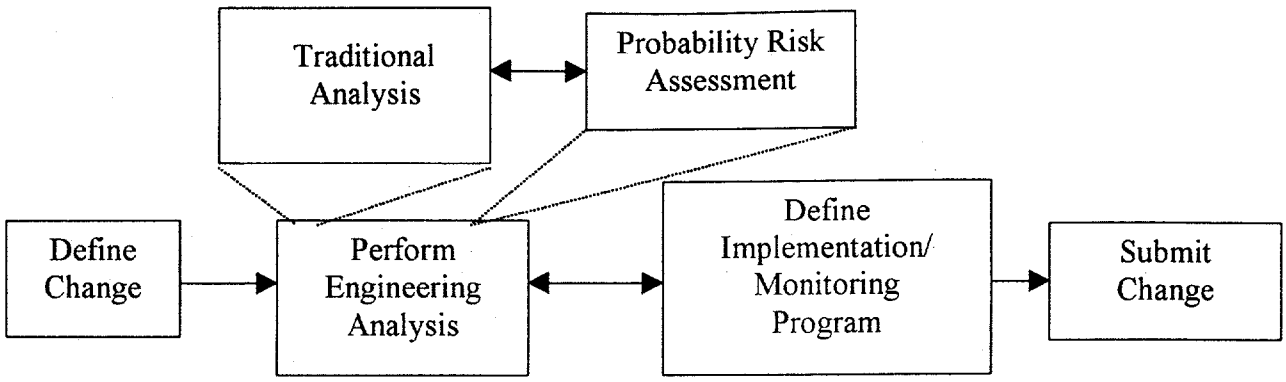

Figure 1: General Description of an Acceptable Approach to Risk-informed Applications (NRC RG 1.174, 1998, p. 7)

3.2 Regulatory Guides

Regulatory Guides are issued to describe and publicize methods acceptable to the NRC staff of implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluating specific problems or postulated accidents, or to provide guidance to applicants. They are not a substitute for regulation, and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if applicants substantiate the findings needed to the issuance or continuance of a permit or license by the commission.[9] In practice, however, compliance with the Regulatory Guides is usually a quicker and less expensive way to gain regulatory approved of a proposed action. Hence, licensees have a strong incentive to comply with Regulatory Guides.

3.2.1 Regulatory Guide 1.174: An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis

NRC Regulatory Guide 1.174 is the first step, and in a certain sense a bridge, towards a risk-informed regulation. As mentioned previously, PRA techniques are not a part of the regulatory system, and the majority of the regulation of nuclear power plants is deterministic. As a result of the Reactor Safety Study (1975), PRA techniques acquired recognition as a valuable tool in dealing with reactor Safety. "...[T]he fault-tree/event-tree methodology is sound, and both can and should be more widely used by the NRC". Since then the PRA methodology has grown in importance while further improving and refining its computational tools. As a consequence of the advances in the methodology, the NRC's policy statement on PRA "encourages greater use of this analysis technique to improve safety decision-making and improve regulatory efficiency" [NRC RG 1.174]. PRA is seen as the most valuable tool to be inserted in the actual regulatory body to reduce unnecessary conservatism, while preserving the DID and SM concepts. RG

[9] The guides are issued in the following ten broad divisions: (1) Power Reactors, (2) Research and Test Reactors, (3) Fuels and Material facilities, (4) Environmental and Siting, (5) Materials and Plant Protection General, (6) Products, (7) Transportation, (8) Occupational health, (9) Antitrust and Financial review, and (10) General.

1.174, for the first time, explicitly suggest using this technique to evaluate the impact of licensing basis changes.

Regulatory Guide 1.174 describes an "acceptable approach for assessing the nature and impact of licensing basis changes by considering engineering issues and applying risk insights" [NRC RG 1.174, p. 4]. The Guide provides the NRC staff's recommendations for using risk information in support of licensee-initiated changes requiring review and approval by the NRC. The acceptance guidelines for the application are based on two metrics: CDF and LERF. The applicant must show that the increase (if any) in each of these two parameters falls under a certain value as specified consistent with the NRC safety goals. In order to demonstrate compliance of the proposed change with these values, the applicant can use a PRA to support its numerical calculation. The licensee risk assessment may be used to address the principle that proposed increases in CDF and risk are small and are consistent with the intent of the NRC's Safety Goal Policy Statement" [NRC RG 1.174].

Figure 2 [RG 1.174] illustrates these concepts. Each block represents one of five principles of acceptance for the design basis change. Notice that the concepts of DID and SM are essential part of the regulation. Block 4 refers in particular to the previous discussion regarding the expected change in the plant CDF. PRA results are used in the decision-making process in two ways. The results address the overall CDF/LERF of the plant and address the change in the CDF/LERF due to the proposed change.

For the results to be considered valuable, a PRA of sufficient quality and detail are required. The quality and level of detail depends on the impact of the proposed change on the plant safety. Obviously major changes will have to be addressed more carefully and will require a higher complexity and completeness of the analysis than minor changes.

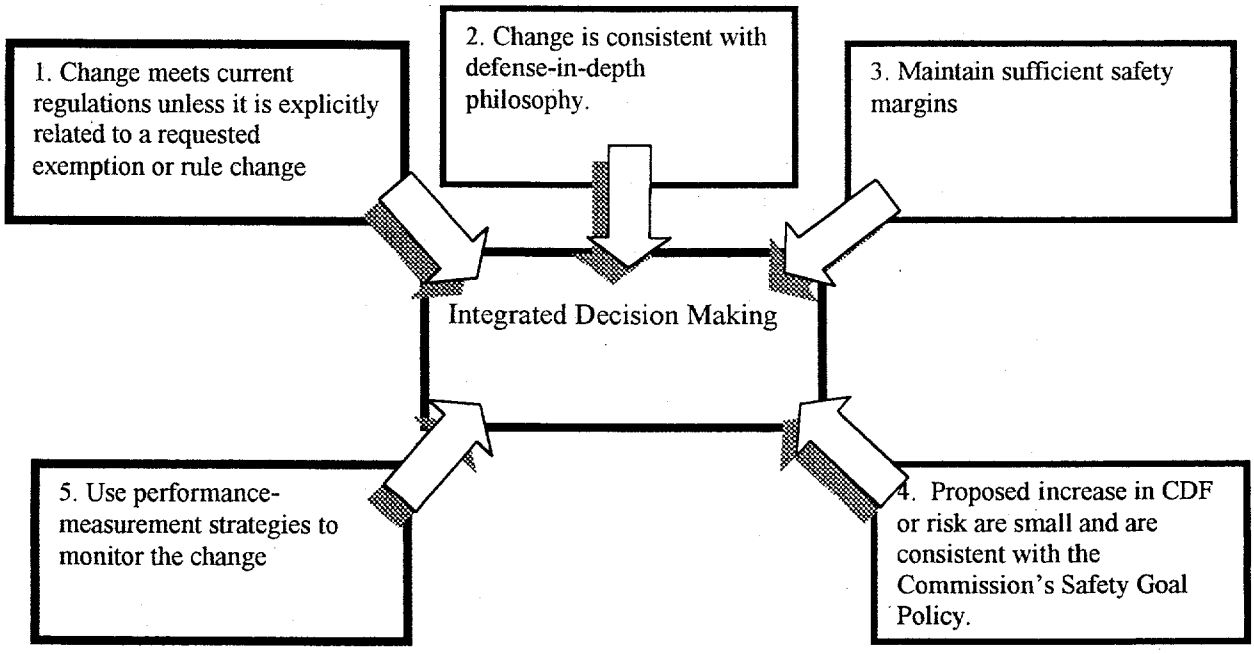

Figure 2: The NRC's Approach to Risk-Informed Decision Making.

PRA "quality" is defined in RG 1.174 to be a "measure of the adequacy of the actual modeling". To assure high quality of its study, one approach a licensee may take is to submit its PRA to peer review. Documentation of the review process, the qualification of the reviewers, the summarized review findings and resolutions to these findings are all to be part of the peer review process. Another approach that RG 1.174 suggests is to adopt an industry-wide PRA certification program.

The NRC staff checks the quality of the PRA based on the following criteria:

1. Use of personnel qualified for the analysis;
2. Use of procedures that ensure control of documentation, including revisions, and provide for independent review, verification, or checking of calculations and information used in the analyses (an independent peer review or certification program can be used as an important element in this process);
3. Provision of documentation and maintenance of records in accordance with the guidelines in Section 3 of the guide;
4. Provision for an independent audit function to verify high quality (an independent peer review or certification program can be used for this purpose);
5. Use of procedures that ensure appropriate attention and corrective actions are taken if assumptions, analyses, or information used in previous decision-making are changed (e.g., licensee voluntary action) or determined to be in error; and
6. Expectation that when performance-monitoring programs are used in the implementation of proposed changes to the LB, those programs will be implemented by using quality assurance provisions commensurate with the safety significance of affected SSCs. An existing PRA or analysis can be utilized to support a proposed LB change, provided it can be shown that the appropriate quality provisions have been met.

Other elements are needed from the licensee to meet NRC requirements. The NRC requires a description of the risk assessment methods used in the analysis, identification of key modeling assumptions that are necessary to support the analysis or that affect the application, the event trees and fault trees necessary to support the analysis, and a list of operator actions modeled in the PRA that affect the application and their error probabilities.

The acceptance guidelines for a licensing basis change are expressed in terms of change in CDF and LERF. The acceptance criteria are illustrated in Figures 3 and 4. The analysis is subject to increased technical review and management attention as indicated by the darkness of the shading of these figures. In the context of integrated decision making, the boundaries between regions should not be interpreted as being definite; the numerical values associated with defining the regions in the figure are to be interpreted as indicative values only.

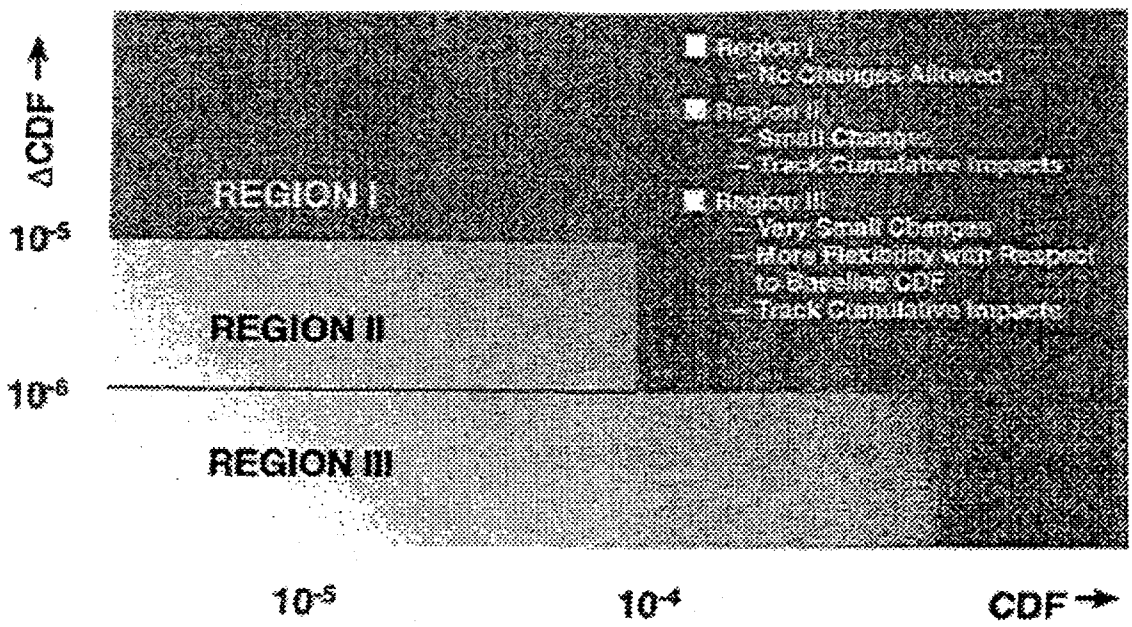

Figure 3: Acceptance guidelines for CDF
(CDF units are in core damage events per year)

The interpretation of these figures is as follows:

- If the application can be shown to result in a decrease in CDF, the change will be considered to have satisfied the relevant principle of risk-informed regulation with respect to CDF.
- When the calculated increase in CDF is very small, which is taken as being less than 10^{-6} per reactor year, the change will be considered regardless of whether there is a calculation of the total CDF (Region III). While there is no requirement to calculate the total CDF, if there is an indication that the CDF may be considerably higher than 10^{-4} per reactor year, the focus should be on finding ways to decrease rather than increase it.
- When the calculated increase in CDF is in the range of 10^{-6} per reactor year to 10^{-5} per reactor year, applications will be considered only if it can be reasonably shown that the total CDF is less than 10^{-4} per reactor year (Region II).
- Applications that result in increases to CDF above 10^{-5} per reactor year (Region I) would not normally be considered.

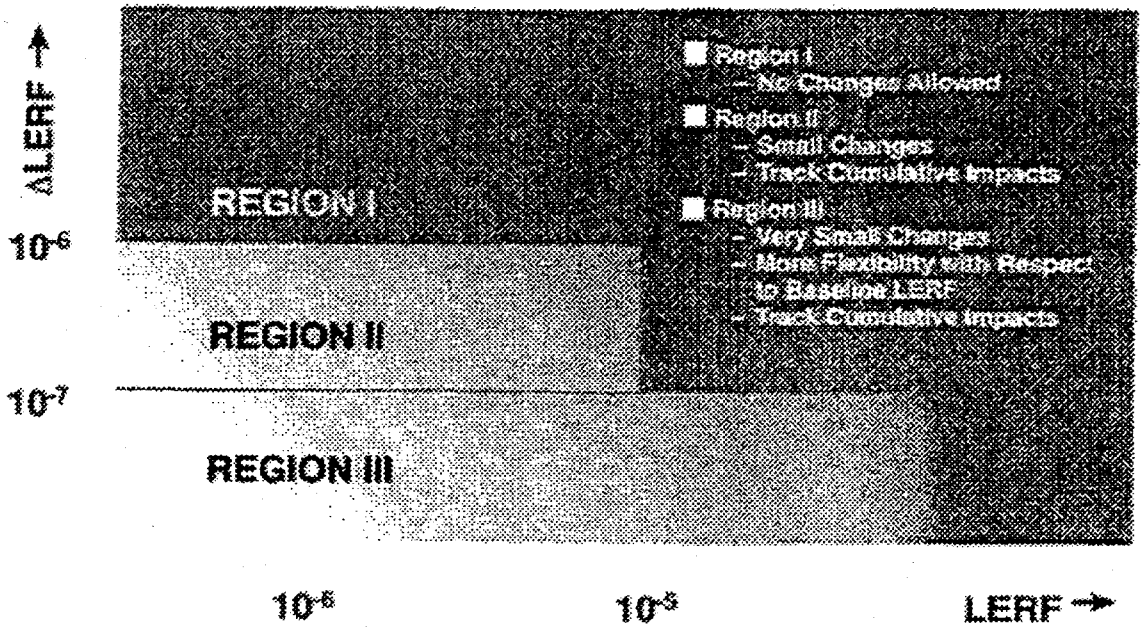

Figure 4: Acceptance Guidelines for LERF

- If the application can be shown to result in a decrease in LERF, the change will be considered to have satisfied the relevant principle of risk-informed regulation with respect to LERF. (Since Figure 4 is drawn with a log scale, this region is not explicitly indicated on the figure.)
- When the calculated increase in LERF is very small, which is taken as being less than 10^{-7} per reactor year, the change will be considered regardless of whether there is a calculation of the total LERF (Region III). While there is no requirement to calculate the total LERF, if there is an indication that the LERF may be considerably higher than 10^{-5} per reactor year, the focus should be on finding ways to decrease rather than increase it.
- When the calculated increase in LERF is in the range of 10^{-7} per reactor year to 10^{-6} per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 10^{-5} per reactor year (Region II).
- Applications that result in increases to LERF above 10^{-6} per reactor year (Region I) would not normally be considered.

In addition to RG 1.174, there are a series of application-specific RGs that provide RI and PB guidance. RG 1.175 addresses the Maintenance Rule, which is discussed below. RG 1.1.76 is concerned with quality assurance practices. RG 1.178 describes an acceptable approach for assessing the nature and impact of proposed permanent technical specification changes.

3.2.2 Uncertainties

Another major change of philosophy found in RG1.174 (and then propagated in RG 1.175, 1.176, 1.177, 1.178) is the explicit consideration of the uncertainties connected with the state of knowledge in the study of reactor safety. Uncertainties are not only mentioned but are qualified and explained. The categorization used consists of the following:

1. Aleatory Uncertainty: which occurs when the events or phenomena being modeled are characterized as occurring in a "random" or "stochastic" manner and probabilistic models [model of the world (MOW)] are adopted to describe their occurrences;
2. Parameter Uncertainty: which reflects the incompleteness of our knowledge of the real value of the MOW parameters;
3. Model Uncertainty: which reflects the incompleteness of our state of knowledge about a certain phenomenon; and
4. Completeness Uncertainty: which is related to the "scope" limitation and refers to those potential risk contributors that we have somehow disregarded in developing the analysis.

In view of the uncertainties related to the PRA methodology, to consider the acceptance guidelines as a pure numerical "table" would be conceptually wrong. Therefore, the approach to these guidelines proposed by the NRC is to complement the numerical results with a full understanding of the contributors in the PRA to these results. Clearly different regions in Figures 3 and 4 require different depths in the analysis. Obviously changes resulting in great differences in CDF or LERF would require a deeper analysis than changes that do not. "The different regions of the acceptance guidelines require different depths of analysis. Changes resulting in a net decrease in the CDF and LERF estimates do not require an assessment of the calculated baseline CDF and LERF. Generally, it should be possible to argue on the basis of an understanding of the contributors and the changes that are being made that the overall impact is indeed a decrease, without the need for a detailed quantitative analysis.... For larger values of CDF and LERF, which lie in the range used to define Region II, an assessment of the baseline CDF and LERF is required". [NRC RG 1.174]. The NRC's approach to require an increase in the control of the analytical methodology and results is consistent with DID and SM concepts.

3.3 Some Commercial Nuclear Power Plant Regulatory Initiatives in the US

3.3.1 NRC's Initiative to Improve the Reactor Regulatory Oversight Process

The NRC's Principles of Good Regulation are: (1) Independence, (2) Openness, (3) Efficiency, (4) Clarity, and (5) Reliability. Although commercial nuclear power plants have operated safely, current NRC regulations have not necessarily satisfied the efficiency, clarity or reliability principles. "Despite [success], the [NRC] has noted that the current inspection, assessment, and enforcement processes (1) are at times not clearly focused on the most safety important issues, (2) consist of redundant actions and outputs, and (3) are overly subjective with NRC action taken in a manner that is at times neither scrutable nor predictable" [NRC SECY-99-007].

The NRC staff has undertaken an initiative, fueled by encouragement and input from industry, to develop a new regulatory oversight framework for commercial reactors that better realizes the NRC's regulatory principles. The staff recommended a RIPB regulatory framework to the Commission after over 6 months' effort in three task groups, focused on (1) Technical Framework, (2) Inspection, and (3) Assessment [NRC SECY-99-007].

The Technical Framework task group first identified and developed "cornerstones of safety". These safety cornerstones chosen are the following [NRC SECY-99-007]:

1. Initiating Events - "Limit the frequency of initiating events";
2. Mitigating Systems - "Ensure the availability, reliability, and capability of mitigating systems";
3. Barrier Integrity - "Ensure the integrity of the fuel cladding, reactor coolant system, and containment boundaries";
4. Emergency Preparedness - "Ensure the adequacy of the emergency preparedness functions";
5. Public Safety - "Protect the public from exposure to radioactive material releases";
6. Occupational Safety - "Protect nuclear plant workers from exposure to radiation";
7. Physical Protection - "Provide assurance that the physical protection system can protect against the design basis threat of radiological sabotage."

Then within each of these cornerstones, the task group identified and developed the following:

1. Objectives, scope, and key attributes of each cornerstone;
2. Areas to be measured to ensure that the cornerstone objectives are met;
3. Performance Indicators (PI's) for each of these areas;
4. Which areas could be monitored sufficiently by the PI's;
5. Inspection and other informational needs to supplement the PI's and verify the validity of the PI data; and
6. PI thresholds to establish "clear demarcation points for identifying fully acceptable, declining, and unacceptable levels of performance" [NRC SECY-99-007].

The task group also identified "cross-cutting issues", aspects of licensee performance that do not belong to one specific cornerstone but are still important to meeting safety goals. Human performance, establishment of a safety conscious work environment, common cause failure, and effectiveness of licensee problem identification and corrective action programs fell in this category.

Table 3 shows some of the key performance indicators proposed by the study. Table 4 shows the conceptual model for evaluating licensee performance indications. This is similar to the NEI's performance bands, and was developed with NEI's input [NEI 1998]. This is clearly a PBR framework utilizing risk information. In addition, the staff's proposed framework and the utilization of inspections is one way to combine the traditional DID concept within PBR. The cornerstones themselves represent the multi-barrier approach, one aspect of DID. In addition, the baseline inspection scheme combines DID with PBR, through the following three types of inspections: (1) complementary inspections in areas that are not measured by performance indicators; (2) supplementary inspections in areas of safety which can not be captured adequately by performance indicators; (3) verification inspections to verify the accuracy and completeness of data used as the basis for PIs [NRC SECY-99-007].

Table 3: Some Performance Indicators Proposed by NRC Staff [NRC SECY-99-007]

Safety Cornerstone	Performance Indicator
Initiating Event	Loss of feedwater frequency
	Loss of ultimate heat sink frequency
	Loss of offsite power frequency
Mitigating Systems	Reliability and availability of turbine-driven pumps
	Reliability and availability of motor-operated valves
	Common cause failure indicator
	Reliability of on-site emergency ac power
Barriers	Reliability and availability of containment spray system trains
	Reliability and availability of containment cooling system trains
	Reliability and availability of containment isolation system trains

Table 4: Conceptual Model for Evaluating Licensee Performance Indicators [NRC SECY-99-007]

GREEN (Acceptable Performance – Licensee Response Band) Cornerstone objectives fully met Nominal Risk/Nominal Deviation From Expected Performance
WHITE (Acceptable Performance – Increased Regulatory Response Band) Cornerstone objectives met with minimal reduction in safety margin Outside bounds of nominal performance Within technical specification limits Changes in performance consistent with changes in CDF less than E-5 Changes in performance consistent with changes in LERF less than E-5
YELLOW (Acceptable Performance – Required Regulatory Response Band) Cornerstone objectives met with significant reduction in safety margin Technical specification limits reached or exceeded Changes in performance consistent with changes in CDF less than E-5 Changes in performance consistent with changes in LERF less than E-5
RED (Unacceptable Performance – Plants not normally permitted to operate within this band) Plant performance significantly outside design basis Loss of confidence in ability of plant to provide assurance of public health and safety with continued operation Unacceptable margin to safety
UNSAFE PERFORMANCE

3.3.2 The Maintenance Rule

The Maintenance Rule is a risk-informed rule that determines which structures, systems, and components (SSCs) that are to be included within the scope of the rule for a particular power reactor, establishes the requirements by the reactor licensee for monitoring the performance or condition of these SSCs, and encourages the licensee to consider the impact on safety when removing SSCs from service for preventive maintenance.[10]

SSCs that are included within the scope of the Maintenance Rule are those that are relied upon to mitigate accidents or transients or are used in emergency operating procedures, whose failure could prevent safety-related SSCs from fulfilling their safety function, and whose failure could cause a reactor scram or an actuation of safety-related system. Licensees must monitor the performance and condition of SSCs within the scope of the rule against licensee-established goals to provide reasonable assurance that these SSCs are capable of fulfilling their intended functions. These performance goals must be commensurate with the SSC risk significance and, when practical, take into account industry-wide operating experience. Furthermore, licensees must take appropriate corrective actions when performance of an SSC within the scope of the rule does not meet established goals. Licensees are allowed to eliminate goal setting and monitoring activities for specific SSCs when the licensee has demonstrated that the performance of those SSCs is effectively controlled through preventive maintenance such that the SSC remains capable of performing its intended function.[11]

At least once during each refueling cycle, but not less frequently than every twenty-four months, licensees must evaluate their performance and condition monitoring activities and associated goals, as well as preventive maintenance activities. Licensees must adjust their programs when necessary to ensure that the objective of preventing failures of SSCs through maintenance is appropriately balanced against the objective of minimizing unavailability of SSCs due to monitoring or preventive maintenance. Finally, licensees should take into account the total of plant equipment that is out of service in order to determine the overall effect on performance and preventive maintenance activities.

The Maintenance Rule has several performance-based elements. Licensees have the flexibility to establish the performance and condition goals and the requisite equipment monitoring regimes, modify established goals on the basis of plant or equipment performance, determine whether to rely on preventive maintenance in lieu of establishing goals and performance or condition monitoring, and allow, for low safety SSCs, plant-level monitoring. The rule has risk-informed aspects as well. It encourages licensees to use assumptions and results associated with PRAs. PRAs can be used to determine which SSCs are within the scope of the rule and what equipment can be removed simultaneously from service.

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[10] The Maintenance Rule was published on July 10, 1991 as Section 50.56 of 10 CFR Part 50. It became effective on July 10, 1996. Supporting documents include NUMARC 93-01 and Regulatory Guide 1.160.

[11] See 10 CFR 50.65, Paragraph (a)(2).

The NRC is reviewing the regulations contained in 10 CFR Part 50 (Part 50).[12] It is considering three major options for a high-level approach for incorporating risk-informed attributes into the Part 50 regulations. These three options are to make no change to Part 50, to make changes to the scope of systems, structures, and components covered by those sections of Part 50 requiring special treatment, or to change specific regulatory requirements. The NRC staff has recommended the second option to the Commission and that the NRC should proceed with a phased implementation strategy with two objectives: First, to develop a risk-informed regulatory framework that will enhance safety; Second, to reduce unnecessary staff and licensee burden.

To advance this process, the NRC staff has recommended to the Commission that licensee conformity with a modified Part 50 should be voluntary rather than mandatory, industry pilot studies with selected exemptions to Part 50 should be utilized as part of the risk-informed development process, the scope of the Maintenance Rule should be changed as an early part of the risk-informed program, and the NRC staff should develop clarification of its authority for applying risk-informed decision making in areas beyond those associated with licensee initiated risk-informed licensing actions. Not only are the regulations regarding existing reactors being reviewed to be made more risk-based, but so are the regulations and the licensing process for new reactors are being reviewed.

3.3.3 The Systematic Assessment of Licensee Performance Program

To assure the conformity of licensee's behavior with the NRC's safety philosophy, the NRC implemented an oversight process known as "Systematic Assessment of Licensee Performance" (SALP). Four plant functional areas are identified: Plant Operations, Maintenance, Engineering and Plant Support. NRC's inspectors evaluated plant safety performance in each of these areas. Procedural adherence, safety related plant equipment maintenance, control room deficiencies, root cause investigation and corrective actions, design activities, worker's sensitivity towards and understanding of radiological controls and alarms were some of the investigated aspects. Performance was ranked in four categories from 4 to 1, where ranking 4 meant a very poor safety performance. As a result the SALP was intended to document" the NRC's observations and insights on a licensee performance" and to "communicate the results to the licensee and the public". It should have provided a vehicle for clear communication with licensee management to focus on plant performance relative to safety risk perspectives. The NRC has utilized SALP results when allocating NRC inspection resources at licensee facilities. [Clinton Power Station, SALP, 1995].

The SALP process has been recently criticized. Redundant actions and outcomes, non-safety-focused inspections, subjectivity were listed among the criticisms. Conceptually, this came from the lack of unanimous agreement on the meaning of "safety-performance".

The issuing in March 1998 of SECY-98-045, "Status of the Integrated Review of the NRC Assessment Process (IRAP) for Operating Commercial Nuclear Power Plants" marked a decisive step in the NRC implementation of a new integrated assessment process. In September

[12] SECY-98-300, December 23, 1998.

1998 the previously used Safety Assessment of Licensee Performance (SALP) process was officially suspended, and it will be definitely terminated in case of success of the pilot program [NRC SECY-99-007].

3.3.4 US Commercial Nuclear Power Industry Initiatives

The US Commercial Nuclear Power Industry has advanced several initiatives to improve current regulations and practices. The focus of these initiatives have been to identify areas "where regulations or regulatory guidance are out of date, where operating experience or improved technology provide a better understanding of a source of risk, and where areas of marginal safety significance can be found that are highly resource intensive [NEI 1998]."

Some of these initiatives include pilot programs to consider changes in allowable equipment outages times, changes to equipment testing intervals, changes to the types, locations and frequency of piping inspections, and reduced quality assurance measures on specific equipment.[13] For example, the nuclear industry, through the Nuclear Energy Institute (NEI), requested that the NRC staff reviews and approves two topical reports that address methods for developing a risk-informed in-service (RI-ISI) program for piping. Lessons learned from pilot plants along with public and staff comments have been used to revise Regulatory Guide 1.178, which provides guidance to reactor licensees on acceptable approaches for developing and implementing a and the Standard Review Plan Section 3.9.8, which provides guidance to the staff on the review of RI-ISI submittals. [NRC SECY-98-139].

In parallel with the NRC staff's development of the IRAP proposal, the Nuclear Energy Institute (NEI) developed an independent proposal for improving the assessment process. The proposed NEI approach conceptually focused on maintaining the barriers to radionuclide release, minimizing events that could challenge the barriers, and ensuring that systems can perform their intended functions. Performance would be measured through reliance on high-level, objective indicators with thresholds set for each indicator to form a utility response band, a regulator response band, and a band of unacceptable performance (performance tiers). In response to the NEI proposal a public 60 days comment period (ended October 6th 1998) was issued by the NRC and after a 4-day public workshop September 28 - October 1, 1998 consensus was reached on the overall philosophy for regulatory oversight. (See section 3.4).

The NEI has also proposed a major initiative in the area of RIPB regulation [NEI 1998] as discussed in Section 3.3.1.

3.4 Swedish Study on a Risk-Based Performance Monitoring System for Commercial Nuclear Power Plants

The Swedish Nuclear Power Inspectorate[14] has undertaken a similar study to move towards RIPB regulation of commercial reactors. The purpose of this study is to develop "methodology for monitoring the safety performance of nuclear power plants... (1) based on probabilistic

[13] Other initiatives include

[14] NRC's equivalent in Sweden

safety assessment (PSA) methodology; (2) identifying the most promising organizational and operational-based safety-related performance indicators, and developing quantitative relationships between values of the performance indicators and changes in PSA inputs (i.e., reliability measures consisting of component failure rates and initiating event frequencies); (3) demonstrating the detailed implementation of the approach and quantitative relationships to a case-study plant; and (4) developing programmatic and decision making guidelines, as well as needed software, for implementing the performance monitoring system at all Swedish NPPs and for making regulatory use of the system" [ERI/SKI 99-401].

Parts 1 and 2 of this study have been completed thus far. The result, after extensive expert elicitation, is a set of 11 key performance indicators, and a final list of five high-worth PI's shown in Table 5. The cross-cutting issues in NRC staff's recommendations for reactor oversight were also identified in the ERI/SKI's study as critical parameters in nuclear power plant (NPP) operation. In the interim, the study concludes that a PBR framework, based on PSA results, is feasible for NPP's [ERI/SKI 99-401].

Table 5: Final List of High-Worth Performance Indicators [ERI/SKI 99-401]

No.	Performance indicator	Mean Worth
1	Annual rate of safety-significant errors (i.e., reportable violations of technical specifications) by plant personnel, contractors, and others.	87
2	Annual rate of maintenance problems (defined as maintenance rework or overdue maintenance)	71
3	Ratio of corrective versus preventative maintenance work requests on safety equipment.	70
4	Annual rate of problems (deviations/failures) with repeated root cause (i.e., a cause previously identified by a vendor, the plant, another plant, the regulator, etc., for a similar plant or group of plants, or for similar components)	80
5	Annual rate of plant changes that are not incorporated into design-basis documents by the time of the next outage following the change.	70

3.5 Performance Assessment of Storage Facilities

Performance Assessments are used for SNF/HLW repositories. A Performance Assessment (PA) is the equivalent of a PRA for a HLW repository. The PA is a quantitative assessment of the long-term behavior of the whole waste disposal system, "with the objective of demonstrating that the chosen system and site are safe" [Savage, 1995]. The nature of the quantitative assessment can vary from country to country; because of the EPA regulatory criteria used in the US, PAs include a probabilistic treatment of aleatory and epistemic uncertainties. PBR regulation of DOE HLW or SNF storage facilities could also use insights from a PA, or existing Hazards Assessments similar to a PA where available.

The form of EPA regulations for the disposal of HLW/SNF and transuranic waste (TRU), presented in 40 CFR 191, leaves a lot of flexibility to the licensee in how to achieve the results. The main guidance, which limits the cumulative release of various radionuclides into the accessible environment over a 10,000 year frame, is the following (40 CFR 191.13), as illustrated in Figure 5:

(a) Disposal systems for spent nuclear fuel or high-level or transuranic radioactive wastes shall be designed to provide a reasonable expectation, based upon performance assessments, that the cumulative releases of radionuclides to the accessible environment for 10,000 years after disposal from all significant processes and events that may affect the disposal system:

(1) Shall have a likelihood of less than one chance in 10 of exceeding the quantities calculated according to Table 1 (appendix A) [designated by R_o, cumulative normalized release limit for the 10,000 year compliance period]; and

(2) Shall have a likelihood of less than one chance in 1,000 of exceeding ten times the quantities calculated according to Table 1 (Appendix A).

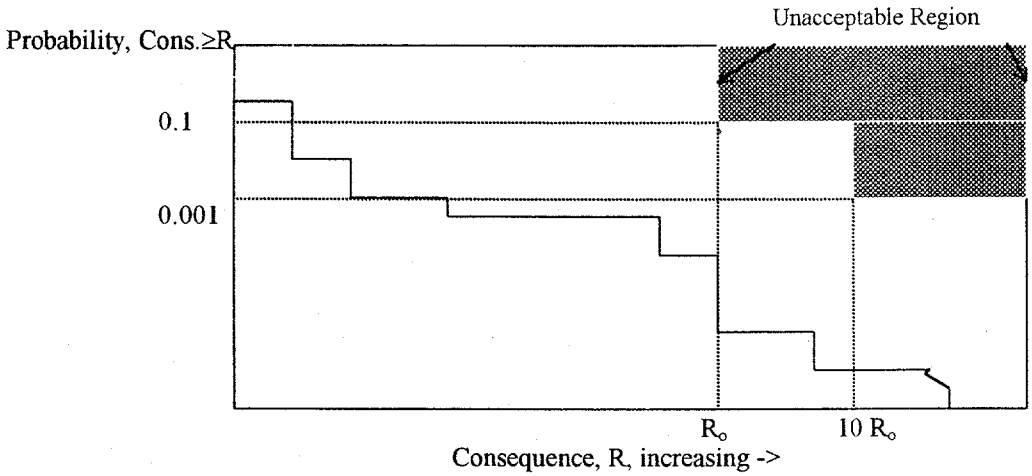

Figure 5: Complementary Consequence Cumulative Distribution Function (CCDF) Curve (where consequence is normalized release of radioactive material, R, to the environment over 10,000 years)

The key words here are "reasonable expectation." They appear again in 40 CFR 191.15 which limits the committed dose to the public: "Disposal systems for waste and any associated radioactive material shall be designed to provide a reasonable expectation that, for 10,000 years after disposal, undisturbed performance of the disposal system shall not cause the annual committed effective dose, received through all potential pathways from the disposal system, to any member of the public in the accessible environment, to exceed 15 millirems." This part goes on to say that assessments need not provide "complete assurance" because of the difficulty of "substantial uncertainties in projecting disposal system performance" over such a long period of time. "Instead what is required in a reasonable expectation" that compliance will be achieved. In addition, the multi-barrier (and heterogeneity of barriers) approach to defense-in-depth is required by Part 191.14: "Disposal systems shall use different types of barriers to isolate the wastes from the accessible environment. Both engineered and natural barriers shall be included."

40 CFR 194 (for the certification of the Waste Isolation Pilot Plant for TRU) then requires that the results of PAs be assembled into a Complementary Cumulative Distribution Function (CCDF) (as in Figure 5) "that represent the probability of exceeding various levels of cumulative release caused by all significant processes and events." These CCDF's represent both aleatory (random) uncertainty, from not knowing which scenarios may occur in the future for example, as well as epistemic (state-of-knowledge) uncertainty, for example from not understanding fully all the thermomechanical-chemical-physical mechanisms that drive the disposal system.

One on-going debate in the PA regulatory community is that of how to account for those uncertainties that can not be quantified. One example of this involves future risks due to events of human intrusion into the repository. Since there is insufficient basis to predict the future state of human society, technological advances, and subsequent resource demands thousands or even hundreds of years into the future, an effective regulatory scheme must include guidance on these scenarios that cannot be handled through the CCDF risk curves. In the case of the Yucca Mountain High Level Waste Repository, the EPA and NRC have proposed (under the National Academy of Science's guidance), to require the repository licensee to analyze one representative human intrusion scenario in order to assess the possible risks from the class of human intrusion events. Whether this is the most effective way to ensure repository safety is still under debate. What is most important for us is to benefit from this debate by gaining perspectives on all sides of the issue, and seeing the possible ways to use a RIPB framework for problems that can not be put explicitly into probabilistic safety analysis. For example, the regulator may require more defense-in-depth, e.g., through multiple barriers, for those cases where there is potential for a large hazard of unknown or uncertain probability.

In the case of facilities in operation, this should be even easier to achieve. For example, large contributors of large potential risk may require more frequent inspections or sampling (e.g., of soil or groundwater underlying a storage site). The safety case of a SNF/HLW repository prepared to remain relatively intact over a 100,000-year history is much harder to demonstrate than the safety case for a facility under active institutional control. So the PA regulatory problem in some sense represents one extreme, a limit, of problems that could be encountered in PBR or RIPBR of DOE facilities.

3.6 Nuclear Materials Safety and Safeguards (NMSS)

In an April 15, 1997 Staff Requirements Memorandum (SRM), the staff was directed to "perform a review of the basis for nuclear materials regulations and processes, and should identify and prioritize those areas that are either now, or could be made, amenable to risk-informed performance-based or risk-informed less prescriptive approaches with minimal staff effort/resources" [NRC SECY-98-138].

In SECY-98-138, the staff made the following preliminary conclusions (which are also applicable to waste storage units and low hazard facilities in general):

PRA may be applicable to only a few specific uses and, for most licensed uses, other system analysis methods that address the three risk questions[15] will need to be considered instead; and

1. Integrating deterministic and probabilistic considerations will likely be a much less important issue, and other issues, such as relating the level of analytic sophistication to the risk associated with specific nuclear materials uses, will likely be much more important in the materials framework [NRC SECY-98-138].

The staff also pointed out the following:

1. Nothing equivalent to the cornerstones for reactors has been issued for nuclear materials;[16]
2. Staff is not aware of any current inadequacies in protecting public health and safety (regarding nuclear materials);
3. Nuclear materials licensees are not anxious for a RIPB regulatory framework (perhaps because they do not have the technical and economic resources to complete the analyses they perceive as necessary under such a regulatory framework); and
4. Experience with system analysis methods will be essential to successful implementation of a RIPB approach to nuclear materials.

With regard to the last point, using system analysis technology, the staff also stated the following:

> ...any increase in the use of system analysis technology must occur within a framework that will ensure that:
> Fundamental regulatory principles are not overlooked in specific applications
> 1. The development of processes and procedures for consistent implementation takes place
> 2. Pilot projects are used for testing of regulatory applications of PRA
> 3. There is an appropriate alignment of level of sophistication of analytic techniques (and their attendant costs and benefits) with risks (real and perceived). The staff also recognizes that any such increase must be accomplished with a commitment of only minimal additional resources [NRC SECY-98-138].

The NRC staff has revised 10 CFR Part 70, Domestic Licensing of Special Nuclear Materials (SNM). The proposed amended rule was released last year, after suggestions from the Nuclear Energy Institute (NEI), and iterative discussions between the NRC staff and the NEI [NRC SECY-97-137]. In the staff's memo to the commissioners, the staff states: "The staff's proposed revisions to Part 70 are intended to provide a risk-informed, performance-based approach for increasing confidence in the margin of safety..." [NRC SECY-98-138]. The major provisions of the revision are:

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[15] The three risk questions are the following: (1) What can happen? (2) How likely is it to happen? and (3) What are the consequences?

[16] The cornerstones for power reactors are initiating events, mitigation systems, barrier integrity, and emergency preparedness. See Section 5 of this report.

1) Performance of a formal ISA (Integrated Safety Assessment), which would form the basis for a licensee's safety program. This requirement would apply to all licensed facilities (except reactors and the gaseous diffusion plants) or activities, subject to NRC regulation, that are authorized to possess SNM in quantities sufficient to constitute a potential for nuclear criticality;
2) Establishment of limits to identify the adverse consequences that licensees must protect against;
3) Inclusion of the safety bases in the license application (i.e., the identification of the potential accidents, the items relied on for safety to prevent or mitigate these accidents, and the measures needed to the continuous availability and reliability of these items). (This is in contrast to the [NEI's] petition's where the ISA results would not be included in the license application.);
4) Ability of licensees, based on the results of an ISA, to make certain changes without NRC prior approval; and
5) Consideration by the Commission, after initial conduct and implementation of the ISA by the licensees, of a qualitative backfitting mechanism to enhance regulatory stability. [NRC SECY-98-185].

4.0 External Regulation of DOE Facilities: DOE/NRC Pilot Plant Interactions

4.1 Introduction

As part of its defense and non-defense missions, the DOE owns and operates approximately 3500 nuclear facilities, involving approximately 34 individual sites across 13 states. These facilities include nuclear research and production reactors, nuclear weapons assembly and disassembly facilities, chemical processing facilities, nuclear material storage vaults, reactor fuel fabrication facilities, tritium recovery facilities, particle accelerators, and research laboratories. A key part of this project is to understand how the DOE currently regulates its facilities and what might be required if the DOE facilities were externally regulated. Results from recent DOE/NRC pilot plant efforts are useful to assess what would be required if DOE facilities were externally regulated. This section summarizes results from this review.

4.2 Current DOE Facility Regulation

Historically, the DOE regulates the design, construction, and operation of its nuclear facilities with statutory authority under the Atomic Energy Act (AEA) to develop and impose requirements to protect the environment and the health and safety of personnel at its facilities. Unlike NRC's authority under the AEA, the DOE self-regulates all radiological, chemical, and physical hazards at its nuclear facilities. The DOE implements this self-regulation through a system of Orders it imposes on DOE contractors through contract provisions. Typically, the Orders are documents prepared by DOE with limited or no public involvement, other than comments received from DOE contractors. This system has gradually developed into an uncoordinated collection of approximately 270 Orders, covering a wide variety of areas, and differing in level of detail, format, and approach. Recently, the DOE has taken steps to reduce or consolidate the Orders. Stimulated by the Price-Anderson Amendments Act of 1988, the DOE has initiated a process to replace the Orders system by one utilizing rules that are promulgated under the public notice and comment requirements of the Administrative Procedures Act. These rules, which will address facility safety, worker health and safety, and environmental protection, will be codified in 10CFR800.

The DOE regulations typically were derived from existing regulations developed by NRC or other agencies. Traditionally, a DOE facility is regulated using selected criteria that exist at the time the facility is built. Because of financial constraints, the DOE typically doesn't make any effort to update facility standards recommended in more-recently-developed criteria.

The DOE's self-regulation is limited to oversight by the Defense Nuclear Facilities Safety Board (DNFSB). The DNFSB is an independent agency that exercises an advisory role with respect to the safety of DOE nuclear defense facilities. The five-member Board was established in 1988 by the Defense Authorization Act. To date, the DNFSB has issued more than 100 formal recommendations to the Secretary. Although all of these recommendations have been accepted by the Secretary of Energy, the DNFSB has no enforcement mechanism and it has not established any new nuclear safety standards during its existence since 1986.

The DOE oversight programs typically contain three components: line management oversight, independent oversight, and enforcement. Line management oversight is provided by the DOE (e.g., Office of Energy Research, Office of Nuclear Energy, Science and Technology, etc.). Typically, this responsibility is assigned to a DOE field office, and DOE headquarters monitors the field office and the contractor's performance. The DOE's Office of Environment, Safety and Health provides independent oversight according to the requirements in DOE's contract with the operating organization for a facility and according to applicable rules (Orders). Formal DOE enforcement is applicable through the Price-Anderson Amendments Act (PAAA) and its implementing regulations. The PAAA of 1988 amended the Atomic Energy Act (AEA) to add Section 234A to provide for a system of civil penalties for contractors who have entered into an agreement of indemnification with the DOE.

Several other organizations have oversight responsibilities for selected DOE facilities. For example, some states have regulatory oversight responsibility for non-radiological air and water quality as well as solid/hazardous waste management activities. This oversight is typically invoked through legislatively mandated state permitting processes. The Environmental Protection Agency (EPA) provides regulations for radiological air quality and toxic substance control.[17]

4.3 Pilot Plant Interactions Exploring External DOE Facility Regulation

In 1995, the DOE created an Advisory Committee on External Regulation (Advisory Committee) to advise and make recommendations on whether and how new and existing DOE facilities and operations might be regulated to better ensure nuclear safety. The Advisory Committee recommended that essentially all aspects of safety at DOE nuclear facilities and sites should be externally regulated and that existing agencies, rather than a new one, should become responsible for such regulation [DOE Advisory Committee], [DOE 1995]. Specifically, the DOE's Advisory Committee recommended that either the NRC or a restructured DNFSB should regulate facility safety at DOE nuclear facilities. In [Hoyle], the NRC endorsed the recommendation that the Commission should have oversight of certain DOE facilities.

To support this effort, it was decided to conduct six DOE facility pilot projects to determine the feasibility of NRC regulatory oversight of DOE nuclear facilities and to support a decision on whether to seek legislation to authorize NRC regulation of DOE nuclear facilities [Hoyle]. The Pilot program tests regulatory concepts at several DOE facilities through simulated regulation by evaluating each pilot facility and its standards, requirements, procedures, practices, and activities against the standards that NRC believes would be appropriate for this type of facility. On November 21, 1997, the DOE Secretary Federico Peña and US NRC Chairman Shirley Jackson signed a DOE/NRC Memorandum of Understanding (MOU) that details the specific conditions and activities associated with the pilot program. The MOU identified eight objectives for the Pilot Program:

1. Determining the value added by NRC regulatory oversight;

[17] Note that in many cases, such as at INEEL, DOE implements and ensures compliance with EPA requirements without actual EPA involvement.

2. Testing various regulatory approaches (e.g., licensing, certification);
3. Determining the status of DOE pilot facilities with respect to meeting existing NRC requirements, or acceptable alternatives, and identifying any significant safety issues;
4. Determining the costs (to the DOE and NRC) of NRC regulation;
5. Evaluating alternative regulatory relations and determining DOE contract changes that might be necessary to provide for NRC oversight;
6. Identifying transition issues and solutions;
7. Identifying legislative and regulatory changes needed; and
8. Evaluating the appropriate process for stakeholder involvement, should the NRC be given broad external regulatory authority over DOE nuclear facilities.

The Lawrence Berkeley National Laboratory (LBNL) pilot program began in the fall of 1997 [NRC SECY-98-080]. On-site work for the LBNL pilot was completed on January 15, 1998, and the site report was issued in Spring 1998. No significant safety issues were observed at LBNL. The second pilot plant was the Radiochemical Engineering Development Center (REDC), at the Oak Ridge National Laboratory (ORNL). Fieldwork for this program was completed in June 1998. A draft report summarizing conclusions from this project indicate that the REDC is licensable without significant changes to the facilities or to their radiation safety programs. The third pilot was the Receiving Basin for Offsite Fuel (RBOF) at the Savannah River Site in South Carolina. Results from this project indicate that the DOE and its contractor, Westinghouse, were controlling risks to acceptable levels and that the facility, as it currently exists, is amenable to NRC regulation. The Pacific Northwest National Laboratory (PNNL) in Washington was selected as the fourth pilot plant. However, this pilot project, which was scheduled to start in the fall of 1998, was postponed in order to involve the participation of state agencies as well as the Occupational Safety and Health Administration [Weapons Complex Monitor, Feb. 1999]. The remaining two pilot plants were never announced (although NRC Chairman Jackson indicated that a non-power reactor and an Environmental Management facility were planned to address concerns that more complex facilities should be considered).

Although the Congress allocated $1 million to the NRC to continue the pilot plant interactions in FY99, this program was delayed indefinitely because of the DOE Secretary Richardson's decision to put external regulation plans and additional studies on hold [Inside NRC] and [Weapons Complex Monitor, Mar. 1999]. Senate Armed Services Committee Chair John Warner, R-VA, stated at a March 15, 1999 hearing on the DOE's FY2000 budget request that he endorsed Energy Secretary Richardson's decision. However, it is not clear if this decision is supported by the House of Representatives or if this decision will be reconsidered in future years.

4.4 Insights gained from Pilot Plant Interactions

A team consisting of DOE, NRC, facility/site operating contractors, and other regulatory bodies (e.g., state government) conducted the pilot plant studies. To compare how the current regulations for a facility/site compared with NRC requirements, the pilot plant studies considered employee training, facility/site organization, procedures, waste management and treatment, emergency preparedness, environmental and personnel monitoring, decommissioning plans, radioactive materials control, and current oversight procedures. Studies focussed on issues

unique to the facility/site and issues applicable to all pilot plants. Issues were addressed through facility/site visits, independent review of facility information, and team discussions. A stakeholder requirements elicitation process was used to identify and address local issues and obtain information for a preferred stakeholder involvement model under potential NRC regulation.

Rather than trying to assess what was required to meet an applicable NRC regulation, the review team often assessed whether the current process was "comparable" or what additional measures were necessary to obtain a level of safety comparable with NRC requirements.

4.4.1 Key Criteria and Principles

The following examples provide insights into key criteria and principles that NRC considers necessary for safe facility operation and possible methods for demonstrating compliance [Predecisonal Draft Document]:

1. In order to avoid unnecessary costs associated with the label of "special nuclear materials," the ORNL REDC review proposed to simply apply the requirements noted in 10CFR Parts 70, 73, and 74 without any materials designation. 10CFR73 requirements for physical protection of nuclear material would be required for materials at fixed sites and during transportation.

2. NRC expects employee training to be commensurate with instructions outlined in 10 CFR 19.12 so that employees can conduct activities in a safe manner consistent with ALARA principals.

3. Criteria must be comparable to 10 CFR 20 personnel radiation requirements (dose limits, personnel monitoring, and posting). It was noted in the REDC review that some analytical techniques, such as use of dose weighting factors for different body regions, differ and would require NRC review and approval.

4. 10CFR20 requirements for waste characterization, treatment, and disposal must be met.

5. Environmental monitoring programs must meet NRC program requirements to assure that no individual in the public receives a dose in excess of 10 CFR 20 dose limits.

6. Safety systems must consider principles such as defense-in-depth and adequate margin of safety. These principles were explicitly noted in the ORNL REDC review of criticality safety.

7. 10CFR30 decommissioning requirements would be applicable to DOE facilities. Specifically, financial assurance for decommission must be provided, which requires a decommissioning funding plan, a cost estimate, and a description of the method for assuring funds for decommissioning. In addition, 10CFR 30.35 requires that records important for decommissioning be maintained. Such records include records of spills, as-built drawings

and modifications of structures and equipment where radioactive materials are used or stored, and locations of possible inaccessible contamination.

4.4.2 Recurring Issues

Several key issues surfaced during the pilot plant interactions. Although the INEEL/MIT program will not attempt to address these issues, they are summarized in this document to provide perspective.

Who should be the regulator?

The DOE Advisory Committee considered this issue with support from the White House Council on Environmental Quality. Their evaluation considered a range of stakeholders (public, federal, state, tribal, industrial, union and academic sectors). The Advisory Committee held eight 2-day public meetings at major DOE sites around the country. In their final report, they recommended that DOE nuclear safety should be externally regulated, and that the external regulator should be either NRC or DNFSB. After a DOE working group reviewed the Advisory Committee report, the DOE accepted the Advisory Committee's recommendations and initiated a process for phasing in NRC regulation within a 10 year period (and phasing out DNFSB). In March 1997, NRC issued a Staff Requirements Memorandum favoring NRC oversight.

Advantages of NRC regulation include:

1. Uniformity in requirements and regulatory programs across the DOE complex.; and
2. Most public comments favored NRC oversight.

Disadvantages of NRC oversight include:

1. Most of the local community (including facility employees) did not favor NRC oversight;
2. Conflict of Interest issues may arise because the NRC uses the DOE laboratory personnel and facilities (hot cells, reactors, etc.) to carry out its research. However, this is likely to become less important as the NRC research program shrinks.

Who should be the licensee?

The DOE owns the facilities, materials, and land on which facilities reside. Typically, the DOE contracts the management and operation of its facilities to organizations, which may be replaced when the DOE contract expires or it is terminated. The contracting organization is responsible for many "typical" licensee decisions, such as when to shutdown a facility for repairs, when to start up a plant, and spending level estimates. However, the DOE must provide the licensee funding required to operate and maintain the facility within regulatory requirements.

Typically, NRC licenses the entity that owns the facilities and materials and holds licensees responsible for all licensed activities, even if some activities are carried out by contractors. However, on-going changes associated with deregulation led NRC to develop criteria for licensing non-owner operators for 10CFR50 licenses for power reactors.

In the ORNL REDC pilot plant interactions, various stakeholders were asked to consider various licensee options: DOE-only license, dual license between the DOE and its operating contractor and contractor-only license. Most stakeholders (NRC, DOE, contractor, and state representatives) preferred the contractor-only license. Reasons cited for this preference include

1. The contractor, who is involved in the daily operations of a DOE facility is best suited to implement nuclear safety standards;
2. A DOE-only license and a dual license would result in duplication of staff (the DOE and the contractor would retain staffs with technical and nuclear safety expertise so that the DOE could find and report potential violations and discrepancies before the NRC found them);
3. A dual (the DOE and contractor) license would complicate compliance and accountability issues.

However, several issues require resolution before contractors could become licensees [Lockheed Martin]:

1. The DOE must be able to define a process for determining that future contractors are qualified to maintain an NRC license;
2. A process for transferring a license to another contractor must be defined. ;
3. The DOE must modify existing maintenance and operation contracts if a facility or site becomes externally regulated by NRC;
4. The DOE must provide firm funding arrangements and commitments to ensure that contractors are guaranteed adequate funding to meeting costs associated with the licensing process, license fees, and compliance obligations;
5. Indemnification of DOE contractors under the PAAA must be continued (or the potential for unlimited liability will deter responsible private companies from competing to construct or operate DOE nuclear facilities); and
6. DOE contractors must not be held financially responsible for decommissioning and decontamination of facilities that they operate on behalf of the DOE.

5.0 Lessons Learned and Future Research Directions

Our review of DOE and NRC regulation has identified several important issues that will assist us in applying PBR to DOE facilities. In the case of the NRC, the principles of requiring DID and maintaining SM have been the basis for the treatment of uncertainties by the current regulatory system. This conservative approach has ensured public health and safety, but it (and the NRC's operational practices) also has caused undue regulatory burdens. A major pitfall in the current framework is that qualitative evaluation of risk does not permit an effective allocation of resources because, prior to PRAs, uncertainties were not quantified. Traditional engineering analysis integrated with PRA, however, has revealed a potentially successful means of addressing and quantifying uncertainties, although many issues of practical implementation remain.

In moving to a RIPBR regulatory structure, two conflicting conceptual concerns have been raised:

- Requirements for DID could be undermined by the introduction of risk-informed regulation; and
- The benefits of the risk-informed regulation could become restricted by the DID philosophy.

The critical question is how to make use of the information available from PRA studies without undermining the DID and SM concepts.

To address this question, two models have been proposed: the "structuralist" and "rationalist". The structuralist model, which has been the historical approach to nuclear regulation, asserts that "defense in depth is embodied in the structure of the regulations and in the design of the facilities built to comply with those regulations" [Sorenson, *et al*]. No matter what the probability is that containment or emergency planning will be required, both have to be provided. DID is primary and PRA is one of the tools employed to assure that DID has been achieved.

The rationalist model establishes quantitative acceptance criteria, such as health objectives, CDF, and LERF as its first step. The second step is to evaluate the uncertainties in the analysis and determine which steps should be taken to compensate for these uncertainties. The role of DID in this model is to "increase the degree of confidence in the PRA results" [Sorenson, *et al*] in supporting the conclusion that adequate safety has been achieved. The fundamental difference between the structuralist and the rationalist models, therefore, is that the structural model accepts defense in depth as the fundamental value, whereas the rationalist model would place DID in a subsidiary role.

The question is how to find a solution to these apparently conflicting ways of looking at the problem of safety. These two models are not generally in conflict, and neither of them provides a perfect answer to this problem. Recently [Sorensen, *et al*], proposed two options. The first option recommends DID as a supplement to risk analysis, which is very similar to the rationalist model. The second option is to combine a high-level structuralist view with a low-level rationalist view. This second approach is more compatible with the current regulatory

structure, although the first option would offer a stronger theoretical foundation for risk-informed regulation [Sorenson, *et al*].

In the combined approach, quantitative goals would be set at a low level or the "Cornerstone Level". This could include goals on initiating event frequencies, safety function or safety systems unavailability [Sorenson, *et al*]. The rationalist approach is concerned with lower levels than the cornerstones as illustrated in Figure 6. Notice that uncertainties increase moving from the left-hand side of this figure to the right-hand side. DID in this approach plays a role in two ways. For events or processes modeled by the PRA, it would be part of the treatment of uncertainties to assure quality of the analysis. In practice in this first role, DID will become part of the overall safety analysis. For systems or events not modeled in the PRA, the structuralist approach would be used to maintain the traditional DID concept. This high-level structuralist and low-level rationalist model can be considered as a pragmatic way of integrating DID and risk-informed approaches.

On a more practical note, although many initiatives have been taken to apply PBR and RI regulation to commercial power reactors, numerous important issues of practical implementation remain. The level of effort, data, and regulatory change that is necessary to move towards PBR and RI regulation is significant. It has only been very recent that proposed PB measures for commercial nuclear power plants have been proposed. Further work in evaluating these measures is necessary including accumulation of data. Incorporating human judgement formally into risk models is another major area that needs additional work. Moreover, the DOE facilities are substantially different from commercial power reactors. Even within certain types of DOE facilities, the range of differences is tremendous. For example, DOE waste storage facilities encompass a very wide set of conditions.

The DOE pilot plant interactions have provided several key insights about DOE facility regulation. Results from that effort suggest that our performance-based regulatory framework should encompass a broad range of issues (waste management and treatment, emergency preparedness, environmental and personnel monitoring, radioactive materials control, etc.). In addition, key NRC regulations (10CFR Parts 19, 20, 30, 70, 73, 74) and principles (ALARA, DID, and SM) referenced in the pilot plant interactions will be considered in our project. Finally, our program should select facilities that will eliminate one criticism of the pilot plant interactions. Specifically, the INEEL/MIT program should consider a nuclear reactor and an environmental management facility to ensure that more complex issues will be addressed.

When considering the merit of a new PBR framework, we must consider the cost of transition to and implementation of the new framework as well. The benefits of the new framework would have to outweigh these costs in order to be lucrative for the regulated entity (the DOE).

In formulating our PBR framework for well-characterized complex DOE facilities (e.g., the ATR or ISFSI), we can draw from the substantial progress in PBR studies for commercial nuclear power plants and long-term HLW/SNF repositories. There is less guidance available for PBR of less complex, lower potential hazard facilities, for which risk assessments may exist in a less formal format. In addition, it will be challenging to formulate a PBR framework for those

inherited DOE facilities which keep/kept few operational records, and may contain poorly characterized materials (e.g., the Hanford HLW tanks).

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Attachment B

Case Studies to Investigate Performance-Based Regulation at DOE Facilities (Project Task 2)

Idaho National Engineering and Environmental Laboratory

INEEL/EXT-99-000535

JUNE 1999

Case Studies to Investigate Performance-Based Regulation at DOE Facilities (Project Task 2)

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Contents

Figures

Tables

1 INTRODUCTION

Performance-based regulation (PBR) focuses on results as primary objectives of regulatory oversight. PBR can potentially reduce operating costs by avoiding unnecessarily burdensome requirements and preventing needless interruptions in production and/or processing. The Idaho National Engineering and Environmental Laboratory (INEEL) and the Massachusetts Institute of Technology (MIT) have initiated a three-year research program aimed at developing a framework for selecting and applying performance-based goals for licensing and regulating nuclear facilities. Project tasks utilize innovative approaches and develop software tools to define goals and a systematic process for demonstrating compliance with the proposed goals. Ultimately, the project seeks to demonstrate that this regulatory approach can be economically and efficiently applied using three U.S. Department of Energy (DOE) facilities as case studies. Many of the methods developed from this research could also be applicable to the commercial power industry and industries that handle or manufacture hazardous materials.

This letter report documents preliminary results from an initial task in the INEEL/MIT project, "Selection of DOE Facilities for Case Studies." In this task, DOE facilities were reviewed to identify key facility groups. From selected groups, representative facilities were chosen for case studies. This report describes these representative facilities, emphasizing aspects of interest to regulation. Criteria under which case study facilities are currently regulated are also discussed.

2 METHODOLOGY

2.1 Background

DOE owns and operates approximately 34 individual sites across 13 states. Facilities located at these sites include nuclear research and production reactors, nuclear weapons assembly and disassembly facilities, chemical processing facilities, nuclear material storage vaults, reactor fuel fabrication facilities, tritium recovery facilities, particle accelerators, and research laboratories. DOE currently regulates all radiological, chemical and physical hazards at its nuclear facilities. In December 1995, however, the Advisory Committee on External Regulation of DOE Safety recommended that all aspects of safety at DOE nuclear facilities and sites should be externally regulated.[1] DOE and the Nuclear Regulatory Commission (NRC) explored the impact of NRC regulating DOE facilities using several pilot projects. However, initial results led to DOE's recommendation to put external regulation plans on hold.[2]

DOE/NRC pilot plant interactions suggested that a prescriptive approach, such as the one currently used by NRC, could significantly increase regulatory costs for DOE facilities.[3] However, development and application of a less prescriptive, more performance-based system of regulation holds substantial promise to reduce licensee burdens and preserve safety. This INEEL/MIT program investigates the potential to utilize recent advances toward PBR to develop a framework for selecting appropriate goals that could be systematically applied to DOE facilities.

Traditionally, most NRC requirements have been prescriptive, providing detailed processes, requirements, or instructions for the licensee to follow. Performance-based requirements, on the other hand, describe the general processes to be followed and the results expected by licensees. This approach gives licensees greater flexibility in developing and adjusting implementation activities to most efficiently utilize and/or merge with their existing programs and policies. Desirable characteristics of a PBR system include: measurable parameters, objective criteria for facility performance monitoring, and flexibility to determine the methods for meeting performance goals. Furthermore, the PBR system should be structured so that failure to meet performance goals does not result in unacceptable conditions. NRC has instituted several examples of PBR, and industry feedback about these regulations has been positive.[4]

2.2 Approach

The proposed framework must be sufficiently broad-based that it is applicable to most DOE facilities. To demonstrate that this framework can be applied to a wide range of DOE facilities, three DOE facilities were selected as case studies for this project. Prior to selecting these case studies, we reviewed various types of DOE facility designs, developed key facility groups, and identified regulatory issues associated with each group. To complete this task, we developed a database describing facilities at various DOE sites. This database lists various facilities, key design features, and facility points-of-contact. We selected representative facilities and contacted appropriate officials to request their participation in this program (volunteering facilities must provide staff to interact with INEEL and MIT). In selecting a representative facility, factors such

as its type, its location, the availability of information that could be used to define performance criteria, and staff willingness to participate, were considered.

This document identifies key DOE facility groups and representative facilities selected for case studies. It describes representative facilities, emphasizing aspects of interest to regulation. The availability of information, such as risk studies, tests, and design data, that could be used to develop PBR is identified for case studies. Furthermore, the criteria under which case studies are currently regulated are discussed. External factors, such as remote siting or co-located facilities that should also be considered in developing regulatory criteria, are identified. Licensing issues of interest, such as a forced shutdown or unreviewed safety questions, are noted.

3 TASK STATUS

To develop a broad-based framework that it is applicable to most DOE nuclear facilities, we reviewed various types of DOE facilities, developed key DOE facility groups, determined a prioritization of facility groups for evaluation as case studies, and selected representative facilities as case studies from three higher-priority DOE facility groups.

To assist us in our review of DOE facilities, we used ACCESS software to create the NUclear FACility (NUFAC) searchable database into which information about various nuclear facilities could be entered. This database allows comparisons between the facilities selected as case studies and other facilities in each group. Figure 1 contains a window from this newly-developed software tool for a representative reactor, the Advanced Test Reactor (ATR). As indicated in this screen, the database includes tabs that provide additional screens containing key facility information, such as design features, current and potential missions, co-located facilities, licensing status, regulatory issues affecting operation, point-of-contacts, and pertinent licensing documentation. The DOE Office of Nuclear Energy, Science, and Technology (DOE-NE) recently decided to refine this software tool and launch it as an internet application. DOE-NE plans to use this tool to identify facilities required to support DOE missions for the next 20 years.[5]

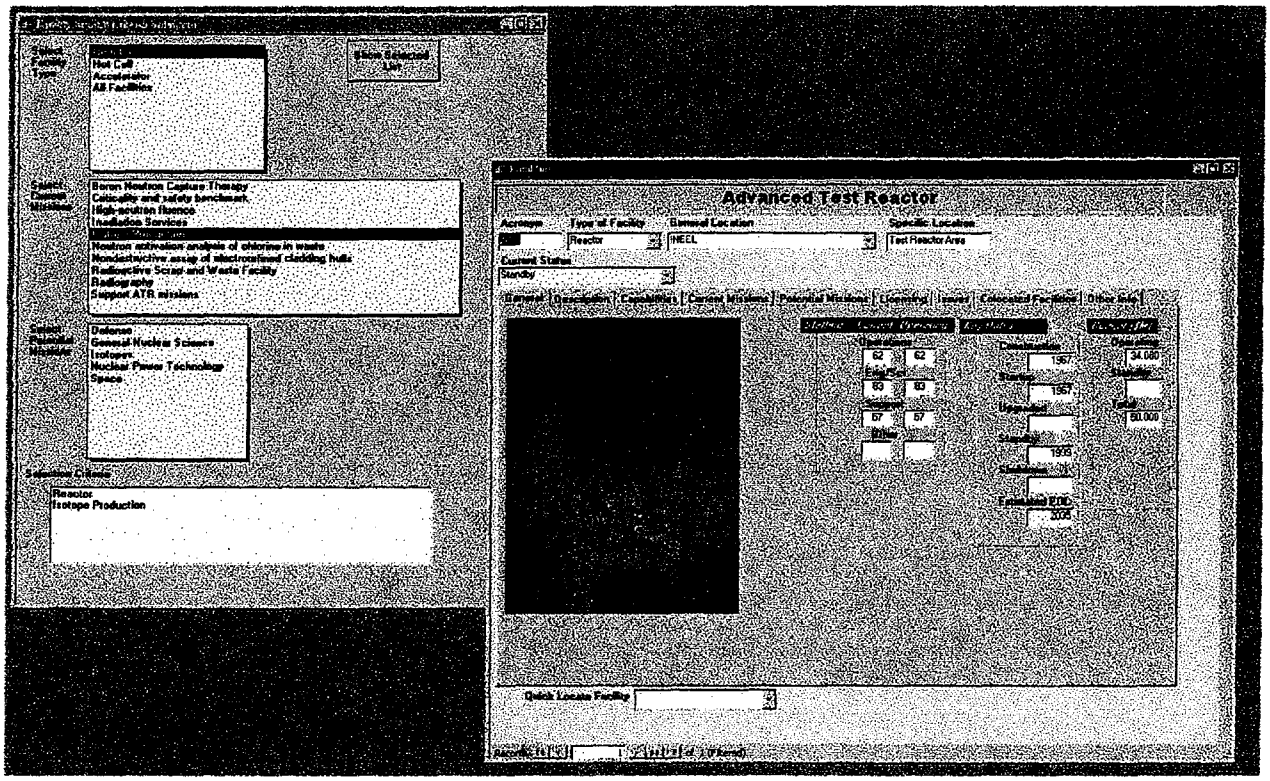


Figure 1. ATR screen in pilot version of NUFAC database.

Review results emphasize the uniqueness of facilities in the DOE complex. For example, currently operating DOE reactors differ in their fuel composition, moderator, and coolant. This

Review results emphasize the uniqueness of facilities in the DOE complex. For example, currently operating DOE reactors differ in their fuel composition, moderator, and coolant. This uniqueness requires that the proposed framework be sufficiently general to encompass most various facility groups and most designs within each group.

To cover the diversity of DOE facilities, we developed the seven key DOE facility groups listed in Table 1. Boundaries between each group may not always be clearly defined. However, we believe that each group has sufficiently different characteristics and licensing issues to warrant separate evaluation. Facility groups are listed in Table 1 according to their priority for evaluation. The reactor group was assigned the highest priority for consideration because of our desire to include examples that are fairly complex, whose results could be directly transferred to the commercial power industry. The hot cell group is the next highest ranked group because results from this example can be compared with results from the recent DOE-NRC pilot plant evaluation of the ORNL Radiochemical Engineering Development Center Hot Cells. The third group that will be considered is a Waste Storage Site. This particular group was considered at this time because of an INEEL facility that was recently constructed and is subject to both NRC and DOE regulation. It is anticipated that evaluation of this facility will yield useful insights about contrasts and similarities in requirements for the two agencies.

Table 1. Proposed DOE facility groups.

Group	Explanation
Reactors	Wide range of unique reactors and critical facilities used for code validation and irradiation services (fuel testing, materials irradiation, isotope production, etc.).
Hot Cells	Structures housing one or more hot cells (heavily shielded enclosures designed to support the remote handling of radioactive materials). Currently support wide range of missions, including fuel characterization and testing, materials evaluation and testing, fuel development, waste characterization and testing.
Waste Storage Sites	Facilities designed to house, monitor, and retrieve radioactive waste.
Nuclear Materials End-Use Facilities	Facilities utilizing low-level radioactive materials for a range of applications (e.g., radiopharmaceuticals, and advanced medical imaging technologies including positron emission tomography, single photon emission computed tomography, and nuclear magnetic resonance imaging).
Contaminated Sites	Buildings and grounds where radioactive materials were previously utilized and/or where radioactive materials are temporarily housed.
Manufacturing/ Process Facilities	Facilities which handle, examine, process, and perform R&D on radioactive materials (including spent fuel).
Accelerators	Systems capable of producing higher energy particles (electrons, protons, neutrons, positrons, heavy ions, etc.) for isotope production, materials testing and irradiation, etc.

Case studies for the three highest priority facility groups (reactors, hot cells, and waste storage sites) were selected based on location, availability of information that could be used to define performance criteria, and organizational support. Because of financial constraints, we limited this study to INEEL facilities. In addition, the selection procedure required that the organization operating each facility would be willing to provide staff to interact with INEEL and MIT. Within these constraints, the following facilities were selected as test cases:

- **Advanced Test Reactor (Reactor Group)** – This 250 MWt light water moderated and cooled reactor is located at the Test Reactor Area (TRA) of INEEL. It performs a range of irradiation services for government, industry, and foreign organizations. It is co-located with hot cells, analytical laboratories, a waste storage canal, and a critical facility.

- **Hot Fuel Examination Facility (Hot Cell Group)** – This facility includes an air-filled decontamination cell and an argon-filled main cell. Located at INEEL's Argonne National Laboratory – West (ANL-W) site, it provides services to DOE fuels and waste programs. It is co-located with the Neutron RADiography (NRAD) reactor.

- **Three Mile Island Independent Fuel Storage Installation (Waste Storage Facility Group)** – This recently-commissioned facility provides horizontal dry storage for the Three Mile Island Unit 2 (TMI-2) core debris. Located at INEEL's Idaho Nuclear Technology and Engineering Center (INTEC), the facility is subject to DOE and NRC regulations through an agreement between the State of Idaho, the Navy, and DOE.[6]

Appropriate documents, such as Final Safety Analysis Reports (FSARs), Technical Safety Requirements (TSRs), Probabilistic Safety Assessments, events reported in the DOE Occurrence Reports and Performance System (ORPS) and the Nonconformance Tracking System (NTS), and the existing DOE Rules, Orders, and Regulations for each facility are being reviewed to assist in selecting performance goals, methods for demonstrating compliance, and appropriate actions when goals aren't met. Suitability of the proposed performance goals will be discussed with appropriate stakeholders (staff, management, DOE, and experts) in subsequent project tasks. Preliminary results from this review are summarized in Sections 4 through 6.

4 REACTOR CASE STUDY: INEEL ADVANCED TEST REACTOR

As noted above, there was a desire to include a reactor facility in the INEEL/MIT study. For our project, we selected the INEEL Advanced Test Reactor (ATR) for evaluation. Several factors motivated ATR's selection as a reactor case study. First, the ATR is the largest operating reactor in the DOE complex. Hence, selection of this facility should avoid criticisms received by the DOE/NRC pilot program that complex facilities weren't considered. Second, INEEL and DOE facility management agreed to participate in this evaluation. Although the INEEL/MIT will minimize ATR staff efforts, this program requires that ATR staff be available to respond to questions and supply facility design and analysis information. Third, the ATR's INEEL location facilitates project interactions. Fourth, the ATR has several reports that should assist in defining performance criteria. This section discusses ATR design features, missions, current regulation, and facility-specific regulatory issues that will need to be considered by this program. This section also lists available ATR documentation that will assist this project.

4.1 ATR Design Features

The ATR, DOE's largest test reactor, is located in INEEL's TRA Building 670. ATR operates at steady power up to 250 MW, but occasionally performs transient testing by rapidly inserting or cycling a test in the core. ATR's primary coolant is demineralized light water operating at subcooled conditions (125-160 °F, 255-355 psi).

ATR utilizes a serpentine arrangement of forty fuel elements, each with 19 fuel plates made from an intermetallic compound of uranium (93% U-235) and aluminum. When viewed from above, the fuel configuration looks like a four-leaf clover with the fuel elements winding in and around a three-by-three array of irradiation positions (see Figure 2). These nine flux traps have significant power differences across the core. Five of the flux traps currently contain pressurized loops, and the beryllium moderator contains dozens of other test locations for smaller irradiation experiments.

ATR systems are upgraded regularly to maintain ATR in top condition. Recently completed upgrades include new reactor and process control systems, cooling tower and underground piping replacement, liquid radioactive waste system upgrade, and completely new Upgraded Final Safety Analysis Report (UFSAR)[7] and Technical Safety Requirements (TSR)[8] documents. Core internals are replaced every seven to nine years. ATR's stainless steel reactor vessel prevents aging problems.

4.2 ATR Missions

The primary mission of the ATR is to provide irradiation services to DOE's Office of Naval Reactors. In addition, irradiation services are provided to other government agencies, commercial sponsors, and international programs. Isotopes, primarily Ir-192 and Co-60, are produced under a commercial contract with International Isotopes of Idaho, Inc.

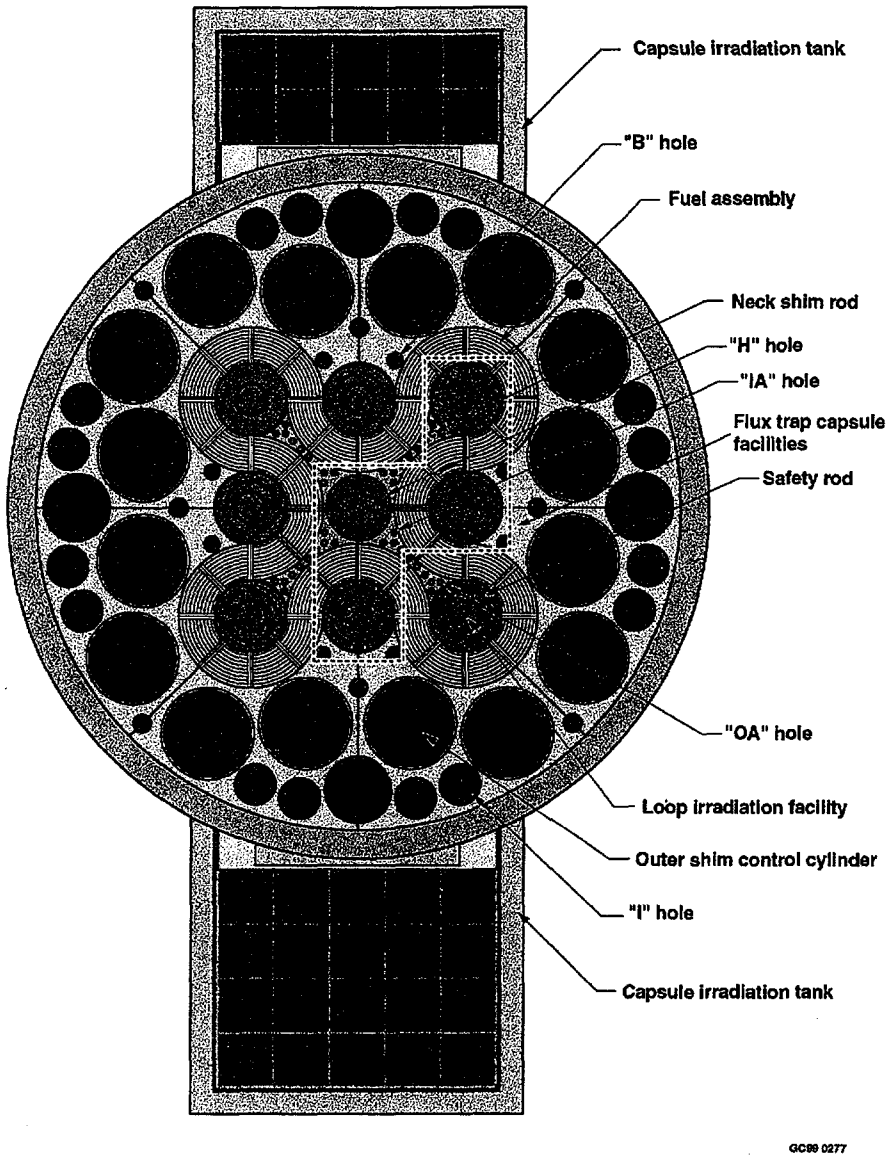


Figure 2. ATR core diagram.

Typically, ATR performs three types of experiments: pressurized water loop experiments; instrumented capsule experiments; and non-instrumented capsule experiments. The active core length is four feet. Diameters of irradiation positions range from under one inch to five inches. The unique control drum arrangement of ATR provides constant axial flux profiles for experiments.

4.3 Current ATR Regulatory Process

Per statutory authority under the Atomic Energy Act (AEA), the DOE regulates all radiological, chemical, and physical hazards at ATR. Chapter 19 of the ATR UFSAR[7] lists applicable statutes, rules, regulations and DOE orders that are binding upon ATR's safety basis and operation. This list includes federal and state statutes, ordinances, and other requirements that establish ATR

safety constraints. DOE Orders span a range of ATR activities, such as emergency preparedness programs, personnel safety, accident analyses, fuel system design, initial testing, in-service inspection, testing and maintenance, environmental design, radioactive and hazardous material management, organizational structure, and the design of structures, components, equipment, and systems. This list indicates that several DOE Orders are applicable to the same activity. Several Code of Federal Regulations (10CFR 100, 1910, 1926, 61.55, 129, 141, etc.), an EPA regulation (EPA 56 FR 1943 on site characteristics), and several State of Idaho statutes on air quality, water quality, and waste treatment and management are included in the ATR list of regulatory criteria. However, DOE has the responsibility for ensuring that these criteria, regulations, and statutes are met. Chapter 19 of the ATR UFSAR also lists permits required for ATR construction and operation. Key ATR operating terms and conditions (controls and commitments are summarized in the Authorization Agreement for the ATR.[9] This document summarizes existing key ATR requirements, such as those listed in the UFSAR, TSRs, National Environmental Policy Act (NEPA) environmental assessment requirements, DOE Order 425.1 for restart assessments, DOE Order 5480.21 for the Unreviewed Safety Question Determination Process, TRA Emergency Preparedness Plan, and Idaho Department of Health and Environmental Quality Water Pollution Control Permit.

Although ATR is not classified as a DOE defense facility, DOE regulation of ATR is indirectly affected by the Defense Nuclear Facilities Safety Board (DNFSB) actions, an independent agency that exercises an advisory role with respect to the safety of DOE nuclear defense facilities. In addition, DOE oversight of ATR includes three components. First, there is line management oversight by the DOE-Idaho Field Office. Second, there is independent oversight by DOE-NE and, to a limited extent, by DOE's Office of Environment, Safety, and Health. Third, there is formal DOE enforcement through the Price-Anderson Amendments Act (PAAA) and its implementing regulations. The PAAA of 1988 amended the Atomic Energy Act of 1954 (AEA) to add Section 234A to provide for a system of civil penalties for contractors who have entered into an agreement of indemnification with DOE. These penalties are decided by DOE-NE, although the field office may make recommendations regarding the magnitude of penalties.

In addition, ATR performance affects the annual incentive award fee paid to its operating contractor. Specifically, DOE and the Lockheed Martin Idaho Technologies Company (LMITCO) contractor, which currently operates the ATR, have agreed to specific performance-based incentives for the ATR. As indicated in Reference 10, incentive objectives include adhering to planned budget, operating efficiency, reduced unplanned outages, personnel exposure, and increased sales. In the case of the ATR, the performance-based financial incentives are substantial. Lost revenue due to performance-based incentive reductions may be more than the penalties applied through PAAA enforcement.

4.4 ATR-Specific Regulatory Issues

ATR significantly differs from commercial PWRs. It has a smaller core, higher power density, lower primary coolant system pressure and temperature, a greater ratio of coolant weight to power, and a confinement structure rather than a containment. The ATR core design requires that the impact of materials being irradiated in the flux trap be evaluated each time that the reactor is

returned to power. The confinement structure is designed to be a barrier to radionuclide release to the atmosphere, similar to a PWR containment building; however, ATR's confinement structure has a higher leakage rate and lower overpressurization structural failure limit. Also, the confinement does not require filtered venting during a severe accident because of ATR's lower pressure and lower power.

The TRA 670 building is approximately 200 x 200 feet and extends 60 feet above grade and 60 feet below grade. It houses the reactor, the experiment loop cubicles and equipment, storage canals, electrical equipment, the ATR Critical Facility, and required auxiliaries. The building is separated into gastight (confinement) and non-gastight areas, with further subdivisions within each subarea. (see Figures 3 and 4). The confinement area, also known as the controlled leakage zone, includes the reactor bay area (reactor main floor area), reactor control room, instrument repair room, data readout room, main floor lunch room, first basement, second basement, sub-basement areas, pipe tunnel and heat exchangers area, and the west and south stairwells. The confinement area is not a pressure containment zone. It is designed not to exceed, when sealed, the leakage rate of more than 2% and 10% of volume/day under normal and worst-case environmental conditions, respectively.

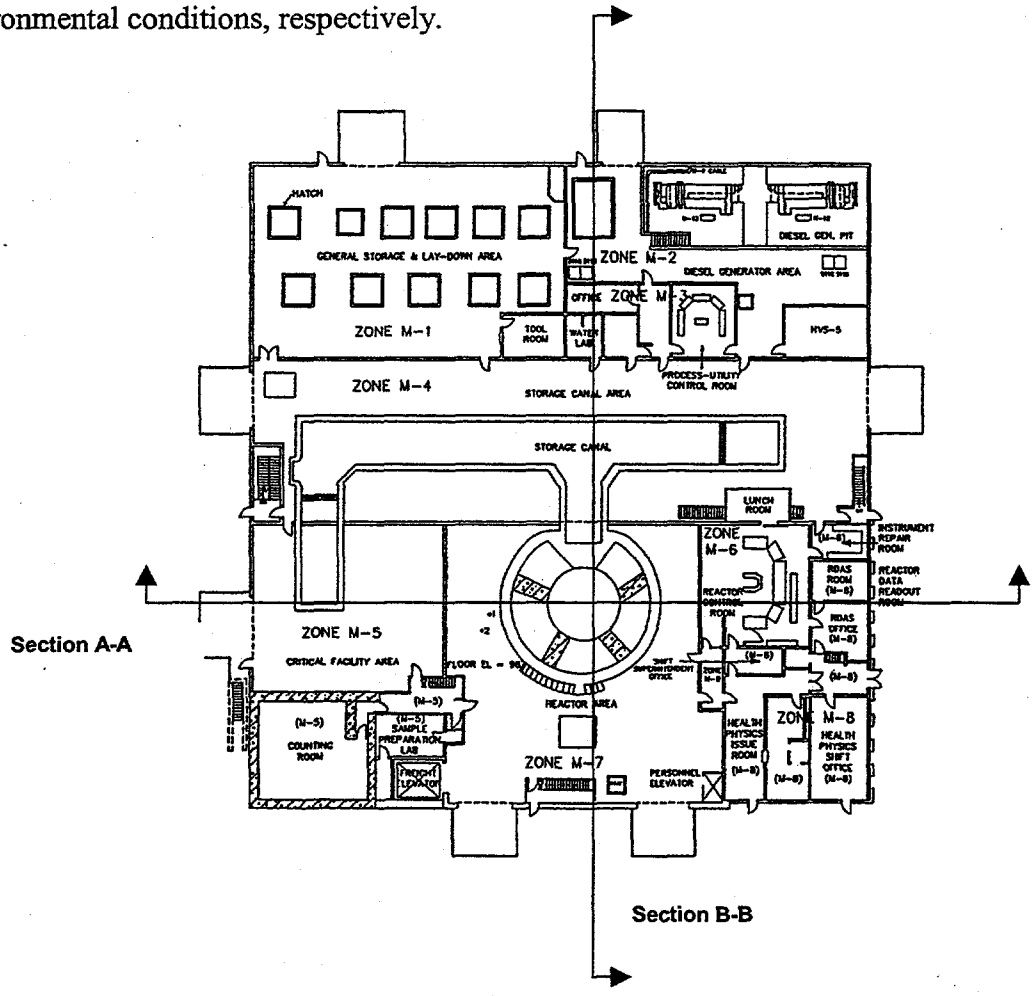
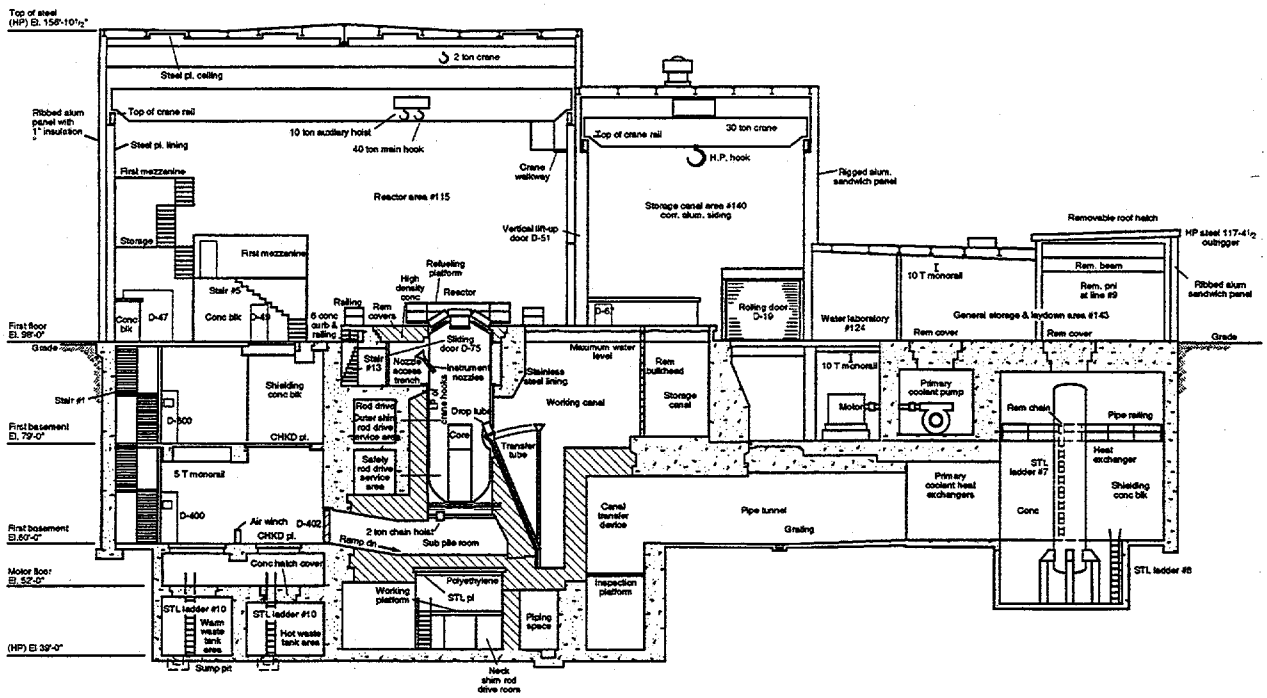
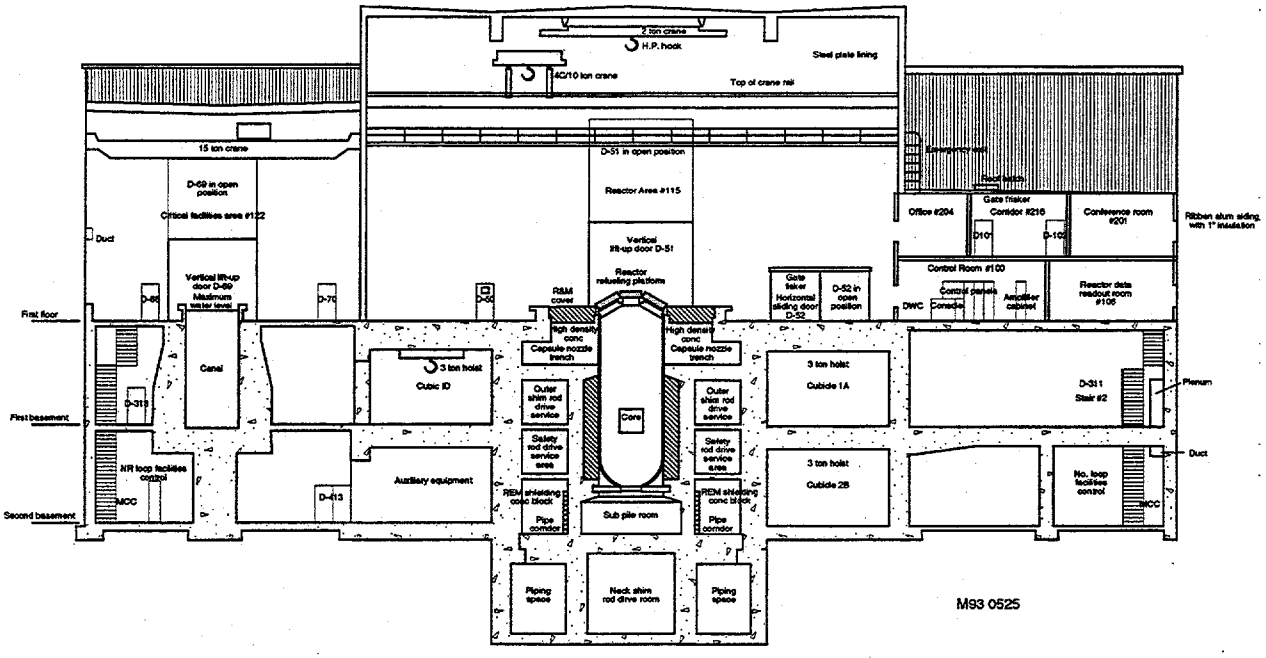


Figure 3. ATR building floor plan.



a) View A-A



b) View B-B

Figure 4. ATR confinement cross-sectional views

4.5 Available ATR Documentation

Preliminary review indicates that several documents should be considered in defining ATR performance criteria. These documents are listed below. Where appropriate, a summary statement about the status of this documentation is provided.

- "Advanced Test Reactor Upgraded Final Safety Analysis Report," 1997 AU, Rev. 1/09-28-98. This document was revised in 1989 to meet DOE Orders 5480.6 and 5481 that required new safety analysis reports that followed NRC guidelines for standard format and content.

- Eide, S. A., et al., 1991, "Advanced Test Reactor Probabilistic Risk Assessment," EGG-PRP-8823, Revision 1, EG&G Idaho, Inc., September 1991. The ATR PRA is being updated. A revised version is scheduled to be released in October 1999.

- Thatcher, T. A., et al., "Update to the Advanced Test Reactor Probabilistic Risk Assessment (Level 1, 2, and 3 Including Shutdown Operations), EGG-PRP-11229, EG&G Idaho, Inc., May 1994. The ATR PRA is being updated. A revised version is scheduled to be released in October 1999.

- Atkinson, et al., "Advanced Test Reactor Risk Summary," EGG-PRP-10025, December 1991.

- "ATR Technical Safety Requirements," Revision 2, April 16, 1999.

- "Authorization Agreement for the Advanced Test Reactor Facility," U.S. Department of Energy Idaho Operations Office and Lockheed Martin Idaho Technologies Company Document, IAG-31, Rev. 01.

- "ATR Performance Incentives," Lockheed Idaho Technologies Company Performance Based Incentive Fee Plan No. 95-6, FY 1995 Available Fee Pool Distribution (Rev. 5), Contract No. DE-AC07-94ID13223, Modification No. MO22, August, 18, 1995.

5 HOT CELL CASE STUDY: ANL HOT FUEL EXAMINATION FACILITY

Hot cell facilities were also considered as high priority for inclusion in the INEEL/MIT study. Hot cells were considered important because we wanted to consider a sufficiently complex facility type that had also been considered in the DOE/NRC pilot plant interactions. The HFEF was selected as a hot cell case study for this project. This facility is advantageous because it is an INEEL hot cell operated by ANL-W, who was willing to participate in this evaluation. The HFEF facility is also of interest because it is co-located with the NRAD reactor.

5.1 Facility Description

The Hot Fuel Examination Facility (HFEF) is located at INEEL's ANL-W site. Completed in 1974, the HFEF is a hot-cell complex for the preparation and examination of irradiation experiments in support of a wide variety of programs and process demonstrations. A wide range of remote operations and examinations may be performed in this facility with its shielded cells, support areas, and equipment. As shown in Figure 5, the three-story facility has two large, adjacent, shielded hot cells. The main cell (Figure 6), which is stainless steel-lined and gas tight, measures 70-ft long by 30-ft wide by 25-ft high and is filled with high-purity argon gas for work involving materials such as sodium, plutonium, and other materials which would react chemically with air. The decontamination (decon) cell (Figure 6), which is air filled, measures 20-ft long by 30-ft wide by 25-ft high. Both cells are surrounded by four-foot thick, high-density concrete to protect workers from the high radiation levels present in the hot cells.

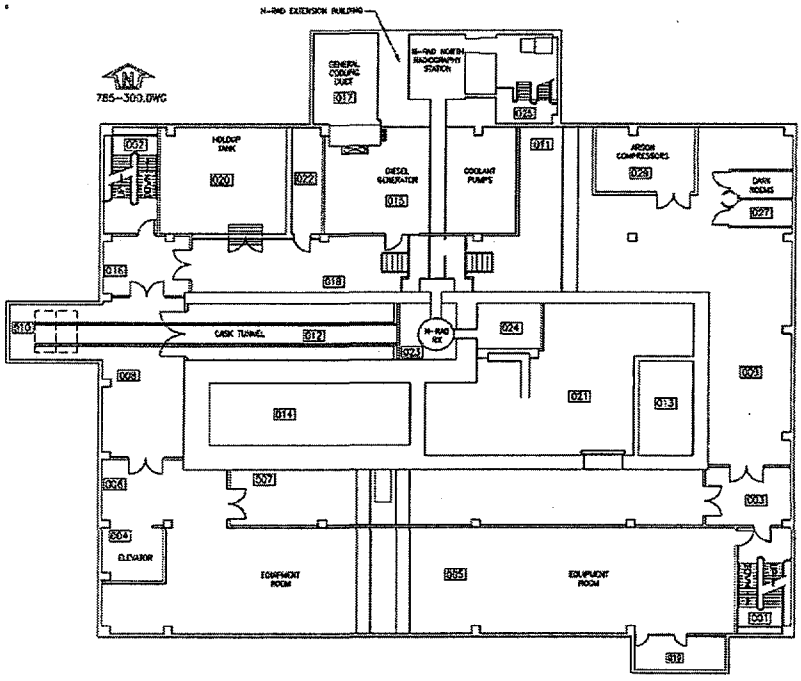


Figure 5. HFEF floorplan.

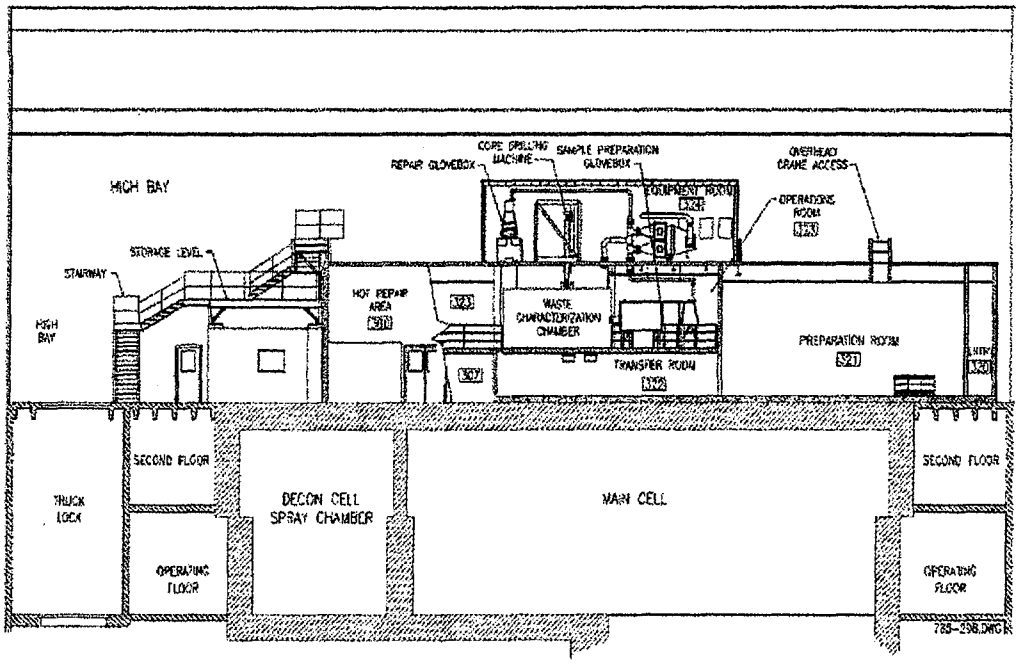


Figure 6. HFEF cross-sectional area.

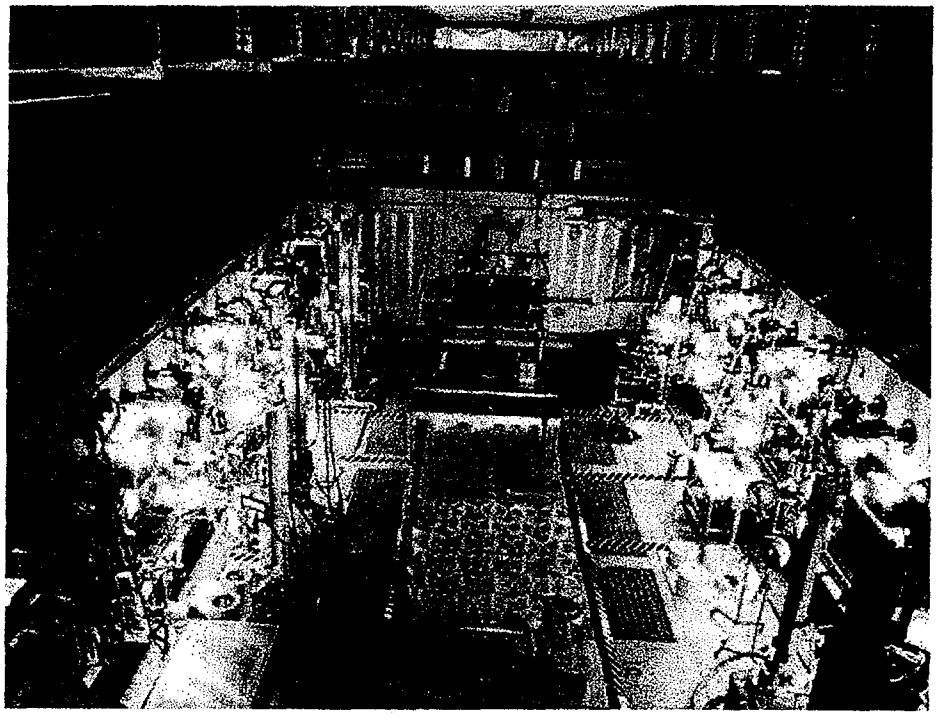


Figure 7. HFEF main cell.

The main cell (Figure 7) has 15 workstations, each with a 4-ft thick window of oil-filled, cerium-stabilized glass and a pair of remote manipulators. Two 5-ton overhead cranes and a 750-lb. electromechanical manipulator can access all locations in the main cell. The decon cell has six similarly equipped workstations, a 5-ton crane, and a 750-lb. electromechanical manipulator. Cell exhaust passes through at least 2 stages of High Efficiency Particulate Air (HEPA) filtration (some locations, such as the metallography containment box have separate exhaust outlets with additional filtration). Offices, laboratories, and other personnel-related areas are located on the operating floor, which is slightly above grade level. A truck lock at the west end of the cell complex is also at this level. The service floor below contains the subcell tunnels, and most of the building support equipment. The second floor contains additional building support equipment and offices.

A high bay area covering the entire cell complex and serviced by a 40-ton bridge crane provides access to the tops of the cells for bottom opening casks. This area contains the repair rooms, change room, and access room, and provides space for clean equipment repair and mockup. The Waste Characterization Area (WCA) is also located in the high bay area. The Waste Characterization Chamber (WCC) is an enclosure with a controlled environment which provides the primary confinement for the contact handled transuranic (CH-TRU) waste during characterization. The characterization operations are performed through sealed glove openings or using robotic manipulators in order to protect personnel, the process, and the environment.

The NRAD reactor, which is located in the HFEF subcell, is a Training, Research, and Isotope, General Atomics (TRIGA)-type 250 kWt nuclear reactor that is installed and operated as a neutron source for neutron radiography of nuclear fuels and materials. Transported from Puerto Rico Nuclear Center, the reactor began operation in 1977 at ANL-W. It is equipped with two beam tubes and two separate radiography stations. Historically, the facility has provided support to reactor development programs, but is now involved in non-destructive assay techniques for spent fuel and waste programs. The NRAD facility is independent with its own ventilation system that is operated by personnel from a different division. Hence, NRAD is not considered as part of the HFEF, but its proximity will be considered in HFEF evaluation.

Radioactive materials are introduced into the hot cells via a shielded cask on a transfer cart that accesses the cells through an under-cell transfer tunnel. At the time of this writing, the transfer tunnel and transfer cart are being modified to accommodate commercial shipping casks up to 28 tons in weight. A penetration through the top of the main cell also allows the introduction of large components or fuel assemblies. Large sodium test loops have been inserted into the cell through this penetration.

5.2 Facility Missions

Significant projects that have been performed in HFEF include 1) examination of fuels and components for the Liquid Metal Fast Breeder Reactor (LMFBR) and Integral Fast Reactor (IFR) programs, 2) preparation and post-test examinations of specimens irradiated in the Transient Reactor Test Facility (TREAT) and the Sodium Loop Safety Facility (SLSF), and 3) examination of full length fuel assemblies and rods as well as sections and other specimens irradiated in the

Fast Flux Test Facility (FFTF). Ongoing, scheduled, and anticipated HFEF programs include: 1) examination of tritium production burnable absorber rods (TPBARs), 2) demonstration of fabrication of waste forms from electrometallurgical treatment of spent fuel, 3) characterization of waste in drums prior to disposal in the Waste Isolation Pilot Plant (WIPP), and 4) proposed examination of fuel rods irradiated in the ATR or the Transient Reactor Test Facility (TREAT).

5.3 Current Regulatory Criteria

Appendix A of the HFEF Safety Analysis Report (SAR)[11] summarizes statutes, rules, regulations, and DOE Orders that are applicable to HFEF operation. As indicated in Appendix A, HFEF adheres to a range of rules, which address facility safety, worker health and safety, environmental protection, emergency planning, and reporting of unusual occurrences. Newer rules are codified in the Code of Federal Regulations (10 CFR Part 800, 40 CFR Parts 200-300, 49 CFR Part 100, etc.); whereas older rules are found in DOE Orders and Standards. It should be noted that these rules are more stringent than other applicable criteria, such as the State of Idaho requirements for liquid and gaseous effluents. To overcome duplication among the various requirements, ANL-W generated their own documents that implement applicable sections of statutes, rule, regulations, and DOE Orders. HFEF "day to day operation" requirements are established in these ANL-W documents. Rules and regulations address safety and environmental issues, such as the handling, management, and disposal of radioactive materials, criticality safety, emergency response, liquid and gaseous effluence, waste, decontamination, and decommissioning.

DOE oversight of HFEF includes three components. First, there is line management oversight by the DOE-Chicago Field Office. Second, there is independent oversight by DOE-NE and, to a limited extent, by DOE's Office of Environment, Safety, and Health. Third, there is formal DOE enforcement through the Price-Anderson Amendments Act (PAAA) and its implementing regulations. The PAAA of 1988 amended the Atomic Energy Act of 1954 (AEA) to add Section 234A to provide for a system of civil penalties for contractors who have entered into an agreement of indemnification with DOE. These penalties are decided by DOE-NE, although the field office may make recommendations regarding the magnitude of penalties. However, facilities operated by "not-for-profit" organizations, such as the University of Chicago, that operates ANL, are not currently required to pay fees assessed under PAAA.

A performance incentive program is also used at HFEF. In this program, DOE and the University of Chicago, which currently operates ANL-W, have agreed to specific goals for HFEF performance, such as limits for personnel exposure and worker injuries. If these goals are met, DOE awards the University of Chicago additional funds.

5.4 Facility-specific Regulatory Issues

The HFEF waste streams include high-level radioactive waste, radioactive remote-handled low-level waste, remote-handled mixed waste, contact-handled low-level waste, and radioactive liquid waste. HFEF regulation addresses typical hot cell issues, such as public protection, worker protection, the handling, management, and disposal of radioactive materials, criticality safety, emergency response, liquid and gaseous effluents, waste, decontamination, and

decommissioning. Regulations also address the five types of accidents considered in the HFEF FSAR: operator errors or material handling equipment malfunctions, cell structural or equipment malfunction, process malfunctions (missiles, chemical reactions, etc.), criticality accidents, and externally initiated events (fires, floods, plane crashes, etc.).

One unique aspect of the HFEF is that the Neutron-RADiography (NRAD) reactor is located in a subcell of the HFEF building. As noted above, the NRAD facility is independent with its own ventilation system and is not considered in HFEF safety analyses. Hence, HFEF regulation does not address any NRAD-related issues.

5.5 Available Documentation

- Argonne National Laboratory, "Safety Analysis Report for the Hot Fuel Examination Facility, Revision 00 (Draft), Doc. No W7850-0117-ES.
- Argonne National Laboratory-West, Technical Safety Requirements for HFEF Use and Application, Revision 0D (Draft), Doc. No W7850-0140-ES.

6 WASTE STORAGE SITE CASE STUDY: TMI-2 INDEPENDENT FUEL STORAGE INSTALLATION

INEEL and MIT considered waste storage sites important to study in this project because of their increasing importance in the DOE complex and because their contamination levels encompass problems associated with lower levels of radiation at contaminated sites. The Three Mile Island Unit 2 Independent Spent Fuel Storage Installation (TMI-2 ISFSI) was selected as a waste storage site case study for this project. This facility is of interest because NRC recently licensed it (because of an agreement between DOE, the Navy, and the State of Idaho).[6] A second reason for selecting this facility is that INEEL and DOE management for this facility agreed to support INEEL and MIT in this effort. Finally, the location of the ISFSI at INEEL facilitates project interactions. This section discusses ISFSI design features, documentation, current regulation, and facility-specific regulator issues that will be considered.

6.1 Facility Description

This ISFSI was recently constructed at INEEL's INTEC site for interim storage of TMI-2 core and core handling debris. A modified NUHOMS® spent fuel storage system, designated NUHOMS®-12T, is used at INEEL's ISFSI facility. NUHOMS® is a proven system for dry storage, which has been in use at reactor sites since March of 1989.

The TMI-2 core debris is currently in stainless steel canisters, which are being transferred from a fuel pool at another INEEL site, Test Area North (TAN). Each NUHOMS®-12T module provides for horizontal dry storage of up to 12 TMI-2 canisters inside a dry shielded canister (DSC), which is placed inside a concrete horizontal storage module (HSM). Figure 8 shows a DSC being transported to the HSMs at ISFSI. The schematic in Figure 9 illustrates the placement of the DSC within a HSM.

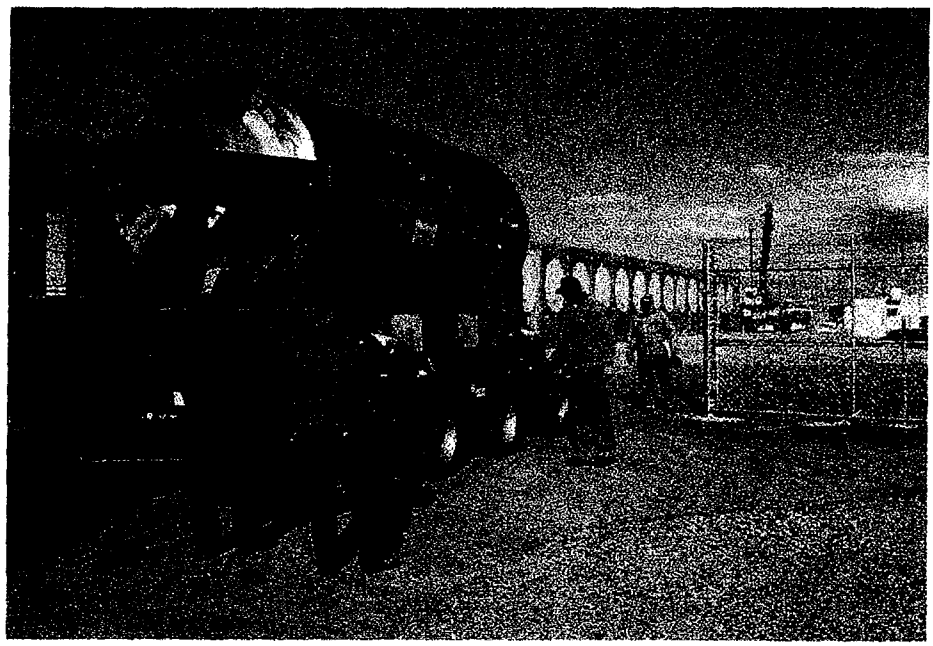


Figure 8. Transport of DSC to HSMs at the TMI-2 ISFSI site.

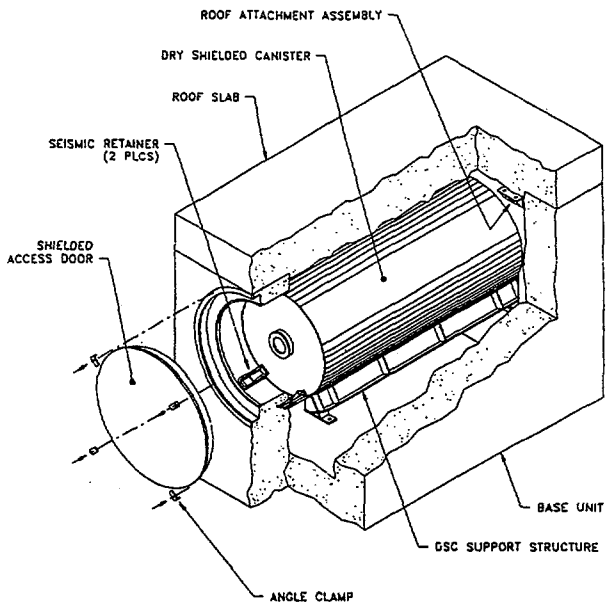


Figure 9. DSC placement within a TMI-2 ISFSI HSM.

The INEEL TMI-2 ISFSI is designed to provide temporary dry storage for 100% of the TMI-2 canisters. The prefabricated concrete HSMs are arranged in two rows, which allow access for loading of the DSCs, and for inspection and monitoring of the vent system. The HSM is a massive reinforced concrete structure that provides protection for the DSC against tornado missiles and other potentially adverse natural phenomena. The HSM also serves as the principal biological shield for the TMI-2 canisters during storage. Adjacent HSMs have at least a six-inch space between them to permit airflow and to allow for independent motion of each HSM during a seismic event. For the TMI-2 ISFSI , the HSM includes an access door on the back wall for monitoring and maintenance of the DSC vent and purging HEPA filters.

Each HSM holds a DSC containing up to 12 TMI-2 canisters. Therefore, 29 DSCs will contain all existing (344) TMI-2 canisters with four canister open spaces. The ISFSI design includes an extra HSM with a pre-installed DSC overpack in case a challenged canister needs additional confinement. The INEEL TMI-2 ISFSI and NUHOMS®-12T components allow retrieval of the TMI-2 canisters for further processing, alternate storage, or disposal.

The ISFSI is unique because it contains the adapted NUHOMS® system contains fuel debris (rather than fuel assemblies). Most notable differences between the TMI-2 canisters and commercial fuel assemblies are:

- TMI-2 core debris is canisterized whereas commercial fuel is clad. The canisters contain TMI-2 core debris and debris from core handling equipment used in TMI-2 examinations and clean up activities.
- The TMI-2 canisters provide a much stronger structural element, as compared to commercial fuel assemblies, for support within the DSC basket.

- The heat load for the TMI-2 canister (maximum 60 watts, average 29 watts) is much less than a commercial spent fuel assembly (approximately 1000 watts).
- The TMI-2 canisters have the potential for hydrogen gas generation due to radiolysis.

The NUHOMS® system was modified to accommodate these conditions. Specifically, the NUHOMS®-12T DSC includes venting of the DSC through HEPA grade filters during storage. The vent system allows hydrogen gas release and monitoring and/or purging of the system during operation.

6.2 Facility Missions

The sole mission of the TMI-2 ISFSI is to provide temporary dry storage for the core debris from the TMI-2 reactor.

6.3 Facility-specific Regulatory Issues

The INEEL TMI-2 ISFSI is located on a DOE facility, which operates under DOE Regulations, Orders and Directives. However, DOE's schedule for licensing, construction and operation of the INEEL TMI-2 ISFSI was established by the Settlement Agreement entered into by the State of Idaho, the Department of Energy, and the Department of Navy.[6] This agreement requires that DOE also have an NRC license for the TMI-2 ISFSI. In areas where NRC and DOE specify different requirements, NRC regulations shall apply and have precedence over DOE Orders, Requirements and Guidelines.

The Manager of DOE-ID is the ISFSI NRC license holder. This authority was delegated and responsibility was assigned to the DOE-ID Manager by the Secretary of Energy pursuant to 10 CFR 72.16(b) in Delegation Order No. 10CFR72.512.1. As the facility owner and licensee, DOE retains ultimate responsibility for the safe operation of the facility and for compliance with all license conditions. The DOE utilizes a Management and Operating (M&O) contractor for TMI-2 ISFSI activities. The authority for the management and operation of the facility is contractually delegated and the responsibility for compliance with license requirements and applicable regulations is contractually assigned to the INEEL M&O contractor. DOE maintains their ultimate responsibility by (1) performing independent audits of the M&O contractor's TMI-2 ISFSI Quality Assurance (QA) program (both the achievement of quality by M&O contractor management and the verification of quality by M&O contractor QA personnel), (2) ensuring the license requirements for the facility are included in the M&O contract, (3) assessing the performance of the M&O contractor against the terms of the contract, (4) retaining the responsibility to budget funds necessary and sufficient to safely operate the facility, and (5) retaining the authority to revise the M&O contract in the event contract deficiencies are found relative to proper implementation of license requirements.

As with any dry fuel storage facility, regulations center on the potential for criticality, the potential for hydrogen production from radiolysis, and the release of radioactive materials. The INEEL TMI-2 ISFSI is designed in accordance with the following general design criteria:

- 10 CFR 72.122 Overall Requirements
- 10 CFR 72.124 Criteria for Nuclear Criticality Safety
- 10 CFR 72.126 Criteria for Radiological Protection
- 10 CFR 72.128 Criteria for Spent Fuel, High-level Radioactive Waste, and Other Radioactive Waste Handling and Storage
- 10 CFR 72.130 Criteria for Decommissioning

The TMI-2 ISFSI storage components that are important to safety are the DSCs (including the vent system) and the HSMs. Consequently, they are designed and analyzed to perform their intended functions under the extreme environmental and natural phenomena specified in 10 CFR 72.122[12] and ANSI-57.9.[12] The design and operation of the TMI-2 ISFSI ensures that a single failure will not result in the release of significant radioactive material.

The ISFSI QA Program satisfies the requirements of 10 CFR Part 72, Subpart G, and the criteria in the NRC's NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities." This QA Program ensures that essential technical and quality requirements for structures, systems, and components (SSCs) classified as important to safety are achieved and documented throughout all design, fabrication, construction, testing, operations, modifications and decommissioning activities. The ISFSI quality assurance program is described in the DOE's Office of Civilian Radioactive Waste Management's Quality Assurance Requirements and Description, DOE/RW-0333P, Revision 5 (QARD).[14]

Consistent with the DOE-ID's overall commitment to keep occupational radiation exposures As Low As Reasonably Achievable (ALARA), specific plans and procedures are followed by ISFSI operations personnel to ensure that ALARA goals are achieved consistent with the intent of NRC Regulatory Guides 8.8[15] and 8.10[16] and the requirements of 10 CFR Part 20. Since the ISFSI is a passive system, minimal maintenance is expected on a normal basis. Maintenance activities that could involve significant radiation exposure of personnel are carefully planned utilizing previous operating experience and conducted using well-trained and certified personnel and proper equipment. Where applicable, formal ALARA reviews are prepared which specify radiation exposure reduction techniques, such as those set out in Regulatory Guide 8.8. 10 CFR Part 20 establishes the policy, requirements, and training necessary for assignment and use of external dosimetry.

Cask transfer operations and maintenance of the HEPA grade filters in the DSC vent system are the only activities that will generate waste during the design operating life of the system. This waste will be in the form of dry radioactive waste. On the average, the filters could be replaced five times during the 50-year life of the system. It is estimated this would consist of about one cubic foot per DSC over the design life of the TMI-2 ISFSI (a total of less than 30 ft^3). Decommissioning activities at the time of TMI-2 ISFSI closure is estimated to generate less than 10 ft^3 per module (a total of less than 300 ft^3). The HSM and concrete basemat would be

disposed of as clean free release material after radiological surveys and any necessary decontamination.

The TMI-2 ISFSI is located within the site boundaries of the INTEC with several other DOE owned facilities and DOE managed programs. The INEEL has its own large security police force, a fire department, medical staff, emergency response teams, and full-time INTEC shift plant supervision. Thus, the INEEL infrastructure will be considered to serve equivalent functions as independent local agencies (similar to local city or county) do for typical commercial licensed sites.

Normal INTEC operations will not affect operation of the TMI-2 ISFSI. Emergency situations, unrelated to the TMI-2 ISFSI operations, which would require personnel to evacuate the plant area, or take cover, could cause temporary interruptions to normal TMI-2 ISFSI operations (loading, unloading, and surveillance). These interruptions would not compromise safety.

6.4 Available Documentation

- "The Safety Analysis Report for the INEEL TMI-2 Independent Spent Fuel Storage Installation," Docket No. 72-20, Revision 1, March 1999.

- "Technical Specification Bases for Three Mile Island Unit 2 Independent Spent Fuel Storage Installation," March 1999.

- U.S. Government, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation (ISFSI)," Title 10 Code of Federal Regulations, Part 72, Office of the Federal Register, Washington, D.C.

7 SUMMARY/KEY ISSUES

In summary, a three-year joint INEEL/MIT effort has been initiated to develop PBR for DOE facilities. As part of this effort, a general framework will be defined from which performance based goals can be selected and appropriate indicators defined for demonstrating compliance. The approach will be demonstrated using three DOE facilities as case studies.

This letter report documents the status of an initial task project, "Selection of DOE Facilities for Case Studies." In this task, we reviewed DOE facilities and identified seven key facility groups: Reactors, Hot Cells, Waste Storage Sites, Nuclear Materials End-Use Facilities, Contaminated Sites, Manufacturing/Process Facilities, and Accelerators. We then determined a prioritization of facility groups for evaluation as case studies and selected representative facilities as case studies from three higher-priority DOE facility groups. To assist us in our review of DOE facilities, we used ACCESS software to create the NUclear FACility (NUFAC) searchable database into which information about various DOE facilities could be entered. This database allows comparisons between the facilities selected as case studies and other facilities in each group.

From groups identified as having higher priority for inclusion as case studies in this program, representative facilities were selected. As noted within this document, the facilities from three facility groups were selected as case studies.

Reactor Group:	INEEL Advanced Test Reactor
Hot Cell Group:	ANL-W Hot Fuel Examination Facility
Waste Storage Site:	INEEL TMI-2 Integrated Spent Fuel Storage Facility

In this letter report, representative facilities are described, emphasizing aspects of interest to regulation. Criteria under which case studies are currently regulated are discussed. Preliminary insights from this review are summarized below.

- Overlap exists between facility group boundaries. For example, it is often difficult to determine whether a facility falls within the "high–level waste" or the "contaminated" site groups. Likewise, it is often difficult to distinguish between "manufacturing" and "end-use" facilities.
- Existing DOE facility regulation includes DOE criteria, Occupational Safety and Health Administration (OSHA), Environmental Protection Agency (EPA), State and NRC (for ISFSI) requirements.
- Existing regulation is so duplicative that licensees often write their own documents identifying their interpretation of the applicable, limiting, regulation.
- Facility regulation focuses on similar issues: criticality safety, personnel safety, emergency preparedness, accident analyses, initial testing, in-service inspection, testing and maintenance, radioactive and hazardous material management, design of structures, components, equipment, and systems, air quality, water quality, waste treatment and management, and facility decontamination and decommission.

During the next year, facility documentation, designs, and regulations will be reviewed in additional detail to identify appropriate performance indicators from which PBR will be proposed.

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1999 URC ANNUAL REPORT
SUBMISSION GUIDELINES

General Information

Requested Information	Your Response
FULL PROJECT TITLE	Thorium Dioxide – Uranium Dioxide Fuels for Light Water Reactors
PROJECT NUMBER	
PRINCIPAL INVESTIGATOR'S NAMES	Mujid Kazimi and J. Stephen Herring
INSTITUTION	Department of Nuclear Engineering, Massachusetts Institute of Technology, and the INEEL
MAILING ADDRESS	77 Mass. Ave, Cambridge, MA 02139 and INEEL MS 3860
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COLLABORATOR'S NAMES AND INSTITUTIONS	Profs. Mujid Kazimi, Michael J. Driscoll and John E. Meyer Massachusetts Institute of Technology J. Stephen Herring, Philip E. MacDonald, Kevan Weaver and Craig Kullberg, Idaho National Engineering and Environmental Laboratory
FUNDING DATE	Nov. 1998
REPORT PREPARATION DATE	June 1999

Publication/Presentation Information

Title	Journal/ Conference	Status/Date	Authors
"Low Cost, Proliferation Resistant, Uranium-Thorium Dioxide Fuels for Light Water Reactors"	***Nuclear Engineering and Design***	Submitted March 29,1999	J. Stephen Herring and Philip E. MacDonald
Proliferation-Resistant, Thorium-Uranium Fuels	***Proliferation-Resistant Nuclear Power Systems: A Workshop on New Ideas***	Presented June 3, 1999	J. Stephen Herring and Philip E. MacDonald
"Advanced, Lower-Cost Proliferation-Resistant, Uranium-Thorium Fuels for LWRs"	***ANS 1999 Annual Meeting, Boston***	Presented June 9, 1999	J. Stephen Herring and Philip E. MacDonald
"Rationale for Reconsidering the Thorium Cycle in Light Water Reactors	***ANS 1999 Annual Meeting, Boston***	Presented June 9, 1999	Xianfeng Zhao, Michael J. Driscoll and Mujid S. Kazimi
"Mixed Thorium-Uranium Dioxide Fuel for High Burnup in Light Water Reactors"	***Global 99, "Nuclear Technology – Bridging the Millennia"***	To be presented Sept. 1-3, 1999	J. Stephen Herring and Philip E. MacDonald

Participant Information

Participants	Educational Status	Fraction of time spent on project (per fiscal year Oct 1-Sept 30)	(Students only) Degree Program	(Students only) Starting date	(Students only) estimated finishing date
Mujid S. Kazimi	Professor	10 %			
Michael J. Driscoll	Professor	15 %			
John E. Meyer	Professor	10%			
Ronald Ballinger	Professor	05%			
Kenneth Czerwinski	Professor	10 %			
Steve Schultz	Res. Scientist	20 %			
Pavel Hejzlar	Professor	15 %			
Kevin Clarno	Student	50 %	SB	Nov 98	June 99
M. P. Reynard	Student	50 %	PhD	Nov 98	2002
Philip J. LaFond	Student	40 %	SM	Nov 98	2001
Xianfeng Zhao	Student	50 %	PhD	Nov 98	2001
Yun Long	Student	100 %	PhD	Nov 98	2001
J. Stephen Herring	INEEL staff	40 %			
Philip MacDonald	INEEL staff	35%			
Kevan Weaver	INEEL staff	40 %			
Craig Kullberg	INEEL staff	60 %			

Personnel Exchange Information

Name of University Personnel who spent time at INEEL Laboratories – and dates attended	Prof. Kazimi , 12/4/98 and1/12/99. Professors. Kazimi, Driscoll, Ballinger, Schultz and Hejzlar Students: Yun Long and Xianfeng Zhao 7/26-28/99
Name of INEEL personnel who spent time at your University Laboratory - and dates attended	Steve Herring 11/19-21/98, 2/22-23/99, 4/5-8/99, 6/7-15/99 Phil MacDonald, 2/22-23/99, 4/5-7/99, 6/7-9/99 Kevan Weaver 4/5-8/99, 6/11-16/99

"Follow-On" Research Proposals

Title	Agency	Proposers	Status
Advanced Proliferation Resistant, Lower Cost, Uranium-Thorium Fuels for Light Water Reactors	DOE-Nuclear Energy Research Initiative (NERI)	Philip E. MacDonald (PI), J. Stephen Herring, Michael Driscoll, Mujid Kazimi, James S. Tulenko, Al Solomon, Joseph Sapyta Collaboration with MIT, Purdue, U of Florida, B&W, Westinghouse, ABB, Siemens and the INEEL (lead)	**Awarded,** 3 yr project, $3 million total funding

Sustainable Nuclear Energy

A Strategic Nuclear Research Collaboration
Between
Massachusetts Institute of Technology
And
Idaho National Engineering and Engineering Laboratory

Proliferation Resistant, Low Cost Thorium Dioxide – Uranium Dioxide Fuels for Light Water Reactors

Status Report
October 1998-June 1999

Project Coordinators

Mujid S. Kazimi
J. Stephen Herring

INEEL/MIT Strategic Nuclear Research Collaboration

Proliferation Resistant, Low Cost Thorium Dioxide – Uranium Dioxide Fuels for Light Water Reactors

Status Report
October 1998- June 1999

Project Coordinators
Mujid S. Kazimi
J. Stephen Herring

Project Objectives

Our objective is to develop a fuel consisting of mixed thorium dioxide and uranium dioxide (ThO_2-UO_2) for existing light water reactors (LWRs) that (a) is less expensive overall than the current uranium-dioxide (UO_2) fuel, (b) allows longer refueling cycles and higher sustainable plant capacity factors, (c) is very resistant to nuclear weapons-material proliferation, (d) results in a more stable and insoluble waste form, and, (e) generates less high level waste. This status report presents the results of our initial investigations and our plans for developing this fuel.

Project overview

Preliminary calculations using the SCALE 4.3 code system indicate that the mixed ThO_2-UO_2 fuel, with about 5 wt % of the total heavy metal U-235, could be burned to 72 MWd/kg (megawatt days per kilogram) using 25 wt % UO_2 and the balance ThO_2. The ThO_2-UO_2 cores can also be burned to about 100 MWd/kg using 35 wt % UO_2 and 65 % ThO_2 with an initial enrichment of about 7 wt % of the total heavy metal fissile material. Economic analyses indicate that the ThO_2-UO_2 fuel will require less separative work and less total heavy metal (thorium and uranium) feedstock. Even if the cost of fabricating the mixed ThO_2-UO_2 fuel is $100/kg greater, the cost of the ThO_2-UO_2 fuel would be 13 % to 25 % less than that of the fuels using uranium only. Our intent is to verify the neutronic calculations using more accurate Monte Carlo based calculations through MCNP-ORIGEN and also the widely used industry code CASMO-4. For the same fuel assembly design conditions, ThO_2-UO_2 fuel will operate cooler, and retain within the fuel more of the fission products, especially the gases, ThO_2-UO_2 fuel can probably be successfully operated to higher burnups than UO_2 fuel. This will allow for longer refueling cycles and better plant capacity factors. This improved fuel conductivety also could allow other , more optimized, design conditions for the PWR fuel .

The uranium in our calculations remained below 20 wt % total fissile fraction throughout the cycle, making it unusable for weapons. Total plutonium production per MWd was a factor of 4.5 less in the ThO_2-UO_2 fuel than in the conventional fuel. Pu-239 production per MWd was a factor of 6.5 less in the ThO_2-UO_2 fuel than in the conventional fuel. The plutonium produced was high in Pu-238, leading to a decay heat 5 times greater than that from plutonium derived from conventional fuel and 40 times greater than weapons grade plutonium. This level of decay heat will require active cooling of any crude weapon, lest the components surrounding the plutonium be melted. Spontaneous neutron production for plutonium from ThO_2-UO_2 fuel was 2.3 times greater than that from conventional fuel and 15 times greater than that from weapons grade plutonium. High spontaneous neutron production drastically limits the probable yield of a crude weapon.

Because ThO_2 is the highest oxide of thorium, while UO_2 can be oxidized further to U_3O_8 or UO_3, ThO_2-UO_2 fuel appears to be a superior waste form if the spent fuel is ever to be exposed to air or oxygenated water. And, finally, use of higher burnup fuel will result in proportionally fewer spent fuel bundles to handle, store, ship, and dispose of in a permanent geological repository.

Outline of MIT Tasks

The following tasks are to be performed under this project:

1) **Neutronic evaluation of the performance of thoria based fuel for a range of burnups from the current 55MWD/kg to double that amount at 110MWD/kg. This will be accomplished in three stages:**

1a) Review the literature to ascertain the lessons from the past experience with thorium based fuels in experiments and reactors as well as in wide scope studies such as the International Nuclear Fuel Cycle Evaluation (INFCE) project.

1b) Apply the most up to date version of CASMO-4 to characterize the reactivity of the mixed fuel in a PWR assembly for a variety of initial loadings and uranium enrichments.

1c) Apply the Monte Carlo code MOCUP (MCNP plus ORIGEN2) to benchmark the behavior of a pin cell, a PWR assembly., and an entire core if necessary.

2) **Fuel performance modeling to assess the behavior of the thoria-urania mixture and the higher burnup contemplated for the lifetime of the fuel. This task will use an MIT-collected Fuel Lifetime Analysis (FLA) package and will concentrate work in three areas:**

2a) Estimate the severity of cladding corrosion for conceptual Th-U fueled cores. Initial calculations will be performed using the code developed for the recent MIT thesis by L. Garcia-Delgado -- that code has been used in the thesis for cladding constructed of Zircaloy-4 with reduced tin content (1.3 w/o Sn).

2b) Explore means to extend the corrosion calculations of item (2a) to newer cladding materials (such as Westinghouse Zirlo or the Siemens duplex material).

2c) Gather information on the fission gas generation and release behavior of Th-U fuels.

2d) Adopt (and possibly modify) the FRAPCON-3 fuel performance code for use in the MIT effort. This code has been developed by Battelle Pacific Northwest Laboratory and by INEEL over a number of years. It should provide a proven basis for new fuel performance analyses.

3) **Waste behavior implications of including thoria and of extending the burnup of the fuel. Several elements are:**

3a) Assessing the long-term implications of the mix of fission products and transuranics for the inherent hazard potential of the radioactivity involved.

3b) Assessing the implications of the presence of uranium dioxide on the behavior of the more stable thorium oxide to determine if the stability holds up within the range expected for uranium in the mixture (to 50%).

3c) Assess the implications of the differences in the decay heat and fission product content on behavior of the waste form and package design.

4) Thermal-Hydraulics and Safety

Review of possible design optimization of thoria based fuels due to differences from urania based fuels. Important features include an enhanced fuel thermal conductivity, a lower peak to average power profile in the core, and a slightly higher decay heat due to the expected higher burnup and to the higher U-233 decay heat compared to U-235.

Outline of INEEL Tasks

1 **Neutronics calculations:** Review literature and preliminary data to develop detailed scope for neutronic calculations materials evaluations, and fuel stability modeling. Modify the depletion-transport code MOCUP, for use with the thoria-urania pin-cell and other calculations. Assist MIT students in using MOCUP.

2 Identify candidate fuel configuration and perform detailed neutronic calculations, particularly using Monte Carlo codes that will later be used in conjunction with nodal methods for whole core evaluations.
2a) Determine burnup effects, decay heat, control and stability characteristics
2b) Power peaking, void and temperature coefficients,

3 Fuel performance evaluations for candidate compositions with favorable neutronic calculations
3a) Cladding lifetime, corrosion, erosion, mechanical interaction
3b) Design of fuel performance tests

4 **Literature search and investigation of long-term stability of waste forms, including radionuclide thermodynamics and mobility**

5 Determine the probable economics of a fuel using mixed thoria and urania.
5a) Compare separative work and raw materials requirements for the thoria-urania fuels and for conventional urania fuel
5b) Include fabrication considerations, including special handling requirements, criticality issues in the fabrication facility, special process for assuring the uniformity in mixing thorium and uranium feedstocks.
5c) Determine the impact of longer cycle lengths, including carrying charges on fabrication and enrichment costs, higher plant capacity factors and less frequent refueling outages.

6 Compare the safety of conventional UO_2 fuel and the mixed ThO_2-UO_2 fuel.
6a) Radionuclide releases from the two types of fuel under accident conditions
6b) Compare the decay heat in the two fuels at various levels of burnup.

Accomplishments October 1, 1998 to June 30, 1999

Highlights:

The MIT team for this work has been formed as follows:
Project Coordinator: Prof. Mujid Kazimi,
Neutronics and Fuel Management: Prof. Michael Driscoll
grad student Xianfeng Zhao and undergraduate student Kevin Clarno
Fuel Performance Modeling: Profs. John Meyer and Ronald Ballinger and Dr. Stephen Shultz
grad student Yun Long,
Waste Behavior: Prof. Kenneth Czerwinski
grad student Mike Reynard

Thermal and Safety Considerations. Prof. Pavel Hejzier (Tech. U. of Prague) and Prof. Kazimi grad student Philip Lafond

The INEEL team is:

Project Coordinator: Dr. J. Stephen Herring,
Neutronics: Dr. Kevan Weaver and Craig Kullberg
Fuel Performance Modeling: Philip E. MacDonald and Richard McCardell
Waste Behavior: Philip MacDonald
Thermal and Safety Considerations. Dr. David A. Petti

A literature search has been conducted in all the four areas of concern and a draft report on the was issued in April 1999 as "On the Use of Thorium in LWRs", MIT-NFC-0016.
A NERI proposal for the development of this fuel was awarded based on a proposal led by INEEL with contributions from MIT. Additional collaborators on the proposal are Argonne National Laboratory, Westinghouse, ABB, B&W/Framatome, Siemens, Purdue University and the University of Florida. Funding is at a level of $1M/yr for a period of three years
MIT took the lead in organizing a special session on the use of thorium in LWRs at the June meeting of the American Nuclear Society which was followed by a panel discussion.

Two paper summaries were contributed to the Special Session on Thorium Fuel at the American Nuclear Society meeting in June 1999. MIT contributed a paper summary on the neutronic imperatives for re-examining the inclusion of thorium in LWR fuel and the INEEL contributed a paper summary on fuel performance characteristics of the mixed thoria-urania fuel.

A report on the motivation to re-examine the use of thorium in LWRs and the needed R&D given past experience with thorium fuels was published by MIT in April 1999.

A paper on the mixed thoria-urania fuel by the INEEL has been accepted for the Global 99 International Conference on Future Nuclear Systems in Jackson, Wyoming, August 29, - September 3, 1999.

INEEL has submitted a journal article to *Nuclear Engineering and Design* based on the INEEL external report, "Characteristics of a Mixed Thorium-Uranium Dioxide High-Burnup Fuel," INEEL/EXT-99-00094."

The INEEL has been invited to submit a paper on the weapons proliferation resistance of thoria-urania fuel to the journal *Science and Global Security.*

Specific Accomplishments

Neutronics and Fuel Management

MIT Faculty: Michael Driscoll MIT Students: Xianfeng Zhao and Kevin Clarno
INEEL Participants: Steve Herring and Kevan Weaver

During this project startup phase, work at MIT was focused on the following:

*A comprehensive review of past literature on the use of thorium in LWR cores.

*Importation of codes and cross section sets judged necessary to carry out project physics and fuel management tasks.

*Assistance in the preparation of a NERI proposal in this area and liaison with cognizant INEEL staff.

*Recruitment of research team members and organization of their task assignments.

The literature review developed information that helps identify the factors motivating the use of thorium in PWR cores at this time in view of several developments over the past three decades: the change in emphasis from recycle to once-through fueling, the mandate to keep initial U-235 enrichment below the weapons-grade limit of 20 w/o, and the doubling of routinely achievable burnup to the present 50 MWd/kg. This latter point is of particular significance, since once-through thorium fueling is only cost-effective at high burnup -- and, serendipitously, Th-232 rich lattices are easier to drive to high burnup because their reactivity decreases more slowly than U-238 based loadings.

This overview was summarized in a transaction paper submitted and accepted for presentation at the June 1999 meeting of the American Nuclear society. It concludes that, while there are many fine points in which U and the Th rich lattices differ, none appear to compromise project goals in any significant manner.

The code CASMO-4 was obtained through agreement with the code developer, STUDSVIK, and is being installed. The code MOCUP (MCNP + ORIGEN) has been obtained and made operational on a new 600 MHz DEC Alpha computer which has been obtained, and reserved for, project reactor physics and fuel management computations. We have also imported the UTXS (University of Texas) cross section set for MCNP to increase the scope of our benchmarking capabilities.

A standard representation of a 17x17 Westinghouse PWR assembly has been prepared for CASMO parametric studies and a set of ground rules and conventions established in collaboration with the INEEL staff. With their help we are also debugging a unit cell burnup run using MOCUP. Production runs using both CASMO-4 and MOCUP. First order CASMO-4 calculations to date have tended to show that, for the potentially achievable burnup of 70 MWd/kg, thoria based fuels may not be economically superior than the all uranium fuel for assembly designs similar to the existing LWRs. The secondary effects of burnable poisons have to be included in the future. Also, pin and assembly optimizations for the throium fuel performance have yet to be explored. The CASMO-4 runs indicate that the reactivity coefficients of the thoria-urania mixtures are only mildly different from the all- uranium fuel cases and do not pose a challenge to the control systems of the existing reactors.

The INEEL has performed a comparative analysis of thoria-urania fuel using SCALE 4.3. Thorium and uranium dioxides have the same crystal structure and can be substituted in a continuous range of proportions. In the course of this work the proportions of uranium and thorium dioxide were varied from 25 wt % UO_2 – 75 wt % ThO_2 to 35 wt % UO_2 – 65 wt % ThO_2. The uranium is initially 19.5 wt % U-235 and 80.5 wt % U-238, while the thorium is 100 wt % Th-232. This results in an effective U-235 enrichment of 5 to 7 wt % of the total uranium and thorium. Burnup ranged from about 72 megawatt days thermal per kilogram of initial thorium and uranium (MWd/kg) to 100 MWd/kg. (In order to have sufficient reactivity for the 100 MWd/kg case and keep the fraction of U-235 in the uranium below 20 wt %, the UO_2 fraction was increased to 35 wt %.) For purposes of comparison, a conventional 4.5 wt % enriched UO_2 fuel irradiated to 45 MWd/kg and a high burnup 8.0 wt % UO_2 fuel irradiated to 72 MWd/kg are also discussed. In all cases, the cladding was Zircaloy and the pin and assembly dimensions were those of a 17x17 assembly in a Westinghouse-type PWR. The maximum enrichment of the uranium in the mixed ThO_2-UO_2 fuel cases was limited to 19.5 wt % in order to reliably remain below the 20 wt % limit at which restrictions come into force due to weapons proliferation considerations. It is likely that some improvement in reactivity and burnup could be achieved in the ThO_2-UO_2 cases by using a matrix containing more fuel and less water, but such variations were beyond the scope of this investigation.

The calculations were performed using the SCALE 4.3 suite of codes, including BONAMI, NITAWL-II, XSDRNPM, COUPLE and ORIGEN-S. SCALE 4.3 is a modular code system for performing standardized computer analyses for NRC licensing evaluations of LWR fuel. BONAMI performs resonance self-shielding calculations for nuclides that have Bondarenko data associated with their cross-sections. NITAWL-II applies a Nordheim resonance self-shielding correction to nuclides having resolved resonance parameters. XSDRNPM is a general 1-D, discrete-ordinates code for zone weighting of cross section, eigenvalue calculations for neutron multiplication (k-effective), and adjoint calculations for determining

importance functions. COUPLE is the interface module for preparation of cross-section and spectral data for ORIGEN-S. ORIGEN-S is the version of ORIGEN used with SCALE. ORIGEN-S is a general-purpose point-depletion and decay code to calculate isotopic, decay heat, radiation source terms and radioactivity levels. All calculations used a 44-group library, which was collapsed from a 238 group Evaluated Nuclear Data File/ Version B, Number V (ENDF/B-V) library using a spectrum for a PWR fuel pin lattice.

The parameters of the urania and mixed ThO_2-UO_2 fuel cycles analyzed are shown in the following two tables. As mentioned above, four cases were analyzed, a conventional UO_2 case with an equilibrium cycle burnup of 45 MWd/kg, a UO_2 case with an extended burnup of 72 MWd/kg, a ThO_2-UO_2 case with a burnup of 72 MWd./kg, and a ThO_2-UO_2 case with a burnup of about 100MWd/kg. The unique aspects of each of the cases are shown in Table 1. The common characteristics of the fuel assembly and of the fuel cycles are shown in Table 2.

Table 1 Unique Aspects of Each Fuel Cycle Evaluated

Case	Uranium Enrichment	UO2 fraction	Capacity Factor	Cycle (days)	Total Cycle (years)	EOL Burnup (MWd/kg)
1	4.5%	100%	76.0%	520	4.5	45
2	8.0%	100%	90.1%	703	6	72
3	19.5%	25%	90.0%	703	6	72
4	19.5%	35%	82.3%	1068	10	100

The composition of the plutonium produced in the thoria-urania fuel during the fuel cycle of case 3, with a burnup of 72 MWD/kg initial heavy metal, is shown in Figure 1.

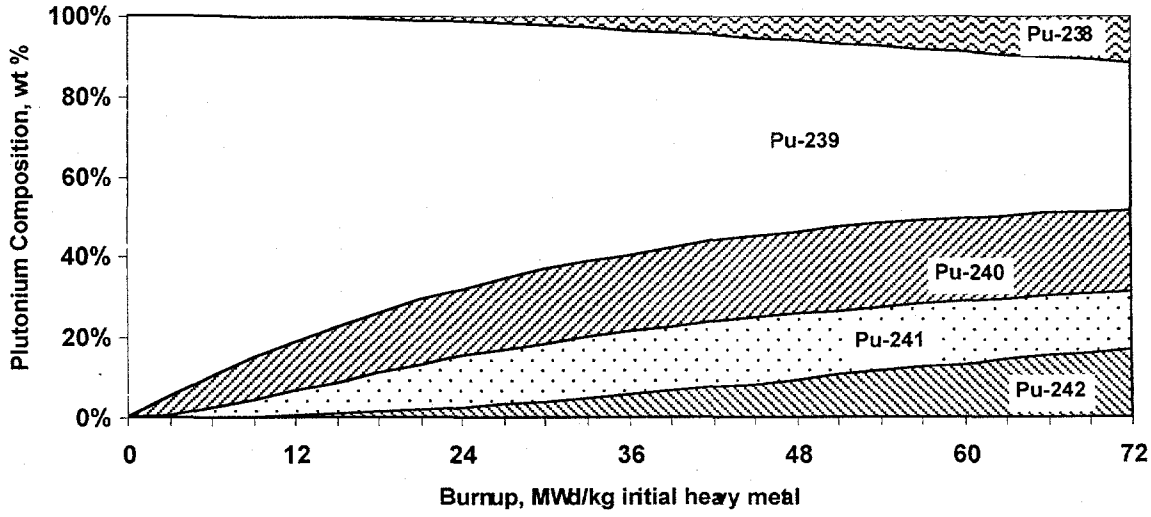

Figure 1 Plutonium composition in a 75% ThO_2-25% UO_2 core burned 6 years to 72 MWd/kg.

A comparison of the total amount of plutonium produced in various types of fuels and of the isotopic composition of that plutonium is shown in Table 3

Table 3 Spontaneous neutron and decay heat

Comparison of Spontaneous Neutron and Heat Production						
	Super grade	Weapons grade	UO2	UO2	ThO2-UO2	ThO2-UO2
Burnup			**45 MWd/kg**	**72 MWd/kg**	**72 MWd/kg**	**100 MWd/kg**
Composition						
Pu-238	0.0%	0.012%	2.4%	4.6%	11.7%	14.7%
Pu-239	98.0%	93.80%	54.2%	52.6%	37.2%	34.4%
Pu-240	2.0%	5.80%	22.3%	21.0%	20.1%	20.0%
Pu-241	0.0%	0.35%	14.9%	15.4%	14.3%	13.2%
Pu-242	0.0%	0.022%	6.2%	6.4%	16.7%	17.7%
Spontaneous Neutron Production						
(n/kg-s)	1.82E+04	5.35E+04	3.72E+05	4.20E+05	7.72E+05	8.65E+05
relative	1.0	2.9	20.4	23.1	42.4	47.5
Decay Heat						
(W/kg)	2.0	2.3	16.9	29.0	68.4	85.0
relative	1.0	1.1	8.5	14.5	34.2	42.5
For a 5.0 kg sphere						
Spontaneous Neutrons						
n/s	9.11E+04	2.67E+05	1.86E+06	2.10E+06	3.86E+06	4.33E+06
Decay Heat						
W	10	11	84	145	342	425
Temperature (C)	79	84	241	308	439	478
(F)	174	184	465	586	823	892

Fuel Performance Modeling

MIT Faculty : John Meyer, Ronald Ballinger and Stephen Shultz
MIT Students: Yun Long
INEEL Participants: Philip MacDonald and Richard McCardell

At MIT, a literature search was conducted to gather insights about the fuel behavior in past reactors and in past experiments. While many well-documented observations exist, they have been obtained largely from conditions that do not represent present day LWR practice. In particular the early reactor experience of Elk River and Indian Point 1 were obtained for mostly thorium fuel with only a small fraction (5%) of highly enriched uranium. In the current studies the uranium fraction is higher than 25%. Furthermore, the cladding material was stainless steel not Zircaloy. On the other hand, the experience from the Light Water Breeder Reactor was for the right clad material but obtained for low uranium content at much lower power ratings and at somewhat lower coolant temperature.

To evaluate fuel performance, FRAPCON-3 and FLA02 computer codes have been used, and demonstrated good agreement on fission gas release for a test UO2 fuel system. The FLA02 code was used to assess the impact of differences in thermal properties alone on gas release. The results show significantly less gas released in the thoria based fuel. More work on thoria modeling needs to be done to extend gas release prediction capability. The FLA01 corrosion and temperature model comparison to FRAPCON-3 do not agree and must be evaluated further and modified to achieve reasonable and consistent results. The results of fuel performance assessment indicate that operating fuel to the six-year level could lead to very severe cladding corrosion. More corrosion resistant materials such as ZIRLO should be evaluated for the extended cycle lifetime.

Waste Behavior

MIT Faculty: Kenneth Czerwinski and Mujid Kazimi
MIT Students: Mike Reynard
INEEL Participant: Philip MacDonald

Thorium based fuel has been used in previous reactors, many of which were light water reactors but some were gas cooled. The behavior of the resulting waste form has been investigated. A review of the literature indicates the repository behavior of Th requires considerable clarification.

From a chemical standpoint, thermodynamic data on basic oxide, hydroxide, and carbonate species is lacking. In addition, the behavior of Th intrinsic and pseudocolloids is not well understood. Work has begun on at MIT estimating the behavior of thorium in the repository near field. A search for existing thermodynamic and speciation data has been undertaken. With some extrapolations, the data was used to assess the solubility of thorium compounds in a repository. Under Yucca Mountain conditions, ThO_2 solubility is expedited to be minimized and U(IV) should oxidize to U(VI). U(VI) is more soluble than Th or U(IV). Dissolution of U can form Th colloids. The ThO_2 may act as a sink for U(VI) and form secondary phases. Additionally, the presence of ThO_2 in the fuel matrix may stabilize the U. Experiments will be conducted in order to obtain Th thermodynamic data for the oxide and mixed uranium species.

From a radiological content standpoint, the long term impact of thorium is to reduce the plutonium content of the fuel but to enhance the presence within that content of Pu-238 and U-232. These isotopes are associated with high specific radioactive decay heat, and may lead to a different thermal load in the repository. Several calculations have been made using ORIGEN to compare the production of radioactive materials with very long half life that may impact the repository performance. In terms of waste management issues, the Th containing fuel is found to create more ^{129}I and ^{234}U due to the higher amount of ^{233}U. However, the U fuel forms more ^{99}Tc, ^{237}Np, and ^{239}Pu and some higher actinides. Hence not only would plutonium production be reduced by the new fuel, but also the long-lived isotopes responsible for the high fraction of the dose to the public, based on the recent viability assessment.

ThO_2-UO_2 WASTE FORM CHARACTERISTICS

The INEEL has performed a review of waste form characteristics for the mixed ThO_2-UO_2 fuel, as compared with conventional UO_2 fuel. The ThO_2-UO_2 fuel is intended for use in a 'once-through' fuel cycle. Thus the resistance of the fuel, as discharged from the reactor, to oxidation, dissolution and erosion in groundwater, is very important.

The Nuclear Waste Policy Act of 1982 and the 1987 amendments to the law were enacted to provide for permanent disposal of high-level radioactive wastes in the United States in a deep geologic repository located at an isolated, arid location. A potential site for the repository is in the volcanic tuff beds at Yucca Mountain, Nevada. This site is a hydrologically unsaturated zone, i.e., damp conditions but limited water flow. The Yucca Mountain ground water contains a number of contaminants which may react with the fuel; the most abundant of which are sodium and silicon. The Yucca Mountain waste containers are expected to resist corrosion failure for at least three thousand, and more likely, tens of thousands of years (about 25% of the waste containers are expected to fail at 100,000 years). The corrosion rates for the Zircaloy fuel cladding are from ten to many thousands of times slower than the waste container material under the extreme acid conditions assumed for Yucca Mountain. Therefore, fuel exposure does not occur until about 30,000 years after the waste packages fail. At about one million years, DOE/RW estimates that about 30% of the spent fuel will be exposed to a humid air environment and may also be exposed to an oxidizing dripping water environment.

To reliably contain the fission products and actinides within the fuel after loss of the waste containers and cladding, the fuel must not undergo significant chemical reactions and resultant changes in crystal structure. Thorium dioxide is the highest oxide of thorium and does not depart significantly from its stoichiometric composition of ThO_2 when exposed to air or water at temperatures up to 2000°K. Spent uranium dioxide fuel fragments, on the other hand, react at a rate of about 1 % per year (i.e. relatively rapidly) with 90°C high-drip-rate water with representative Yucca Mountain contaminants (Finn et. al 1998b). Air-oxidation of UO_2 spent fuel at temperatures near but below 200°C produces U_4O_9 after only several years of exposure. ThO_2-UO_2 mixtures appear to be susceptible to corrosive attack in air or oxygenated water, but significantly less susceptible than UO_2. The available information of this subject is discussed in the following paragraphs.

Dry Air Oxidation. Because thoria is the highest oxidation-state of thorium, the oxidation and physical damage of pressed and sintered ThO_2/UO_2 mixtures is caused entirely by the chemical properties of the urania. Urania has the property of nonstoichiometry, so that a series of urania compositions indicated by the following molecular formulas may exist:

$$UO_{2-x},\ UO_2,\ UO_{2+x},\ U_4O_{9-x},\ U_4O_9,\ U_4O_{9+x},\ U_3O_{8-x},\ U_3O_8,\ UO_3$$

where x is a number less than 1. Urania crystallizes in a fluorite lattice. Upon oxidation, unirradiated UO_2 takes up extra oxygen until U_4O_{9-x} begins to precipitate. Further oxidation then results in U_4O_9 precipitation and so on. However, high burnup UO_2 has displayed enhanced oxidation resistance with a stoichiometry that equilibrates near $UO_{2.4}$ with the structure of U_4O_9. Apparently, it takes only a relatively low level of impurity to stabilize the matrix and change the oxidation sequence of UO_2.

The particular molecular structure is important because the theoretical densities of UO_2, U_4O_9, and U_3O_8 are 10.97, 11.4, and 8.35 Mg/m^3. Therefore, conversion of UO_2 to U_3O_8 results in about a 30% volume increase, grain boundary separation, and powdering of the fuel. If the fuel is contained in typical LWR cladding, oxidation of the UO_2 to U_3O_8 is likely to cause severe splitting of the cladding. However, oxidation to U_4O_9 results in a slight densification (3.4%), some micro cracking, and no structural damage to the fuel.

Cohen and Berman investigated the extent of oxygen solubility in unirradiated UO_2-ThO_2 solid solutions as a function of temperature and composition. They found that after oxidation, the value of x in (Th, U)O_{2+x} increased continuously from 0, at ThO_2, to a maximum value of 0.25 for 50% mixtures. In other words, heavily oxidized high thoria solid solutions, from $(U_{0.5}Th_{0.5})O_{2+x}$ up to pure ThO_2, are in equilibrium with oxygen with urania molecular structures between UO_2 and U_4O_9 and, therefore, swelling, grain boundary separation, and fuel powdering does not occur. Higher urania oxides do not form when there is at least 50% thoria present because the thoria stabilizes the fluorite structure and only one interstitial oxygen can be accommodated per unit cell, only where an appropriate number or uranium ions are adjacent to the space occupied by the oxygen. Thomas et. al (1993) report a somewhat similar stabilization due to the fission products in high burnup spent LWR UO_2 fuel oxidized at temperatures below 200°C in air.

Moist Air and Water Oxidation. The oxidation of UO_2 in water requires the presence of dissolved oxygen, but "proceeds in a completely different manner from oxidation in air or gaseous oxygen" (Belle 1961). The major reaction product upon contact with pure water is $UO_3 \bullet 0.8H_2O$, a hydrate called dehydrated schoepite. Markowitz and Clayton (1970) investigated the corrosion behavior of a group of nuclear fuel oxides, including two compositions (20 and 50% UO_2) of urania-thoria, in high temperature (360°C), alkaline (pH 10) flowing water. The ThO_2 samples displayed excellent corrosion resistance. The weight gains of the urania-thoria material exposed to water oxygenated to about 100 ppm "were much larger than for any of the other materials tested, in any of the media, even those which failed and fell apart; yet all the specimens retained their mechanical integrity". The urania-thoria fuel remained mechanically intact in spite of the growth of an oxidized surface phase thought to be of the M_4O_9 type. $UO_3 \bullet 0.8H_2O$ was apparently not found in or on the Markowitz and Clayton specimens.

Finn et. al (1998b) has reported on the results of tests with spent uranium dioxide fuel fragments which were exposed for a number of years to 90°C drip-water containing representative Yucca Mountain contaminants. As mentioned above, this material reacted (dissolved) at a rate of about one percent per year in the high-drip-rate tests. The major alteration products were Na-boltwoodite, $Na[(UO_2)(SiO_3OH)]\bullet H_2O$, which is formed from the sodium and silicon in the simulated groundwater, and dehydrated schoepite. Although there is no information available on the behavior of mixed urania-thoria fuels exposed to similar conditions, the results of Cohen and Berman (1966) and Makowitz and Clayton (1970) would suggest far less structural damage than observed by Finn et. al with UO_2.

Thermal Design and Safety

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A Review of the thermal hydraulic implications of the thorium based fuels led to the conclusion that in uniformly mixed fuels in typical LWR lattices, the thermal behavior may be essentially similar to the usual U alone fuel. Some areas were identified where an effort needs to be exerted in the course of this investigation. These involve the effect of the higher decay heat because of both the expected increased burnup and the slightly higher decay heat of U233 in comparison with that of U235. The presence of U232, which is characterized by intense gamma ray generation may have to be considered. The higher gamma energy may lead to deposition of some of the heat in the peripheral areas thus lowering the peak to average decay heat generation. However, this seems to be a small effect.

In the area of thermal hydraulics, the key differences affecting safety involve higher decay heat levels, moderately higher thermal conductivity of thoria-based fuels, and differences in power peaking. Thoria-urania fuel was found to exhibit better performance during LOCA due to its higher thermal conductivity and lower specific heat capacity and density in comparison to the case of UO_2 fuel for the same conditions. The decay heat data from the ORIGEN2 calculations indicate that thoria-urania fuels generate more decay heat in the long term (roughly a month after shutdown), but the differences are not so substantial to significantly affect capabilities of heat removal systems.

Note that in the competing other concept for thorium fuels , the seed-blanket concept, considerably more complicated thermal and safety considerations are encountered. This is due to the much higher power density in the seed, to the use of metallic fuel in the seed and due to the need to manage the flow in a way that recognizes the buildup of the power fraction in the blanket over the lifetime of the fuel in the core.

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Low Cost, Proliferation Resistant, Uranium-Thorium Dioxide Fuels for Light Water Reactors

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ABSTRACT

Our objective is to develop a fuel for the existing light water reactors (LWRs) that (a) is less expensive to fabricate than the current uranium-dioxide (UO_2) fuel, (b) allows longer refueling cycles and higher sustainable plant capacity factors, (c) is very resistant to nuclear weapons-material proliferation, (d) results in a more stable and insoluble waste form, and, (e) generates less high level waste. This paper presents the results of our initial investigation of a LWR fuel consisting of mixed thorium dioxide and uranium dioxide (ThO_2-UO_2).

Our calculations using the SCALE 4.3 code system indicate that the mixed ThO_2-UO_2 fuel, with about 5 wt % of the total heavy metal U-235, could be burned to 72 MWd/kg (megawatt days per kilogram) using 25 wt % UO_2 (using uranium below 20% enrichment) and the balance ThO_2. The ThO_2-UO_2 cores can also be burned to about 100 MWd/kg using 35 wt % UO_2 and 65 % ThO_2 with an initial enrichment of about 7 wt % of the total heavy metal fissile material. Economic analyses indicate that the ThO_2-UO_2 fuel will require less separative work and less total heavy metal (thorium and uranium) feedstock. Even if the cost of fabricating the mixed ThO_2-UO_2 fuel is $100/kg greater, the cost of the ThO_2-UO_2 fuel is 13 % to 25 % less than that of the fuels using uranium only.

Because ThO_2-UO_2 fuel will have better thermal properties than pure UO_2, and will retain within the fuel more of the fission products, especially the gasses, ThO_2-UO_2 fuel can probably be successfully operated to higher burnups than UO_2 fuel. This will allow for longer refueling cycles and better plant capacity factors.

The uranium in our calculations remained below 20 wt % total fissile fraction throughout the cycle, making it unusable for weapons. Total plutonium production per MWd was a factor of 4.5 less in the ThO_2-UO_2 fuel than in the conventional fuel. Pu-239 production per MWd was a factor of 6.5 less in the ThO_2-UO_2 fuel than in the conventional fuel. The plutonium produced was high in Pu-238, leading to a decay heat 5 times greater than that from plutonium derived from conventional fuel and 40 times greater than weapons grade plutonium. This level of decay heat will require active cooling of any crude weapon, lest the components surrounding the plutonium be melted. Spontaneous neutron production for plutonium from ThO_2-UO_2 fuel was 2.3 times greater than that from conventional fuel and 15 times greater than that from weapons grade plutonium. High spontaneous neutron production drastically limits the probable yield of a crude weapon.

Because ThO_2 is the highest oxide of thorium, while UO_2 can be oxidized further to U_3O_8 or UO_3, ThO_2- UO_2 fuel is stoiciometrically more stable than UO_2 and appears to be a superior waste form if the spent fuel is ever to be exposed to air or oxygenated water.

And, finally, use of higher burnup fuel will result in proportionally fewer spent fuel bundles to handle, store, ship, and permanently dispose of.

INTRODUCTION

Trends in the world's population and energy use during the past century show dramatic and relatively parallel increases in both. These trends are expected to continue in the near future (at least the next 20 years), and the total world energy consumption in 2015 will be about 54 % higher than it is today, led by growing demand in Asia (DOE-EIA 1997). The demand for electricity is expected to increase more rapidly than the demand for other forms of energy throughout the world and nearly double by 2015. Coal will be used to generate much of that electricity in the developing countries. In the industrialized world, there are also dramatic structural changes underway in the electric power industry to enhance competition in the generation segment of the business. This, along with ample natural gas supplies and relatively low gas prices, has made natural gas the preferred fuel for many power producers in the United States and elsewhere. These developments (increasing energy demand and increasing use of natural gas and coal) are expected to *increase* the amount of carbon emitted to the atmosphere from the world's electrical power plants by about 70 % over the next 20 years (DOE-EIA 1997).

Nuclear energy is the only fully developed technology able to supply large amounts of electricity without generation of greenhouse gases, and therefore should be a key element in the strategy to control greenhouse gas emissions. However, several problems cloud the future of nuclear power in the United States and need to be addressed for nuclear power to be a preferred electric power generation option. President Clinton's Committee of Advisors on Science and Technology (PCAST 1997) was recently asked to look at these issues and concluded that

> *"Fission's future expandability is in doubt in the United States and many other regions of the world because of concerns about high costs, reactor-accident risks, radioactive-waste management, and potential links to the spread of nuclear weapons. We believe that the potential benefits of an expanded contribution from fission in helping address the carbon dioxide challenge warrant ...*[finding] *...out whether and how improved technology could alleviate the concerns that cloud this energy option's future."*

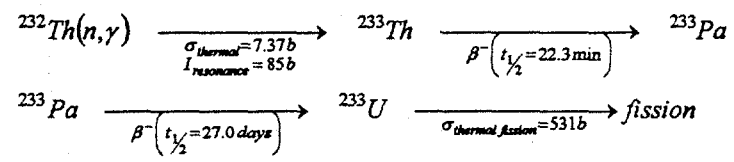
Our objective is to develop a fuel for the existing light water reactors (LWRs) that

- is less expensive to fabricate than the UO_2 fuel,
- allows longer refueling cycles and higher sustainable plant capacity factors,
- is very resistant to nuclear weapons-material proliferation,
- results in a more stable and insoluble waste form, and,
- generates less high level waste.

Urania-thoria cores offer a real promise of improved performance over all-uranium LWR cores, which seem to be approaching their economic limits on burnup and have stymied attempts to press on further into the long refueling cycle, ultrahigh-burnup operating regime. These potential benefits are discussed in the following paragraphs.

Fuel Costs. Extended burnup reactor cores using conventional UO_2 fuel require high U-235 enrichments and significant quantities of burnable poisons for reactivity control which significantly

increases costs. A recent generic study by the Massachusetts Institute of Technology (MIT) investigated the economics of longer refueling cycles (Handwerk et al. 1998, McMahon et al. 1997). The MIT study indicates that, for the pressurized water reactor (PWR) and boiling water reactor (BWR) studied, fuel cycle economics favor cycle lengths of about 20 and 22 months, respectively, with peak fuel burnup values only about 20 to 30 % above current limits. This result is due, in part, because the ultimate cost benefit strongly depends on the fuel carrying charges and the enrichment, both of which increase as the fuel cycle length increases. However, the reactivity in a ThO_2-UO_2-fueled reactor remains more constant during long irradiations than in a UO_2 core because of the high conversion ratio of the thorium (see sketch below). Therefore, a ThO_2-UO_2 core designed for long cycles and high burnup could require less enrichment, less separative work, and less total heavy metal feedstock than a traditional UO_2 core.



Production of ^{233}U from ^{232}Th

Improved Nuclear Power Plant Economics. An important objective of our study is to reduce plant-operating costs and the price of electricity produced by nuclear power plants. Most (80 %) of the U.S. plants are currently operating with 18-month or shorter refueling cycles. Only about 12 plants are operating with 24-month cycles, primarily with mostly 2-cycle fuel to stay within the current burnup limits. With improved fuel, many of the U.S. plants could move to 24- to 36-month refueling cycles (the refueling cycles at some plants will be limited by the need to inspect or repair other equipment such as steam generators). An improvement to 24-month cycles is worth about 2.5 % in plant capacity and an improvement to 36-month cycles would increase plant capacity factors by about 5 %. Having the same plants generate 5 % more electricity would save U.S. utilities and thus ratepayers about $1 billion per year (at a production cost of 2¢ per kilowatt-hour)[1] and help make nuclear energy more competitive. Because most worker exposure and low level waste generation at commercial nuclear plants occurs during refueling, longer refueling cycles will also reduce worker exposures to radiation and the amount of low-level waste generated.

Proliferation. LWRs generate plutonium from the U-238 capturing neutrons. This plutonium, while far from ideal for weapons applications, may be useful to a potential proliferant. Today, worldwide, there are about 270 tons of separated civilian plutonium, primarily in France, Germany, the United Kingdom, Russia, and Japan. In addition, about 700 tons of plutonium is contained in spent LWR fuel worldwide. While it takes only about 6 kilograms of pure Pu-239 to build a weapon (Mark 1992), uranium-based commercial LWR fuel contains other plutonium isotopes that make it much less attractive for nuclear weapons proliferation.

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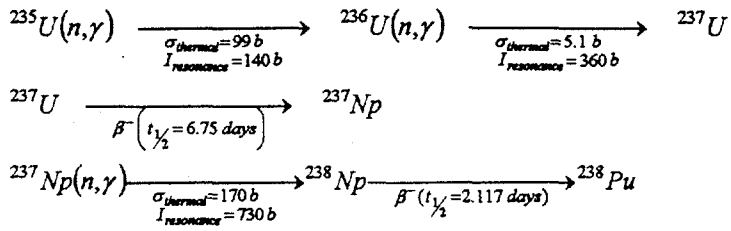
[1] The average production cost of electricity from the nuclear and coal fired nuclear plants in the US is slightly under 2c/kw-hr (Fertel 1998, Yang 1997), the total cost of electricity from our current nuclear plants is close to 7c/kw-hr (Sweet 1997), and the estimated total cost of electricity from new Advanced LWRs is 4.5 to 5c/kw-hr (Booras 1998). The future wholesale cost of electricity in the US in a deregulated environment is uncertain.

At very high burnups, LWR fuel contains significant quantities of Pu-238. Of all the plutonium isotopes, Pu-238 best serves to reduce the weapons usability of plutonium because of a very high spontaneous neutron fraction and a very high thermal output. Pu-238 is primarily produced in the three-step neutron absorption in U-235, shown below.



Production of ^{238}Pu from ^{235}U

The fraction of Pu-238 increases approximately with the square of the fuel burnup. (The Pu-238 increases approximately with the cube of burnup when the quantity of U-235 is constant, with a diminishing supply of U-235, the rate of increase of Pu-238 is approximately to the square power of burnup.) The plutonium isotopes in high burnup LWR fuel release spontaneous neutrons that significantly decrease the probable yield of a nuclear weapon. They also release enough heat to melt and render ineffective the explosives commonly used in nuclear weapons (unless these weapons are actively cooled, which is very difficult and has never been done). High burnup ThO_2-UO_2 fuels are even more proliferation resistant than high burnup UO_2 fuel because they contain less U-238 and therefore less Pu-239 and relatively more Pu-238.

High-Level Waste Form. Because ThO_2 is the highest oxide of thorium, while UO_2 can be oxidized further to U_3O_8 or UO_3, ThO_2-UO_2 fuel appears to be a better waste form than conventional UO_2 fuel.

Spent Fuel Minimization. The facility operating portion of the planned system to dispose of the nation's spent nuclear fuel and high-level waste has been estimated to be about $13.6 billion over about 40 years (DOE/RW 1995), or $32,000 per BWR fuel bundle and $60,500 per PWR fuel bundle. Approximately 4,000 BWR and 3,400 PWR fuel assemblies are discharged each year in the United States (DOE-EIA 1996). If the equilibrium cycle discharge burnups in the United States could be increased to 75 MWd/kg, for example, and the 105 plants currently operating continue to operate, the Federal Government could save more than $100 million per year, because fewer spent fuel bundles will need temporary storage, handling, transportation, and permanent disposal.

Other considerations. It should also be mentioned that a large body of work on the thorium cycles was performed in support of the Light Water Breeder Reactor (LWBR) prototype at Shippingport, Pennsylvania (Belle and Berman 1984). The Shippingport reactor used movable, hexagonal fuel in a seed and blanket configuration to achieve breeding in a LWR. Chemical reprocessing of the spent fuel was assumed to recover the bred ^{233}U. The Shippingport reactor also used uranium highly enriched in ^{235}U. The fuel discussed in this paper will be directly usable in existing LWRs with a once-through fuel cycle, without reprocessing. The ^{233}U will be bred and fissioned *in situ*.

Furthermore, the fuel will be taken to much higher burnups than previously contemplated in earlier work.

Further work on a thorium cycle reactor has been pursed by Radkowsky and Galperin (1998). They have proposed use of an oxide blanket and metal fuel seed core. The focus in this project is on a homogenous urania-thoria assembly design. This will avoid the geometrical and chemical stability issues during irradiation of metal fuels (for discussions of metal fuel performance see Hofman and Walters 1994 and Eyre et al. 1993) and it will avoid the need for post-irradiation seed fuel processing before final disposal. It will also avoid the need for frequent shutdowns and replacement or movement of the seed rods. It is our opinion that the neutronic conversion ratio advantages of the seed and blanket design are more than offset by its more complicated fuel manufacturing and management needs.

CORE NEUTONICS DESIGN

Method of Calculation

Thorium and uranium dioxides have the same crystal structure and can be substituted in a continuous range of proportions. In the course of this work the proportions of uranium and thorium dioxide were varied from 25 wt % UO_2 – 75 wt % ThO_2 to 35 wt % UO_2 – 65 wt % ThO_2. The uranium is initially 19.5 wt % U-235 and 80.5 wt % U-238, while the thorium is 100 wt % Th-232. This results in an effective U-235 enrichment of 5 to 7 wt % of the total uranium and thorium. Burnup ranged from about 72 megawatt days thermal per kilogram of initial thorium and uranium (MWd/kg) to 100 MWd/kg. (In order to have sufficient reactivity for the 100 MWd/kg case and keep the fraction of U-235 in the uranium below 20 wt %, the UO_2 fraction was increased to 35 wt %.) For purposes of comparison, a conventional 4.5 wt % enriched UO_2 fuel irradiated to 45 MWd/kg and a high burnup 8.0 wt % UO_2 fuel irradiated to 72 MWd/kg are also discussed. In all cases, the cladding was Zircalloy and the pin and assembly dimensions were those of a 17x17 assembly in a Westinghouse-type PWR. The maximum enrichment of the uranium in the mixed ThO_2-UO_2 fuel cases was limited to 19.5 wt % in order to reliably remain below the 20 wt % limit at which restrictions come into force due to weapons proliferation considerations. It is likely that some improvement in reactivity and burnup could be achieved in the ThO_2-UO_2 cases by using a matrix containing more fuel and less water, but such variations were beyond the scope of this investigation.

The calculations were performed using the SCALE 4.3 suite of codes, including BONAMI, NITAWL-II, XSDRNPM, COUPLE and ORIGEN-S (USNRC 1995). SCALE 4.3 is a modular code system for performing standardized computer analyses for NRC licensing evaluations of LWR fuel. BONAMI performs resonance self-shielding calculations for nuclides that have Bondarenko data associated with their cross-sections. NITAWL-II applies a Nordheim resonance self-shielding correction to nuclides having resolved resonance parameters. XSDRNPM is a general 1-D, discrete-ordinates code for zone weighting of cross section, eigenvalue calculations for neutron multiplication (k-effective), and adjoint calculations for determining importance functions. COUPLE is the interface module for preparation of cross-section and spectral data for ORIGEN-S. ORIGEN-S is the version of ORIGEN used with SCALE. ORIGEN-S is a general-purpose point-depletion and decay code to calculate isotopic, decay heat, radiation source terms and radioactivity levels. All calculations used a 44-group library, which was collapsed from a 238

group Evaluated Nuclear Data File/ Version B, Number V (ENDF/B-V) library using a spectrum for a PWR fuel pin lattice (McLane 1988).

The parameters of the urania and mixed ThO_2-UO_2 fuel cycles analyzed are shown in the following two tables. As mentioned above, four cases were analyzed, a conventional UO_2 case with an equilibrium cycle burnup of 45 MWd/kg, a UO_2 case with an extended burnup of 72 MWd/kg, a ThO_2-UO_2 case with a burnup of 72 MWd./kg, and a ThO_2-UO_2 case with a burnup of about 100MWd/kg. The unique aspects of each of the cases are shown in Table 1. The common characteristics of the fuel assembly and of the fuel cycles are shown in Table 2.

Table 1 Unique Aspects of Each Fuel Cycle Evaluated

Case	Uranium Enrichment	UO2 fraction	Capacity Factor	Cycle (days)	Total Cycle (years)	EOL Burnup (MWd/kg)
1	4.5%	100%	76.0%	520	4.5	45
2	8.0%	100%	90.1%	703	6	72
3	19.5%	25%	90.0%	703	6	72
4	19.5%	35%	82.3%	1068	10	100

Table 2 Common Characteristics of the Fuel Assemblies and Fuel Cycles Evaluated

Fuel Assembly Parameters		
Pins	17 x 17	
Pitch	1.27	cm
Pellet Diameter	0.823	cm
Fuel Density	94.5	%
Clad Outer Diameter	0.9424	cm
Fuel Pins per Assembly	264	
Active Fuel Length	363	cm
Fuel inventory	464.5	kg ihm/assembly
Cladding	Zircalloy	
Cladding thickness	0.0597	cm
Assembly dimension	21.7 x 21.7	cm
Assembly pitch	21.8	cm
non-fuel positions	25	
Non-fuel inner diameter	1.2243	cm
Non-fuel outer diameter	1.1430	cm
Specific Power	37.935	kWth/kg ihm
Fuel Temperature (average)	811	K
Cladding Temperature (ave.)	750	K
Coolant Density	0.644	g/cm3
Coolant Temperature (bulk)	605	K
Refueling Outage	28	days
Thermal Efficiency	33.7%	

Reactivity Changes During the Irradiation Cycles

The changes in the assembly reactivity for the ThO_2-UO_2 cases are shown in Figure 1. The neutron multiplication factor for a large array of assemblies, the assembly k infinity values, were computed using SCALE 4.3 assuming an infinite array of fuel assemblies. There was no burnable poison in the fuel and no boron in the coolant in the ThO_2-UO_2 core calculations. The fuel,

cladding, and coolant are at the temperatures, and include the absorbing materials, indicated in Table 2. Each of the batches is assumed to be at the same average power level. The core loading was divided into three equal batches in all the cases analyzed. One third of the core, that is one batch, was replaced at each refueling outage. These reactivity values should be regarded as preliminary. More detailed calculations are now being performed for pin-cell and assembly configurations using MOCUP (a combination of MCNP-4b and ORIGEN-2) (Moore et. al 1995) and using CASMO-4.

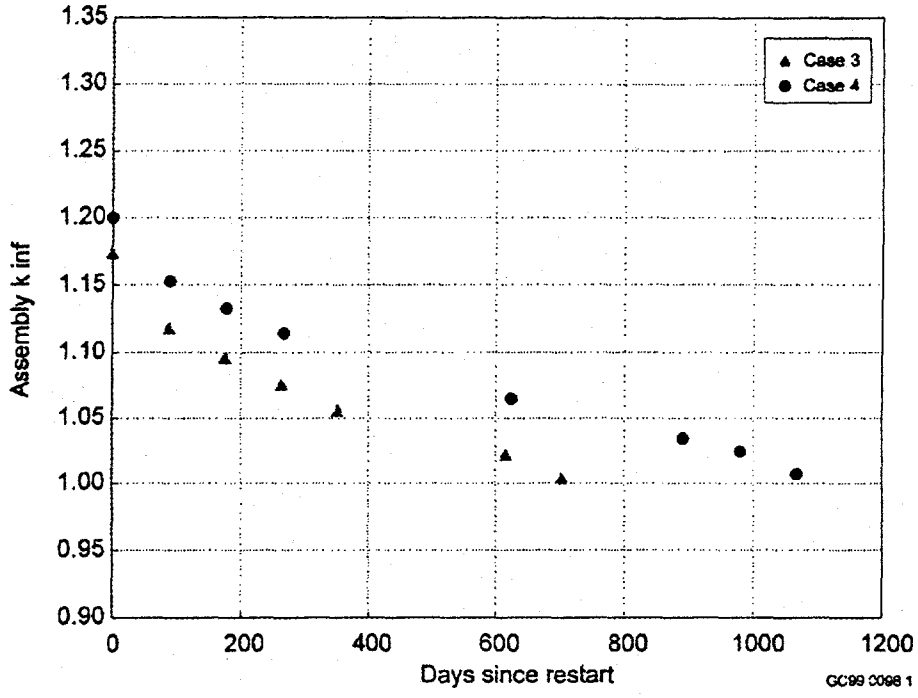


Figure 1 Core average reactivity versus days since restart for mixed ThO_2-UO_2 cores.

It is important to note that fuels containing thorium produce U-233 via the Pa-233 intermediate, which has a 27.0-day half-life. The intermediate for Pu-239 breeding is Np-239, with a 2.355 day half-life. Thus one could expect a small increase in reactivity after a refueling outage with ThO_2-UO_2 fuel, due to the buildup of U-233 from the equilibrium Pa-233 inventory.

The uranium contents of the fuel in Cases 1 and 3 are shown graphically in Figures 2 and 3. In these plots of the changing isotopic composition, each isotope's fraction is shown as the distance between the lines. Therefore, in Figure 2 at 20 MWd/kg, the uranium is 97 wt % U-238, 0.3 wt % U-236 and 2.7 wt % U-235. This type of graphical representation will be used throughout this paper.

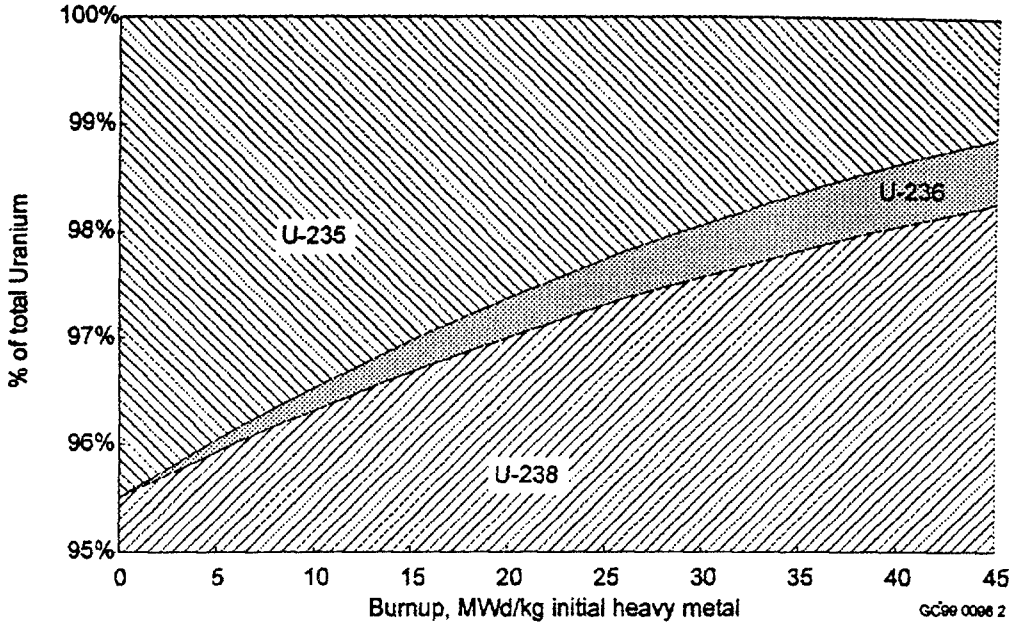


Figure 2 Uranium composition in a UO_2 core burned 4.5 Years to 45 MWd/kg.

The uranium content of the ThO_2-UO_2 fuel is shown in Figure 3. Note that the U-233 increases while the U-235 is being consumed. At discharge, after 72 MWd/kg, the fissile content is still 7.7 wt % of the total uranium and 1.6 wt % of the initial heavy metal in the fuel, but most of the fissile material is now U-233. Also, the U-236 content is 3.4 wt %, compared with 0.61 wt % in the case for conventional fuel at 45 MWd/kg, leading to high Pu-238 production in any subsequent use of the uranium, for the reasons cited earlier. U-236, which has a half-life of 23 million years, is the limiting step in the production of Pu-238.

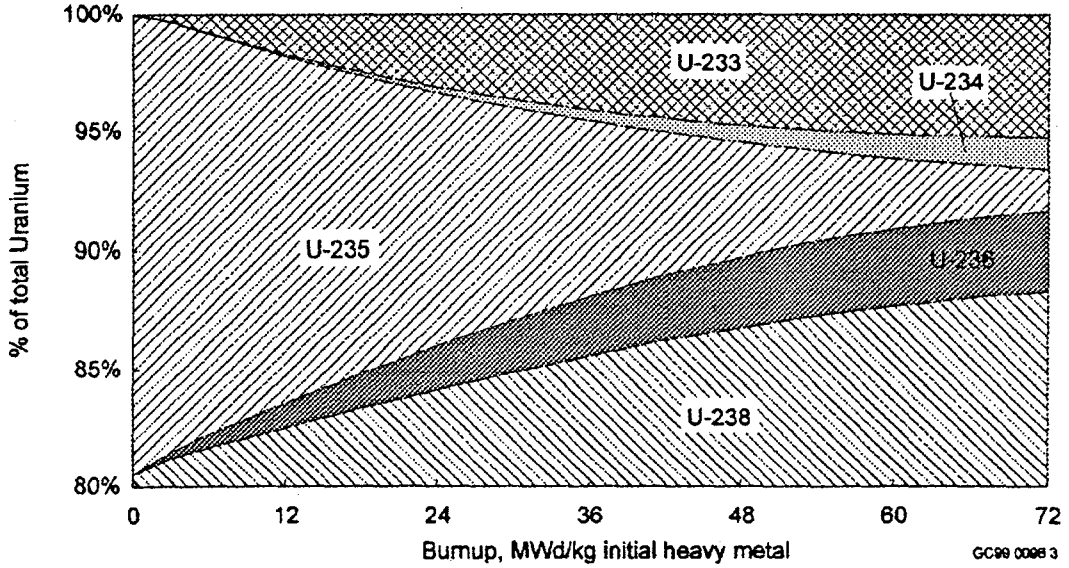


Figure 3 Uranium composition in a 75%ThO_2-25%UO_2 core burned 6 years to 72 MWd/kg.

Changes in Plutonium Composition During Irradiation

The isotopic compositions of the plutonium in the four cases identified in Table 1 are shown in the following figures. Note that the Pu-238 content is increasing at somewhat less than the cube of the burnup (at about the square of burnup). This is to be expected since the source of the Pu-238, i.e. the U-235, is being fissioned and replaced with U-233. Nevertheless, the Pu-238 is about 14 % of the total plutonium at the end of life in the 100 MWd/kg ThO_2-UO_2 cycle, compared to about 2 % in the conventional uranium dioxide fuel.

The plutonium content of the fuel in Case 1 is shown in Figure 4. The Pu-238 fraction is about 1 % at 30 MWd/kg, increasing to 2.3 wt % at 45 MWd/kg. This is to be expected because of the increasing inventory of the U-236 and Np-237 targets for the three-step transmutation.

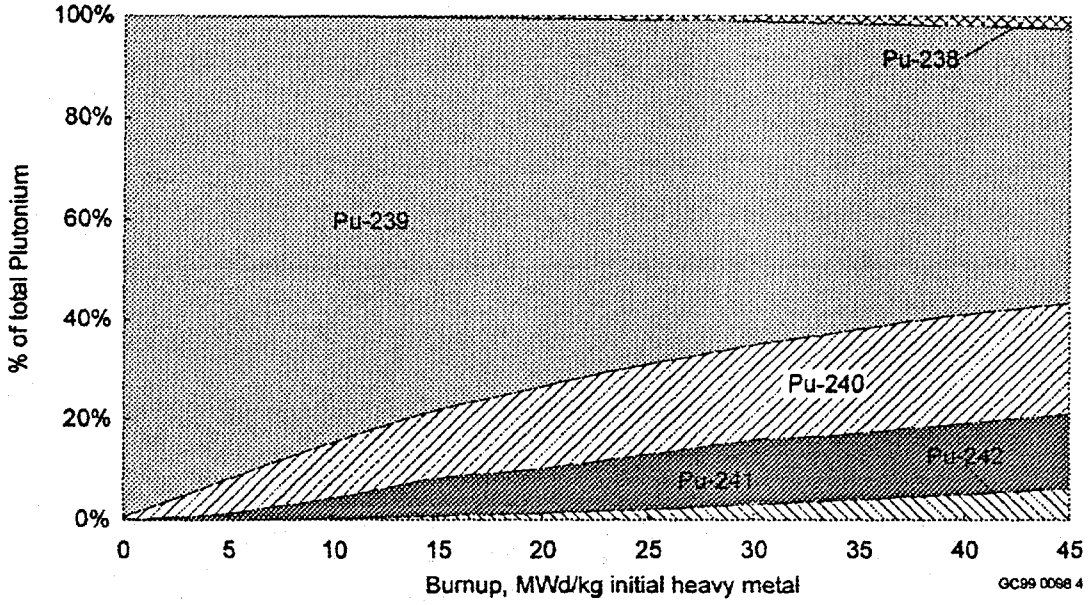


Figure 4 Plutonium composition in a UO_2 core burned 4.5 years to 45 MWd/kg.

In Case 2, a fuel cycle using 8 wt % enriched uranium dioxide was irradiated to 72 MWd/kg in three batches in a 6-year cycle. While this burnup would not be permitted today because of NRC burnup limitations, such an extended fuel cycle is being evaluated as a means of making nuclear power more economically competitive. The plutonium content and composition in the extended burnup UO_2 fuel are shown in Figure 5. Note that 4.4 wt % of the material is Pu-238, nearly double the 2.3 wt % in the 45 MWd/kg base case. However, the total plutonium content is only16 grams per kg of initial heavy metal in the fuel, versus 11 grams for the 45 MWd/kg base case.

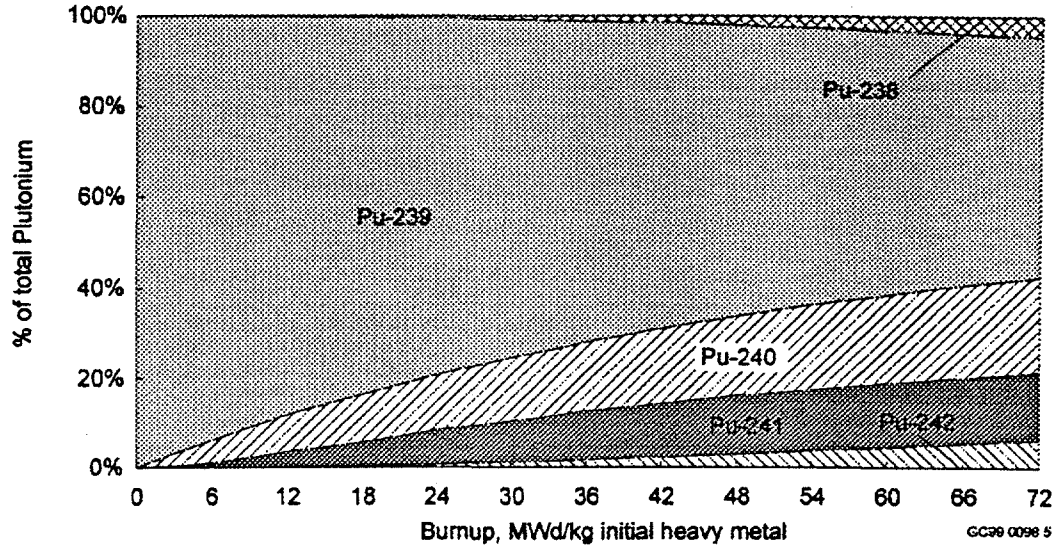


Figure 5 Plutonium composition in a UO_2 core burned 6 years to 72MWd/kg.

The plutonium content of the mixed thorium-uranium fuel during the fuel cycle is shown in Figure 6. Note the significant decrease in Pu-239 and the increase in the other plutonium isotopes.

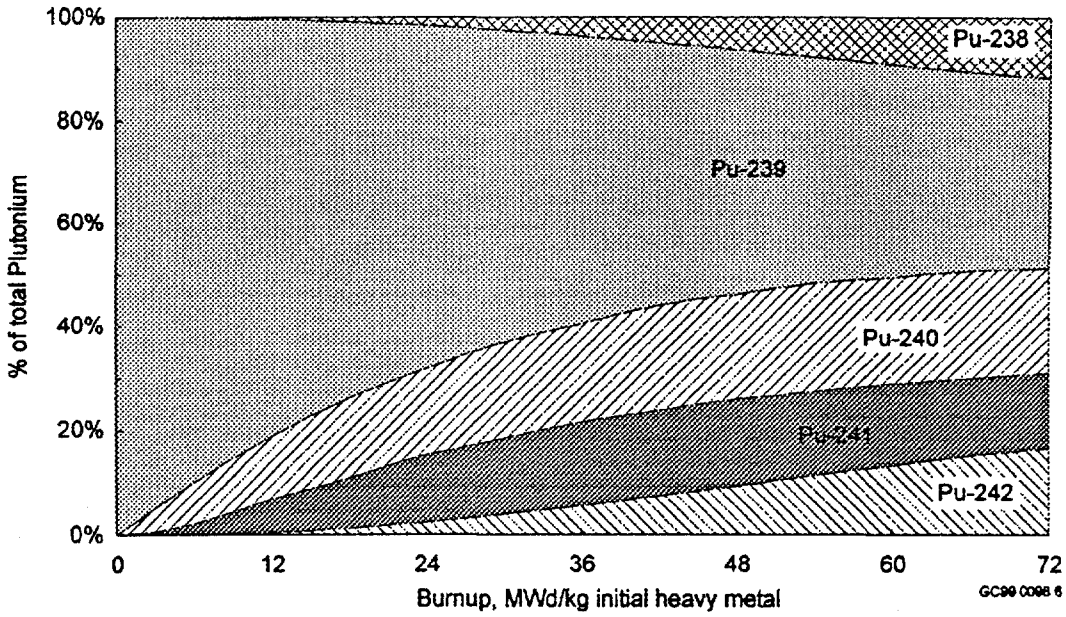


Figure 6 Plutonium composition in a 75% ThO_2-25% UO_2 core burned 6 years to 72 MWd/kg.

The plutonium content of the ThO_2-UO_2 fuel after an irradiation to 100 MWd/kg is shown in Figure 7. The Pu-238 content has become nearly 14 wt %. The high decay heat and spontaneous neutron production of the Pu-238 have strong ramifications for the handling of the plutonium, should it ever be separated from the fuel. These ramifications are discussed in the section entitled "Proliferation Considerations" below.

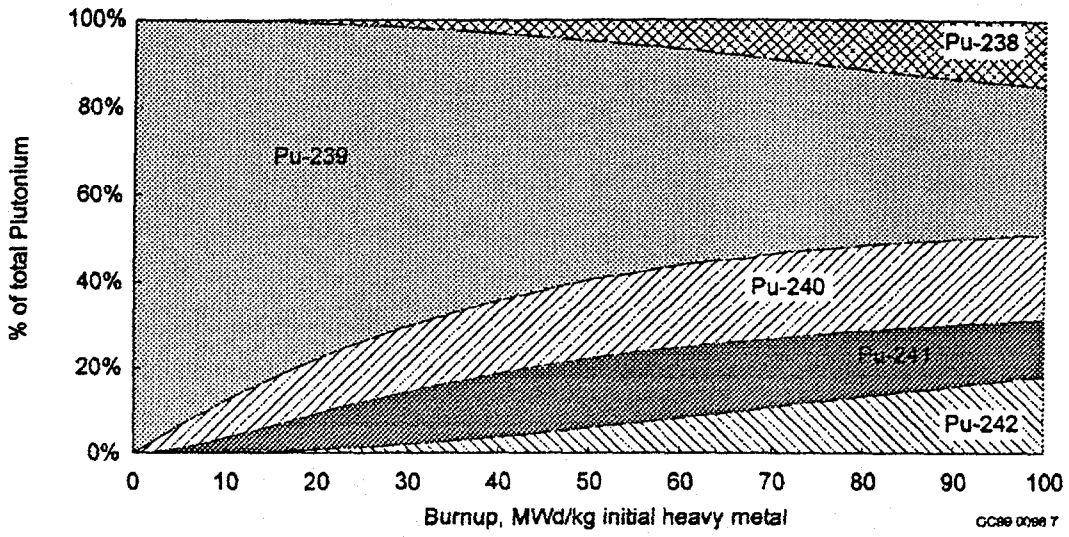


Figure 7 Plutonium composition in a 65% ThO_2-35% UO_2 core burned 10 years to 100 MWd/kg.

Economic Comparisons

The results of the economic comparisons among the two uranium dioxide fuels and the two mixed ThO_2-UO_2 fuels are shown in Tables 3 and 4. These calculations show a 3 to 33 % advantage for the mixed ThO_2-UO_2 fuel compared with the two UO_2 fuels, depending on cycle length and materials cost.

The specific power for each of the fuels was held constant at 37.94 MWth/kg of initial heavy metal. The cycle parameters for Cases 2 and 3, the ultra-high burnup urania fuel and the mixed ThO_2–UO_2 fuel are identical. The feed to the cycle is assumed to be natural uranium and thorium. The tails assay of depleted uranium leaving the enrichment plant is assumed to be 0.3 wt %, reflecting the low current prices of U_3O_8. The use of 0.2 wt % tails assay would result in a greater advantage for the mixed ThO_2-UO_2 fuel because of the somewhat lower enrichment per MWd required. Note that the use of thorium results in a large decrease in the total amount of uranium and thorium mined per MWD of energy produced.
The prices of uranium and thorium are approximately the current market prices; thorium is now co-produced with the lanthanide metals for only a few specialty uses. If thorium were used in large quantities in the nuclear fuel cycle, the price might be lower; its natural abundance is about three times that of uranium, but it does not concentrate in geological formations as well as uranium. Because of the uncertainties in the costs for fabricating ThO_2-UO_2 fuel, we have assumed in Table 3 that the fabrication cost of the mixed ThO_2-UO_2 fuel is $300/kg, 50 % higher than that of the current LWR fuel. We have included the cost of burnable poison in the UO_2 fuels. However, we have not included any waste disposal adjustment for the smaller volume of spent fuel resulting from an extended irradiation. An extended burnup to 72 MWd/kg mixed ThO_2-UO_2 fuel case would result in about 50% less spent fuel volume per unit of electricity generated, when compared with the 45 MWd/kg UO_2 case.

The interest charges reflect the fact that the raw materials, separative work and fabrication charges much be paid before the fuel is loaded into the reactor, while electricity is only generated a number of years later. The interest charges are calculated by multiplying the interest rate, the total cost of

the fuel per kilogram and one half the total cycle length. Note that the interest paid on the initial investment in the fuel is a significant penalty for extended burn fuels as compared with conventional fuels. Avoiding the need for as many refueling outages in the longer cycles may counteract that penalty, but cost benefit of fewer refueling outages and the additional plant capacity factor is not included in this analysis.

Table 3 Cost comparison using current prices.

Fuel Cost Comparison Using Current Prices					
	Current PWR Uranium Dioxide Fuel	Ultra High Burnup Uranium Dioxide Fuel	Mixed Uranium/ Thorium Dioxide Fuel 72 MWd/kg	Mixed Uranium/ Thorium Dioxide Fuel 100 MWd/kg	units
Specific Power	37.935	37.935	37.935	37.935	kWth/kg
Total cycle length	4.5	6	6	10	years
Effective Full Power Days	1186	1897	1897	2635	efpd
Burnup	45.0	72.0	72.0	100.0	MWd/kg HM
Feed U-235 content	0.72%	0.72%	0.72%	0.72%	atom %
Product U-235 enrichment	4.50%	8.00%	19.70%	19.70%	atom %
Tails U-235 content	0.30%	0.30%	0.30%	0.30%	atom %
Feed U-235 content	0.711%	0.711%	0.711%	0.711%	wt %
Product U-235 enrichment	4.446%	7.907%	19.500%	19.500%	wt %
Tails U-235 content	0.300%	0.300%	0.300%	0.300%	wt %
Fraction uranium in fuel	1.000	1.000	0.250	0.350	
Separative work	6.125	12.994	9.310	13.034	kg-SWU/kg fuel
Burnable poison	0.00	0.10	0.00	0.00	kg/kg fuel
Natural uranium	10.091	18.517	11.684	16.358	kg/kg fuel
Natural thorium	0.000	0.000	0.750	0.650	kg/kg fuel
Total heavy metal mined	10.091	18.517	12.434	17.008	kg/kg fuel
	0.224	0.257	0.173	0.170	kg/MWd
Rates					
Interest rate	8.0%	8.0%	8.0%	8.0%	per year
Natural uranium	$ 25.00	$ 25.00	$ 25.00	$ 25.00	/kg
	$ 13.40	$ 13.40	$ 13.40	$ 13.40	/lb U3O8
Natural thorium	$ 88.50	$ 88.50	$ 88.50	$ 88.50	/kg
Gadolinium	$ 115.00	$ 115.00	$ 115.00	$ 115.00	/kg
Conversion U3O8 to UF6	$ 5.00	$ 5.00	$ 5.00	$ 5.00	/kg enr U
Separative work	$ 75.00	$ 75.00	$ 75.00	$ 75.00	/kg-SWU
Costs					
Raw materials	$ 252.28	$ 474.43	$ 358.48	$ 466.47	/kg fuel
Separative work	$ 459.34	$ 974.58	$ 698.27	$ 977.57	/kg fuel
Conversion	$ 5.00	$ 5.00	$ 1.25	$ 1.75	/kg fuel
Fabrication	$ 200.00	$ 250.00	$ 300.00	$ 300.00	/kg fuel
Total cost	$ 916.62	$ 1,692.51	$ 1,358.00	$ 1,745.80	/kg fuel
Interest during use	$ 164.99	$ 406.20	$ 325.92	$ 698.32	/kg fuel
Total fuel cost	$ 24.04	$ 29.16	$ 23.40	$ 24.45	/MWth-day
	$ 0.294	$ 0.356	$ 0.286	$ 0.299	/million BTU
	$ 2.97	$ 3.61	$ 2.89	$ 3.02	/MWe-hr
Difference from minimum	2.7%	24.6%	0.0%	4.5%	

A comparison at higher uranium prices and comparable thorium prices is shown in Table 4. The yellowcake prices are about double today's low levels, while the price of thorium has been reduced to $60 per kg. Because of the higher uranium costs, the tails assay from the enrichment plant has also been lowered to 0.2 wt % U-235. Thorium is produced from monazite ore, a rare-earth-thorium-phosphate mineral and also as a byproduct of the processing of heavy mineral sands for titanium, zirconium or tin (Hedrick 1996). Thorium is about three times as abundant in nature as uranium, but has a higher price today, primarily because of the smaller amounts mined. On the other hand, about 60,000 tons of uranium are mined worldwide annually. Thus one would expect that the economies of scale have already been achieved for uranium. The continued use of low enriched uranium in LWRs, where five to ten kilograms of uranium must be mined per kilogram of fuel, can be expected to put upward pressure on the price of uranium in the next century.

Note that a higher price for uranium and a slight decrease in the thorium price result in a 20 to 30 % cost advantage for the mixed ThO_2-UO_2 fuel. While the uranium used in the mixed ThO_2-UO_2 fuel is more highly enriched, the total UO_2 content is only 25 % or 35 % of the uranium only fuels. Therefore, less total uranium is required, 9.4 kg of uranium per kg of fuel for the mixed ThO_2-UO_2 fuel versus 15 kg of uranium per kg of fuel for the UO_2 fuel, where the burnup for both is 72 MWd/kg. The cost of the mixed ThO_2-UO_2 fuel should be relatively insensitive to the price of thorium, since no enrichment is required (or possible since Th-232 is the only isotope.) Therefore, no enrichment tails are generated and only 0.75 or 0.65 kg of natural thorium is needed per kilogram of fuel.

Table 4 Cost comparison using higher uranium prices.

Fuel Cost Comparison Using Future Prices					
	Current PWR Uranium Dioxide Fuel	Ultra High Burnup Uranium Dioxide Fuel	Mixed Uranium/ Thorium Dioxide Fuel 72 MWd/kg	Mixed Uranium/ Thorium Dioxide Fuel 100 MWd/kg	units
Specific Power	37.935	37.935	37.935	37.935	kWth/kg
Total cycle length	4.5	6	6	10	years
Effective Full Power Days	1186	1897	1897	2635	efpd
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Product U-235 enrichment	4.50%	8.00%	19.70%	19.70%	atom %
Tails U-235 content	0.20%	0.20%	0.20%	0.20%	atom %
Feed U-235 content	0.711%	0.711%	0.711%	0.711%	wt %
Product U-235 enrichment	4.446%	7.907%	19.500%	19.500%	wt %
Tails U-235 content	0.197%	0.197%	0.197%	0.197%	wt %
Fraction uranium in fuel	1.000	1.000	0.250	0.350	
Separative work	7.613	15.862	11.182	15.654	kg-SWU/kg fuel
Burnable poison	0.00	0.05	0.00	0.00	kg/kg fuel
Natural uranium	8.273	15.014	9.398	13.157	kg/kg fuel
Natural thorium	0.000	0.000	0.750	0.650	kg/kg fuel
Total heavy metal mined	8.273	15.014	10.148	13.807	kg/kg fuel
	0.184	0.209	0.141	0.138	kg/MWd
Rates					
Interest rate	8.0%	8.0%	8.0%	8.0%	per year
Natural uranium	$ 50.00	$ 50.00	$ 50.00	$ 50.00	/kg
	$ 26.80	$ 26.80	$ 26.80	$ 26.80	/lb U3O8
Natural thorium	$ 60.00	$ 60.00	$ 60.00	$ 60.00	/kg
Gadolinium	$ 115.00	$ 115.00	$ 115.00	$ 115.00	/kg
Conversion U3O8 to UF6	$ 5.00	$ 5.00	$ 5.00	$ 5.00	/kg enr U
Separative work	$ 75.00	$ 75.00	$ 75.00	$ 75.00	/kg-SWU
Costs					
Raw materials	$ 413.66	$ 756.44	$ 514.88	$ 696.83	/kg fuel
Separative work	$ 570.94	$ 1,189.61	$ 838.63	$ 1,174.08	/kg fuel
Conversion	$ 5.00	$ 5.00	$ 1.25	$ 1.75	/kg fuel
Fabrication	$ 200.00	$ 250.00	$ 300.00	$ 300.00	/kg fuel
Total cost	$ 1,189.60	$ 2,195.30	$ 1,654.75	$ 2,172.66	/kg fuel
Interest during use	$ 214.13	$ 526.87	$ 397.14	$ 869.06	/kg fuel
Total fuel cost	$ 31.20	$ 37.83	$ 28.51	$ 30.43	/MWth-day
	$ 0.381	$ 0.462	$ 0.348	$ 0.372	/million BTU
	$ 3.86	$ 4.68	$ 3.53	$ 3.76	/MWe-hr
Difference from minimum	9.4%	32.7%	0.0%	6.7%	

PROLIFERATION CONSIDERATIONS

This section of the report compares the characteristics of plutonium produced in a high burnup thorium-based fuel with that produced in conventional fuel irradiated to 45 MWd/kg in a PWR. The comparisons are based on the heat and spontaneous neutrons produced by a given mass of separated plutonium.

The clandestine chemical separation of plutonium from spent commercial reactor fuel is a concern with the wider use of uranium fission for generation of electricity. While the safeguards and inspections specified in the Nuclear Non-Proliferation Treaty of 1967 (NPT) can reduce the risk that a clandestine nuclear weapon will be fashioned by a national or sub-national group, the inherent characteristics of the spent fuel can add another layer of protection. The chemical separation of plutonium from spent fuel is difficult, but, it is far easier than the separation of the various plutonium isotopes.

Characteristics of Plutonium from Various Sources

Several standard grades of plutonium have entered the nomenclature, as summarized in Table 5 (Mark 1992). Obviously, pure Pu-239 would be the most desirable for weapons use. Super-grade and weapons-grade plutonium are produced by irradiating natural or depleted uranium targets to - relatively low fluences. Reactor-grade and mixed U-Pu oxide (MOX) grade plutonium is produced at higher fluences in low enrichment fuel. Of course, these grades are not fixed and the particular mixture of isotopes produced in any fuel cycle will depend on the neutron spectrum and flux, the length of the cycle, and the cooling allowed both during refueling outages and after discharge from the reactor.

Table 5 Isotopic composition of various grades of plutonium.

	Isotopes				
Grade	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242
Super-grade		98.0%	2.0%		
Weapons-grade	0.012%	93.8%	5.8%	0.35%	0.022%
Reactor-grade	1.3%	60.3%	24.3%	9.1%	5.0%
MOX-grade	1.9%	40.4%	32.1%	17.8%	7.8%

The spontaneous neutron generation rate and the amount of decay heat produced by the various plutonium isotopes determine the usability of a particular mixture of isotopes for use in a clandestine weapon. The properties of the dominant plutonium and americium isotopes are shown in Table 6 (from Mark 1992). Am-241 is the produced by the 14.4 year beta decay of Pu-241 and will grow into separated plutonium over time. The reasons for the desirability of Pu-239 and the corresponding undesirability of Pu-238, Pu-240, Pu-242 and Am-241 for weapons use are apparent. The spontaneous neutron generation rate and the decay heat rate for Pu-239 is orders of magnitude lower than for the adjacent isotopes. In order to make any separated plutonium as undesirable as possible for clandestine use in a weapon, the fractional of Pu-238 should be raised as high as possible. Pu-238 decays with a 5.5 MeV alpha particle and 87.7 year half-life to stable and non-fissile U-234.

Table 6 Properties of dominant plutonium and americium isotopes.

Isotope	Halflife years	Bare Crit kg, α–phase	Spontaneous Fission Neutrons neutrons/gm-s	Decay Heat Watts/kg
Pu-238	87.7	10	2600	560
Pu-239	24,100	10	0.022	1.9
Pu-240	6,560	40	910	6.8
Pu-241	14.4	10	0.049	4.2
Pu-242	376,000	100	1700	0.1
Am-241	430	100	1.2	114

Production of Plutonium in Reactor Fuel

The total amounts of plutonium produced in the various fuel cycles are compared in Table 7. Note that the total amount of plutonium produced in any of the mixed ThO_2-UO_2 fuels is about a factor of 4 to 4.5 less than that produced in the conventional UO_2 fuel burned to about 45 MWd/kg fuel. This follows from the fact that the U-238 content of the mixed ThO_2-UO_2 fuels is only about 20 to 28 % of that present in conventional fuels. Furthermore, the amount of Pu-239 is a factor of 6.5 less than that of the conventional fuel, first due to the smaller amount of U-238 present and secondly due to the higher burnup during which part of the Pu-239 is fissioned.

Table 7 Plutonium production in U and ThO_2-UO_2 cycles.

Plutonium Production in UO2 and Mixed ThO2-UO2 Cycles				
	UO2	UO2	ThO2-UO2	ThO2-UO2
Irradiation Time (years)	4.5	6	6	10
Burnup (MWd/kg)	45	72	72	100
Production	gram/kg ihm			
Pu-238	0.270	0.661	0.443	0.815
Pu-239	5.969	7.512	1.405	1.904
Pu-240	2.454	2.997	0.760	1.106
Pu-241	1.636	2.195	0.540	0.732
Pu-242	0.687	0.917	0.630	0.979
Total Pu	11.015	14.282	3.777	5.535
Production per MWd				
grams Pu/MWd	0.245	0.198	0.052	0.055
relative	4.67	3.78	1.00	1.06
grams Pu-239/MWd	0.133	0.104	0.020	0.019
relative	6.97	5.48	1.02	1.00

Difficulty in Weapons Fabrication

The spontaneous neutron production and the decay heat production in the various plutonium grades are shown in Table 8. Note that the spontaneous neutron rate from any plutonium separated from

the ThO_2-UO_2 fuel is more than double that of the conventional fuel and at least 15 times that of either of the weapons grades. More importantly, the decay heat production is four to five times higher than the conventional fuel and some forty times greater than either of the weapons grades.

Table 8 Spontaneous neutron and decay heat.

Comparison of Spontaneous Neutron and Heat Production						
	Super grade	Weapons grade	U 4.5 Yr, 45 MWD/kg	U 8 % enr, 6 yr, 72 MWD/kg	Th-U 6 yr, 72 MWD/kg	Th-U 10 yr, 100 MWD/kg
Composition						
Pu-238	0.0%	0.012%	2.4%	4.6%	11.7%	14.7%
Pu-239	98.0%	93.80%	54.2%	52.6%	37.2%	34.4%
Pu-240	2.0%	5.80%	22.3%	21.0%	20.1%	20.0%
Pu-241	0.0%	0.35%	14.9%	15.4%	14.3%	13.2%
Pu-242	0.0%	0.022%	6.2%	6.4%	16.7%	17.7%
Spontaneous Neutron Production						
(n/kg-s)	1.82E+04	5.35E+04	3.72E+05	4.20E+05	7.72E+05	8.65E+05
relative	1.0	2.9	20.4	23.1	42.4	47.5
Decay Heat						
(W/kg)	2.0	2.3	16.9	29.0	68.4	85.0
relative	1.0	1.1	8.5	14.5	34.2	42.5
For a 6-kg sphere						
Spontaneous Neutrons						
n/s	1.09E+05	3.21E+05	2.23E+06	2.52E+06	4.63E+06	5.19E+06
Decay Heat						
(Watts)	12	14	101	174	410	510
Temperature (C)	87	93	262	333	471	512
(F)	189	200	504	631	881	954
(see text for explanation of temperature calculation)						

In order to estimate the difficulty of handling separated plutonium from any of the cycles, a 6-kg mass of plutonium was assumed. (A plutonium mass of 6 kg was chosen because it was the approximate size of the Trinity pit (Mark 1992). A larger mass of plutonium would, of course, be needed when using plutonium from LWR fuel, which would result in even higher temperatures.) The total decay heat for such a mass was calculated and an equilibrium temperature was calculated. This equilibrium temperature assumed that the plutonium was in a sphere 88 mm in diameter and that all the heat removal was via blackbody radiation to room temperature with a surface emissivity of 1.0. The temperatures shown are the surface-temperature of the sphere. The actual surface temperature in a nuclear weapon will be significantly higher because the plutonium sphere will be enclosed inside a blanket of high explosives and a metal casing. Note that the surface temperatures for the ThO_2-UO_2 cases (471 and 512 °C) are relatively close to the melting point of plutonium (650 °C).

Another consideration is the heating of the high explosive surrounding the separated plutonium. The high Pu-238 content results in a heat generation of about 400 to 500 W for the mixed ThO_2-UO_2 fuels, compared with about 100 W for the conventional reactor fuel and less than 15 W for the two weapons grades.

While the thermal conductivity for the high explosive used in U.S. weapons is not available (and may vary), one may surmise that it is about 0.2 to 0.5 W/m-K. This admittedly simplistic analysis assumed a spherical geometry and a high explosive thickness of 60 mm. Using this range of thermal conductivity, Table 9 shows that peak temperatures at the plutonium-high explosive interface are above the melting/damage point for the high explosive[2] and in some cases above the melting point of the plutonium. For this analysis, the plutonium mixture from the 6-year, 72 MWd/kg thorium-uranium fuels was assumed. Clearly, the use of plutonium containing 10 to 14 % Pu-238 in a weapon would present a severe thermal management challenge.

Table 9 Peak temperatures at the Pu-high explosive interface.

Peak High Explosive Temperatures			
(using Pu from 72 MWd/kg UO2-ThO2 fuel)			
k HE	T interface		
(W/m-K)	K	C	F
0.2	1714	1441	2626
0.3	1243	970	1778
0.4	1007	734	1354
0.5	866	593	1100
0.75	677	404	760
1	585	312	594
2	441	168	335

Predetonation due to Spontaneous Neutrons

Using the methods of Von Hippel and Lyman[3] the pre-detonation probability for a plutonium mixture can be estimated. The probability that the yield, Y, will exceed a fraction, x, of the design yield, Y_o, is

$$P\left(\frac{Y}{Y_0} > x\right) = \int_x^1 dx'\left(\frac{dP}{dx'}\right) = \exp\left(-\frac{1}{2}Nt_0x^{2/3} + 45N\tau\right)$$

where t_o is the time of maximum supercriticality, assumed by Mark to be 10^{-5} s
τ is the average time between neutron generations, assumed by Mark to be 10^{-8} s
N is the spontaneous neutron generation rate, neutrons per second.

The integral probability is shown in Figure 8. Note that the probability of the yield exceeding 0.10 of the design yield is less than 10 % for both of the thorium-uranium fuels.

It is apparent that all plutonium mixtures derived from power reactors substantially reduce the probable yield of a crude weapon. The two weapons' grades have so few spontaneous neutrons that their probabilities of exceeding 99 % of the design yield are 60 % and 23 % respectively. The probabilities that any of the reactor grades will exceed 4 % of the design yield range from 75 % for

[2] High explosives, such as RDX, melt in the 200° C range. The explosive may still function in a liquid state, but the weapon would have lost the necessary precision in its explosive configuration.
[3] Frank Von Hippel and Edwin Lyman, Center for Energy and Environmental Studies, Princeton University, Princeton, New Jersey, cited in Mark 92.

the conventional 45 MWd/kg fuel to 52 % for the 100 MWd/kg ThO_2-UO_2 fuel. Both of the ThO_2-UO_2 fuels have less than a 7 % probability that the yield will exceed 10 % of the design yield. While 10 % of the design yield is still a significant blast, these probabilities would make plutonium derived from thorium-uranium fuel of little or no interest to a potential weapons state.

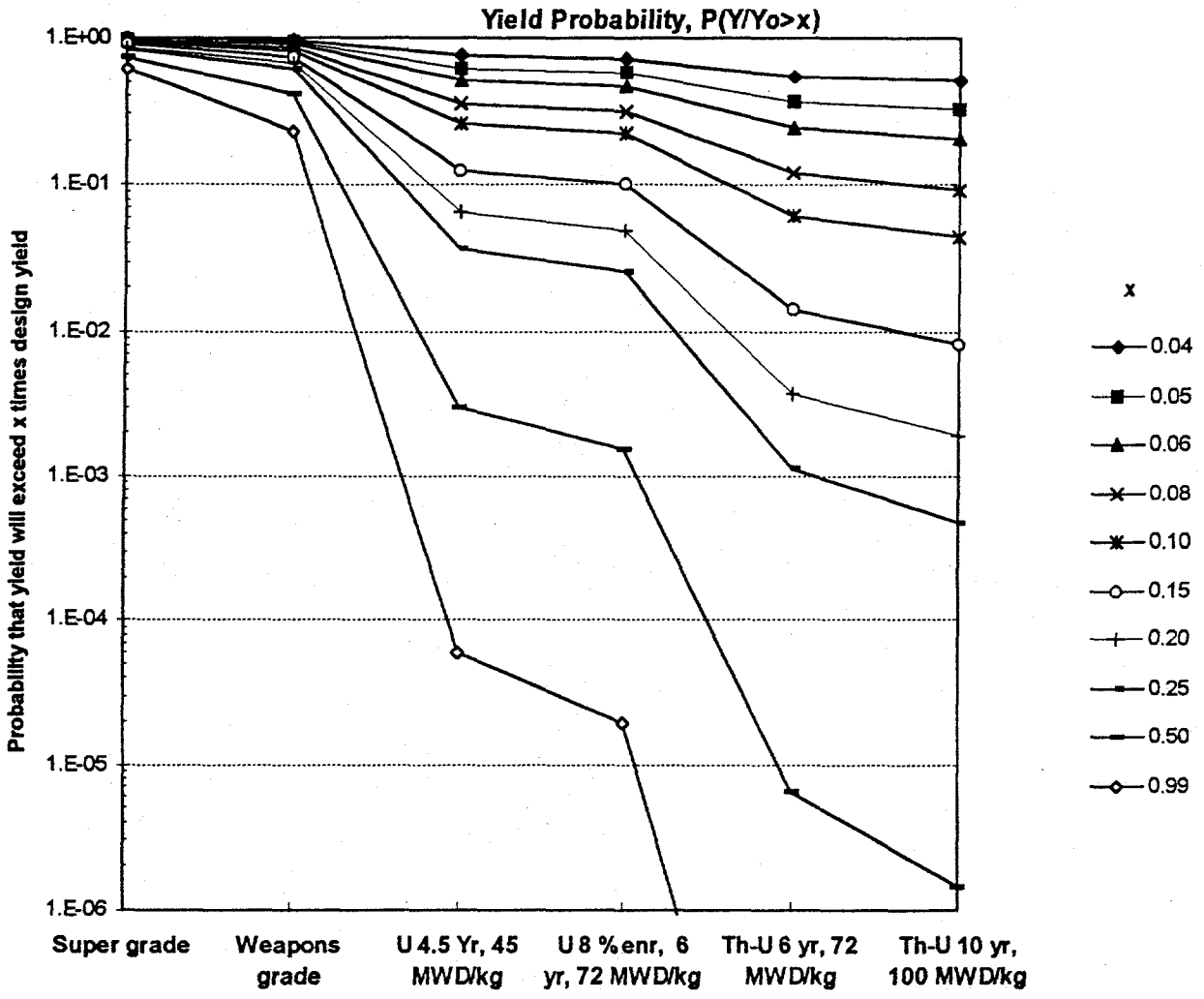

Figure 8 Yield limitation due to pre-detonation.

ThO_2-UO_2 WASTE FORM CHARACTERISTICS

The Nuclear Waste Policy Act of 1982 and the 1987 amendments to the law were enacted to provide for permanent disposal of high-level radioactive wastes in the United States in a deep geologic repository located at an isolated, arid location. A potential site for the repository is in the volcanic tuff beds at Yucca Mountain, Nevada. This site is a hydrologically unsaturated zone, i.e., damp conditions but limited water flow. The Yucca Mountain ground water contains a number of contaminates which may react with the fuel; the most abundant of which are sodium and silicon (Finn et. al 1998a). The Yucca Mountain waste containers that are dripped on are expected to resist corrosion failure for at least three thousand, and more likely, tens of thousands of years (about 25% of the waste containers that are dripped on are expected to fail at 100,000 years,

containers that are not dripped on fail considerably later) (DOE/RW 1998). The corrosion rates for the Zircaloy fuel cladding are from ten to many thousands of times slower than the waste container material under the extreme acid conditions assumed for Yucca Mountain. Therefore, fuel exposure does not occur until about 30,000 years after the waste packages fail. At about one million years, DOE/RW estimates that about 30% of the spent fuel will be exposed to a humid air environment and may also be exposed to an oxidizing dripping water environment.

To reliably contain the fission products and actinides within the fuel after loss of the waste containers and cladding, the fuel must not undergo significant chemical reactions and resultant changes in crystal structure. Thorium dioxide is the highest oxide of thorium and does not depart significantly from its stoichiometric composition of ThO_2 when exposed to air or water at temperatures up to 2000°K (Belle and Berman 1984). Spent uranium dioxide fuel fragments, on the other hand, react relatively rapidly (i.e. at a rate of about 1 % per year) with 90°C high-drip-rate water with representative Yucca Mountain contaminates (Finn et. al 1998b). Air-oxidation of UO_2 spent fuel at temperatures near but below 200°C produces U_4O_9 after only several years of exposure (Thomas et. al 1993). Because of the UO_2 content, ThO_2-UO_2 mixtures appear to be susceptible to corrosive attack in air or oxygenated water, but significantly less susceptible than UO_2. The available information of this subject is discussed in the following paragraphs.

Dry Air Oxidation. Because thoria is the highest oxidation-state of thorium, the oxidation and physical damage of pressed and sintered ThO_2/UO_2 mixtures is caused entirely by the chemical properties of the urania. Urania has the property of nonstoichiometry, so that a series of urania compositions indicated by the following molecular formulas may exist:

$$UO_{2-x},\ UO_2,\ UO_{2+x},\ U_4O_{9-x},\ U_4O_9,\ U_4O_{9+x},\ U_3O_{8-x},\ U_3O_8,\ UO_3$$

where x is a number less than 1 (Belle 1961, Thomas et. al 1993). Urania crystallizes in a fluorite lattice. Upon oxidation, unirradiated UO_2 takes up extra oxygen until U_4O_{9-x} begins to precipitate. Further oxidation then results in U_4O_9 precipitation and so on. However, high burnup UO_2 has displayed enhanced oxidation resistance with a stoichiometry that equilibrates near $UO_{2.4}$ with the structure of U_4O_9 (Thomas et. al 1993). Apparently, it takes only a relatively low level of impurity to stabilize the matrix and change the oxidation sequence of UO_2.

The particular molecular structure is important because the theoretical densities of UO_2, U_4O_9, and U_3O_8 are 10.97, 11.4, and 8.35 Mg/m^3. Therefore, conversion of UO_2 to U_3O_8 results in about a 30% volume increase, grain boundary separation, and powdering of the fuel. If the fuel is contained in typical LWR cladding, oxidation of the UO_2 to U_3O_8 is likely to cause severe splitting of the cladding (Novak et. al 1983). However, oxidation to U_4O_9 results in a slight densification (3.4%), some micro cracking, and no structural damage to the fuel.

Cohen and Berman (1966) investigated the extent of oxygen solubility in unirradiated UO_2-ThO_2 solid solutions as a function of temperature and composition. They found that after oxidation, the value of x in (Th, U)O_{2+x} increased continuously from 0, at ThO_2, to a maximum value of 0.25 for 50% mixtures. In other words, heavily oxidized high thoria solid solutions, from $(U_{0.5}Th_{0.5})O_{2+x}$ up to pure ThO_2, are in equilibrium with oxygen with urania molecular structures between UO_2 and U_4O_9 and, therefore, swelling, grain boundary separation, and fuel powdering does not occur. Higher urania oxides do not form when there is at least 50% thoria present because the thoria stabilizes the fluorite structure and only one interstitial oxygen can be accommodated per unit cell,

only where an appropriate number or uranium ions are adjacent to the space occupied by the oxygen. Thomas et. al (1993) report a somewhat similar stabilization due to the fission products in high burnup spent LWR UO_2 fuel oxidized at temperatures below 200°C in air.

Moist Air and Water Oxidation. The oxidation of UO_2 in water requires the presence of dissolved oxygen, but "proceeds in a completely different manner from oxidation in air or gaseous oxygen" (Belle 1961). The major reaction product upon contact with pure water is $UO_3 \bullet 0.8H_2O$, a hydrate called dehydrated schoepite. Markowitz and Clayton (1970) investigated the corrosion behavior of a group of nuclear fuel oxides, including two compositions (20 and 50% UO_2) of urania-thoria, in high temperature (360°C), alkaline (pH 10) flowing water. The ThO_2 samples displayed excellent corrosion resistance. The weight gains of the urania-thoria material exposed to water oxygenated to about 100 ppm "were much larger than for any of the other materials tested, in any of the media, even those which failed and fell apart; yet all the specimens retained their mechanical integrity". The urania-thoria fuel remained mechanically intact in spite of the growth of an oxidized surface phase thought to be of the M_4O_9 type. $UO_3 \bullet 0.8H_2O$ was apparently not found in or on the Markowitz and Clayton specimens.

Finn et. al (1998b) has reported on the results of tests with spent uranium dioxide fuel fragments which were exposed for a number of years to 90°C drip-water containing representative Yucca Mountain contaminates. As mentioned above, this material reacted (dissolved) at a rate of about one percent per year in the high-drip-rate tests. The major alteration products were Na-boltwoodite, $Na[(UO_2)(SiO_3OH)] \bullet H_2O$, which is formed from the sodium and silicon in the simulated groundwater, and dehydrated schoepite. Although there is no information available on the behavior of mixed urania-thoria fuels exposed to similar conditions, the results of Cohen and Berman (1966) and Makowitz and Clayton (1970) would suggest far less structural damage than observed by Finn et. al with UO_2.

ThO_2-UO_2 FUEL PERFORMANCE

ThO_2-UO_2 fuel has significantly different properties than UO_2 fuel. These differences include (1) slightly higher decay heat, (2) significantly higher thermal conductivity at LWR operating temperatures and lower thermal conductivity at high temperatures, (3) slightly higher fission gas production per fission, but a significantly lower rate of release of fission gases, and (4) higher melting temperature (Belle and Berman 1984, Goldberg et al. 1978). These differences in properties will cause ThO_2-UO_2 fuel rods to behave somewhat differently than UO_2 fuel rods during both normal operation and design basis accident conditions. During normal operation, ThO_2-UO_2 fuel operated at the same power level as UO_2 fuel will experience somewhat lower fuel temperatures (except at the fuel surface) and less fission gas release than UO_2 fuel. This will allow higher pre-pressurization and thereby minimize cladding creepdown and fuel-cladding mechanical interactions at high burnup and thereby possibly allow for higher burnup use of this material. During an accident such as a large break loss-of-coolant accident (LOCA), ThO_2-UO_2 fuel will have less stored energy but a slightly higher internal heat generation rate than UO_2 fuel at similar power levels. As a result, certain parameters for accident evaluation such as the maximum cladding temperature and the timing of fuel rod rupture are expected to be slightly different. These expected differences in behavior between ThO_2-UO_2 fuel rods and UO_2 fuel rods need to be quantified for an objective evaluation of the performance of ThO_2-UO_2 fuel.

CONCLUSIONS

Several important conclusions can be drawn from this analysis.

The mixed ThO_2-UO_2 fuel, using a mixture of 25 wt % UO_2 and 75 wt % ThO_2, where the uranium is initially enriched to 19.5 wt % U-235, appears to have sufficient reactivity to be used for extended burn cycles to 72 MWd/kg in LWRs. Likewise, a mixture containing 35 wt % UO_2, with the same enrichment, and 65 wt % ThO_2, appears suitable for extended cycles approaching 100 MWd/kg of initial heavy metal. The *in situ* breeding of U-233 maintains a more uniform reactivity during the course of irradiation and eliminates the need for burnable poisons.

The mixed ThO_2-UO_2 fuel reduces the amount of total plutonium production by a factor of 4.5 and the Pu-239 production by a factor of 6.5, when compared to conventional UO_2 fuel irradiated to 45 MWd/kg.

The cost per unit energy of the mixed ThO_2-UO_2 fuel is 13 to 25 % less than the conventional or extended burn UO_2 fuels, if present uranium prices are used. If projected future prices of uranium and thorium are used, the advantage for mixed ThO_2-UO_2 fuels is 20 to 33%.

At no time during the fuel cycle can a uranium component be chemically separated from the fuel that could be usable in a nuclear weapon.

The plutonium that is produced in the mixed ThO_2-UO_2 fuel is high in Pu-238, producing copious amounts of decay heat and spontaneous neutrons. The high decay heat from a 6 kg sphere of plutonium would melt or burn any surrounding explosive used in fashioning a crude weapon, unless the weapon was actively cooled. The spontaneous neutrons drastically limit the probable yield of any such crude weapon.

A matrix of ThO_2 and UO_2 is more resistant to long-term corrosion in oxygenated water than is a matrix of only UO_2. Thus ThO_2-UO_2 is a superior waste form if the spent fuel is slated for direct disposal rather than reprocessing.

ACKNOWLEDGEMENT

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PROLIFERATION RESISTANT, LOW COST, THORIA-URANIA FUEL FOR LIGHT WATER REACTORS*

Annual report

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June 1999

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Table of Content

List of Figures

List of Tables

1. Summary

Project Objectives:

Our objective is to develop a fuel consisting of mixed thorium dioxide and uranium dioxide (ThO_2-UO_2) for existing light water reactors (LWRs) that (a) is less expensive overall than the current uranium-dioxide (UO_2) fuel, (b) allows longer refueling cycles and higher sustainable plant capacity factors, (c) is very resistant to nuclear weapons-material proliferation, (d) results in a more stable and insoluble waste form, and, (e) generates less high level waste. This project is being conducted in collaboration with INEEL. This annual report presents the MIT progress in the investigations from October 1998 up to June 1999.

Technical Progress:

Neutron physics analysis was performed using the CASMO-4 and MOCUP codes. CASMO-4 simulation of one assembly runs were made surveying and comparing the reactivity-limited burnup capability of thorium-uranium fuel over the full range of current and potential fuel; compositions. This survey assumed the same geometry fuel would be utilized for the new fuel as being used in today's reactors. The first order survey does not identify any significant economic incentives for the use of thorium in preference to uranium. However, second order effects of burnable poison needs were not considered. Furthermore, the effects of different moderator to fuel ratio and size of the fuel pin have yet to be investigated.

To evaluate fuel performance, NRC-sponsored FRAPCON-3 and MIT-developed FLA02 computer codes have demonstrated good agreement on fission gas release for a test UO2 fuel system. The FLA02 code was used to assess the impact of differences in thermal properties alone on gas release. The results show significantly less gas released in the thoria based fuel for the same power and energy output. More work on thoria modeling needs to be done to extend gas release prediction capability. The FLA01 corrosion and temperature model comparison to FRAPCON-3 do not agree and must be evaluated further and modified to achieve reasonable and consistent results. The results of fuel performance assessment indicate that operating Zircaloy clad fuel to the six-year level could lead to very severe cladding corrosion. More corrosion resistant materials such as ZIRLO should be evaluated for the extended cycle lifetime.

In the area of thermal hydraulics, the key differences affecting safety involve higher decay heat levels, moderately higher thermal conductivity of thoria-based fuels, and differences in power peaking. Thoria-urania fuel was found to exhibit better performance during LOCA due to its higher thermal conductivity and lower specific heat capacity and density in comparison to the case of UO_2 fuel for the same conditions. The decay heat data from the ORIGEN2 calculations indicate that thoria-urania fuels generate more decay heat in the long term (roughly a month after shutdown), but the differences are not so substantial to significantly affect capabilities of heat removal systems.

In terms of waste management issues, the Th containing fuel is found to create more ^{129}I and ^{234}U due to the higher amount of ^{233}U. However, the U fuel forms more ^{99}Tc, ^{237}Np, and ^{239}Pu and some higher actinides. Hence not only would plutonium production be reduced by the new fuel, but also the long-lived isotopes responsible for the high fraction of the dose to the public, based on the recent viability assessment. Under Yucca Mountain conditions, ThO_2 solubility should be minimized and U(IV) should oxidize to U(VI). U(VI) is more soluble than Th or U(IV). Dissolution of U can form Th colloids. The ThO_2 may act as a sink for U(VI) and form secondary phases. Additionally, the presence of ThO_2 in the fuel matrix may stabilize the U. Experiments must be conducted in order to obtain Th thermodynamic data for the oxide and mixed uranium species.

Other Accomplishments

A proposal to the Nuclear Energy Research Initiative (NERI) of the Department of Energy (DOE) has been successful. The proposal expands the investigation of the thorium fuel cycle to include four fuel vendors (Westinghouse, Framatome, Siemens and ABB-CE) and two universities (Purdue and U. of Florida) as well as Argonne National Laboratory. The NERI award is for $1M/yr for three years.

A special session on Thorium Fuel Cycle was organized for the June 1999 meeting of the American Nuclear Society in Boston. Four papers were presented by each of MIT and INEEL members of this project.

Discussions were held with visiting Russian scientists from the Kurchatov Institute in Moscow on the program they are following for the development and testing the Radkowsky thorium fueled seed-blanket assembly. This program is being supported by DOE through Brookhaven National laboratory.

A review of the previous studies of thorium-based fuels in LWRs was completed and issued in April 1999.

2. Introduction

Further development of nuclear power depends on improvements of its performance with regards to economics, safety, waste production and proliferation resistance. The use of thorium in a once-through fuel cycle has the potential to improve all four features. It is likely to reduce the need for control materials in a fresh core while enhancing the production of fissile materials during the fuel residence, both of which will reduce the fuel cost. Thorium dioxide has a higher melting point and a higher thermal conductivity compared to uranium dioxide, which will enhance the thermal margin of safety. Thorium fuel reduces the free fission gases inside the cladding, which might allow for a substantial increase in fuel efficiency thus reducing the spent fuel volume for a given amount of electricity. While the above potential benefits will need to be evaluated in specific designs, one benefit is almost design independent, and that is the benefit of added proliferation resistance. Thorium will reduce the plutonium production in the fuel and so change the isotopic content of the plutonium to one that is less effective for weapons, both of which enhance nonproliferation.

In the past, the use of thorium in light water reactors (LWR) was always evaluated in the context of recycling the fuel. In this project we aim to develop a once-through fuel cycle that will maximize the potential benefits due to uranium-233 generation from thorium-232 to the cycle cost and proliferation resistance. There are two constraints for the thorium use. One is that the fresh fuel has to rely on uranium-238 enriched with uranium-235 to less than 20%. The other is the inclusion of enough uranium-238 to ensure that the total quantities of uranium-235 and uranium-233 do not exceed the non-proliferation guidelines. Under these constraints, the fuel to be used is a mixture of uranium and thorium dioxide. The project will evaluate the best approach to design such a fuel for use in a Pressurized Water Reactor such that it will be at least 10% more economic than the current all uranium fuel.

The project will evaluate four aspects of this fuel cycle;

1. The neutronic behavior of the fuel and the achievable burnup with thoria-urania fuel,
2. The behavior of the fuel under long irradiation periods representative of what might be needed to assure an economic cycle,
3. The thermal hydraulic and safety considerations for the new fuel, and
4. The implications for the spent fuel behavior both during the storage period and eventually in a repository.

This project, which is being supported under the INEEL/MIT Strategic Nuclear Research Collaboration Program for Sustainable Nuclear Energy, was initiated in October 1998.

3. Neutronics and Fuel Management

M.J. Driscoll, Xianfeng Zhao, K. Clarno

3.1 Introduction

During the period covered by this report the following work was accomplished:

a) The CASMO-4 [Edenius et. al., 1995] and MOCUP [Moore et. al., 1995] codes were acquired, made operational, and employed for some initial benchmark comparisons.

b) CASMO-4 assembly runs were made surveying and comparing the reactivity-limited burnup capability of thorium-uranium fuel over the full range of current and potential future interest. A report (SB thesis) was prepared and issued on this topic [Clarno, 1999].

c) Prior literature was surveyed and evaluated to identify the principal neutronic differences between thorium-rich and all-uranium PWR fueling [Kazimi et. al., 1999]. A summary of the findings has been published [Zhao et. al., 1999], and a series of parametric studies using CASMO-4 has been carried out to validate and quantify these projections.

d) Input and draft review was provided in support of the successful submission by INEEL of a NERI proposal in the subject area.

In the sections that follow, the more important findings are summarized.

3.2 Code Acquisition and Benchmarking

MIT now has the latest versions of both CASMO-4 and MOCUP operational on a new DEC-alpha computer. Good agreement was realized in initial testing of an all-uranium unit cell using CASMO-3, CASMO-4 and MOCUP, following which the latter two codes were exercised on a 75% Th, 25% U unit cell. The results are shown in Figure 3.1, and show satisfactory agreement. Note that because of limitations on the then available MOCUP (MCNP) library, the comparison was made at room temperature. Temperature dependent cross sections for thorium are available in the UTXS library and will be imported via INEEL, based on the procedural arrangements made at the INEEL-MIT SNRC project review workshop on April 6-7, 1999. Once this is done, analysis shown in Figure 3.1 will be repeated under hot full power conditions and extended to ~ 1200 efpd.

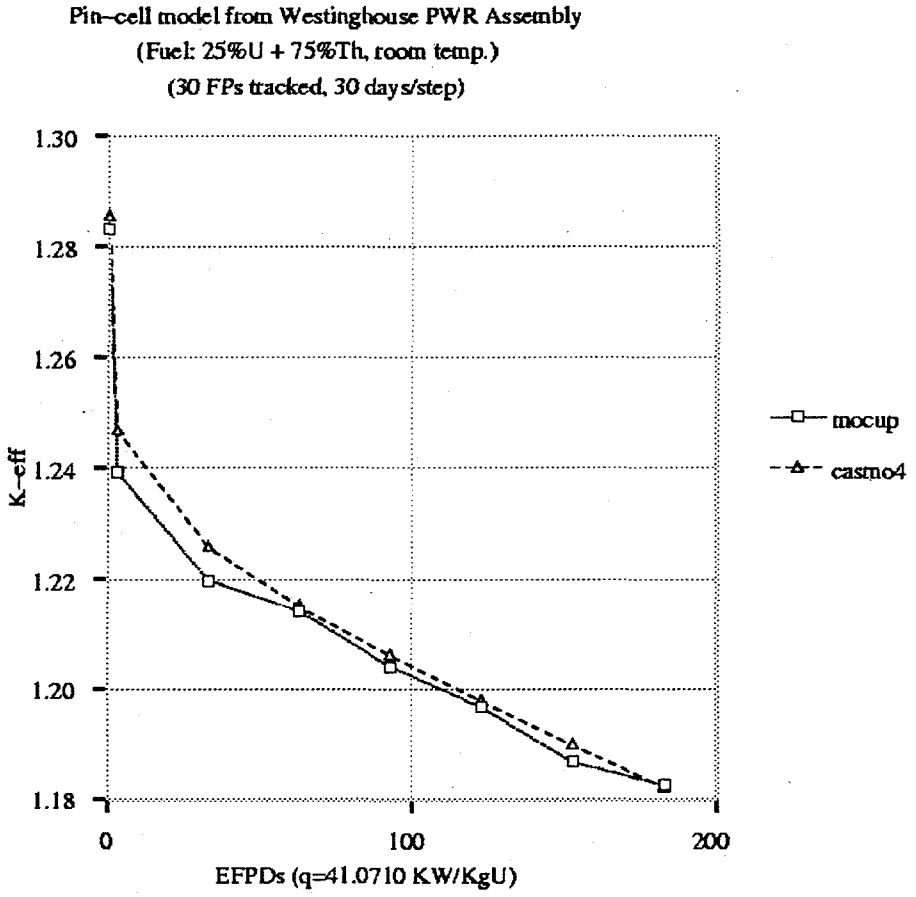

Figure 3.1 Comparison of MOCUP and CASMO-4 Unit Cell Calculations

3.3 CASMO-4 Parametric Survey

A matrix of uniform lattice compositions (standard Westinghouse 17 x 17 assemblies) was evaluated in which the weight percent uranium (in U+Th) was varied over the range 25 to 100%. Maximum U-235 enrichment was kept □ 19.5 w/o and the U^{235}/(Th+U) ratio

held constant — which also keeps the natural uranium requirement very nearly constant. Details are documented in Clarno [1999].

Figure 3.2 shows a typical set of k_∞ (B) plots, in this case for a 25% U, 75% Th case and its 100% U counterpart. Burnup capability was assessed by interpolating to find B_1 at which $k_\infty \equiv 1.03$ (i.e., 1.0 plus an allowance for whole core leakage). For n batch management ($1/n^{th}$ of core replaced each refueling, here 1/3) discharge burnup $B_d = [2n/(n+1)]\, B_1$. As can be seen, the all-U case has a longer reactivity-limited burnup.

Table 3.1 summarizes the overall results of this survey. The first member of each "Set" employs 19.5 w/o U-235 enrichment which is progressively reduced as the w/o of total U is increased. As can be seen, the natural uranium requirements are essentially constant within each set; the separative work requirements somewhat less so, but not vastly different. The final three columns in the table compare "utilization": the energy produced divided by the resource consumed — a good surrogate for fuel cycle economics. As can be seen, only at the highest burnups do thorium rich lattices match all-U performance, and the compositions involved are really ~ 50/50 Th/U mixtures.

3.4 Thorium Lattice Neutronics

Prior evaluations of thorium fueling dating back to the 1960s were reviewed, including the comprehensive NASAP and INFCE studies completed and reported in 1980. To the extent feasible, their findings were used to prepare the performance comparison of thorium and uranium lattices summarized in Zhao et. al. [1999]; Table 3.2 from that compilation is reproduced here. The overall conclusion is that while there are a variety of individual differences in the parameters which determine burnup and licensing transient performance, none are large enough to be outside the envelope of acceptability defined by current PWR technical specifications and reload submissions to NRC.

However, it must be noted that the information in Table 3.2 is based on studies operating under different criteria and constraints than those of current applicability: burnups of only ~ 30 MWd/kg, and use of highly enriched (> 90 w/o U-235) uranium, in particular. Hence, a series of CASMO-4 runs have been carried out to quantify the variation of core parameters with burnup for our reference 75% Th, 25% U lattice. Selected results are shown in Figures 3.3 and 3.4, which plot the values of the (poison-free) moderator temperature coefficient and the fuel doppler reactivity coefficient over the burnup range of interest. Note that core-average values will be a weighted average of fresh, once and twice burned assemblies over an operating cycle (roughly 20 MWd/kg). Differences consistent with and explainable by the difference in U-235/U-233, U-238/Th-232 nuclear properties are evident, but again nothing to raise serious concern.

Table 3.1 Summary of fuel cycle perfromance calculations

Set	%U	Burnup 1-Batch (MWD/kg-Ihm)	Burnup 3-Batchse (MWD/kg-Ihm)	Natural Uranium Requirement (kgUnat/kg-Ihm)	Separative Work Requirement (kg-SWU/kg-Ihm)	Natural Uranium Utilization (MWD/kg-Unat)	HM Resource Utilization (MWD/kg-HMR.)	Separative Work Utilization (MWD/kg-SWU)
A	25.0	32.50	48.75	11.664	6.621	4.179	3.927	6.672
	50.0	32.99	49.49	11.482	6.383	4.310	4.131	7.764
	80.0	36.92	53.88	11.263	6.796	4.784	4.700	7.928
	100.0	39.44	59.17	11.117	6.944	5.322	5.322	8.521
B	35.0	50.61	75.92	16.928	9.830	4.649	4.471	7.723
	50.0	50.48	75.71	16.219	10.208	4.668	4.629	7.417
	80.0	52.04	78.07	16.000	10.636	4.879	4.819	7.340
	100.0	54.65	81.97	15.654	10.794	5.171	5.171	7.594
C	45.0	66.45	99.67	20.998	14.044	4.747	4.626	7.097
	80.0	66.96	100.43	20.742	14.583	4.842	4.796	6.887
	100.0	68.79	103.19	20.596	14.747	5.010	5.010	6.997
D	55.0	80.96	121.44	25.661	18.252	4.732	4.651	6.653
	80.0	80.99	121.49	25.479	18.588	4.768	4.731	6.536
	100.0	82.14	123.21	25.333	18.756	4.864	4.864	6.569

Table 3.2 Predicted Effects of Th/U-233 on PWR Neutronics Compared to all-U Fueling

Parameter	With Th/U-233	Principal Causes	Notable Effects
Moderator temperature coefficient (MTC)	Progressively negative with burnup	Lattice neutronics less sensitive to spectral hardening	Less effect of lattice design changes
Doppler coefficient	More negative, less so with burnup	Th-232 effective resonance integral, while smaller than that of U-238, is more strongly increased by temperature; less Pu-240 buildup	Improved transient response to rapid severe reactivity, (hence power) increases
Xenon worth	Slightly less	U-233 has lower yield of I-135 + Xe-135, but higher direct yield of Xe-135	• Reduces reactor control needed • Higher direct yield of Xe-135 increases stability against Xenon oscillations
Fission product poisoning	Slightly different	• U-233 has a different yield mix than U-234 and lower σa than Pu • somewhat offset by higher thermal σa of Th-232 vs. U-238	Only slightly disadvantageous
Delayed neutron fraction, β	Decrease with burnup is slightly more than in all-U core	β of U-233 is considerably less than that of U-235, but comparable to that of Pu-239, Pu-241	• Roughly same detrimental effect on accidents involving sudden large reactivity insertions • More rapid power decrease during scram
Reactivity loss due to burnup	Appreciably less	Higher η of U-233 produces more excess neutrons for increasing the conversion ratio	• Less poison reactivity requirement at BOC • Burnup prediction more sensitive to errors in reactivity prediction
Hot to cold reactivity difference	Smaller	Smaller MTC outweighs larger fuel (Doppler) TC	No control modification needed to accommodate use of Th
Control requirements	Reduced overall; but rod worth a bit less at low burnup	Smaller change with burnup dominates other effects; lower σa than Pu-239, 241; increases poison worth	• Can reduce (or eliminate) soluble or burnable poison concentrations • Easier to design long-cycle/high burnup cores
Local power peaking	Somewhat less both assembly- and pin-wise	U-233 has smaller σf than Pu-239, 241, smaller local $\Delta\rho$ due to burnup	More thermal-hydraulic margin, easier to meet design constraints
Fertile capture product	Pa-233 more important absorber than Np-239	Pa-233 has T 1/2 = 27 days vs. 2.4 d for Np-239	• Delays U-233 production • Both neutrons and U-233 are lost by captures in Pa-233

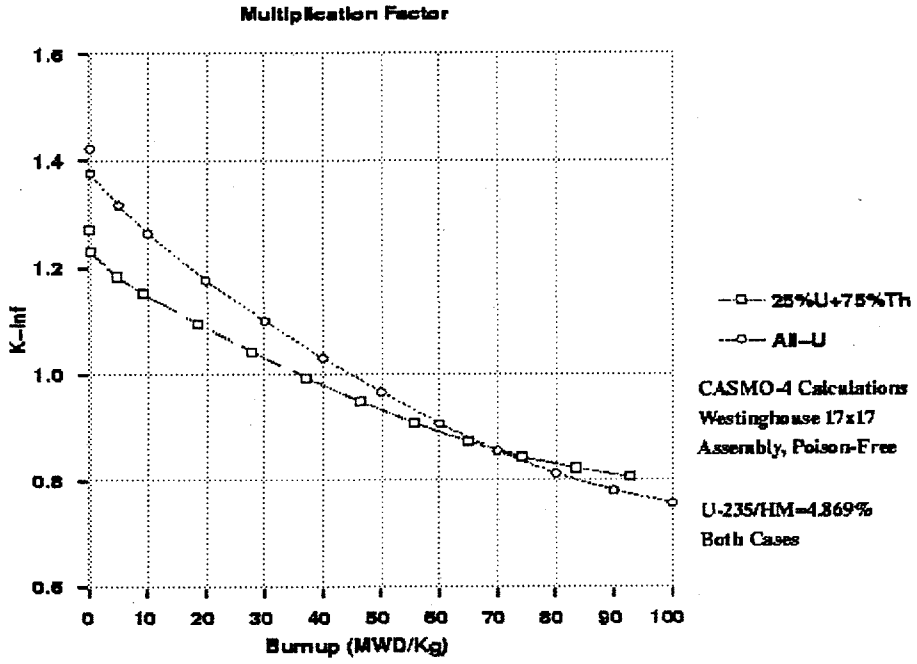

Figure 3.2 Comparison of all-U and Thorium-Rich Assembly Burnup Histories

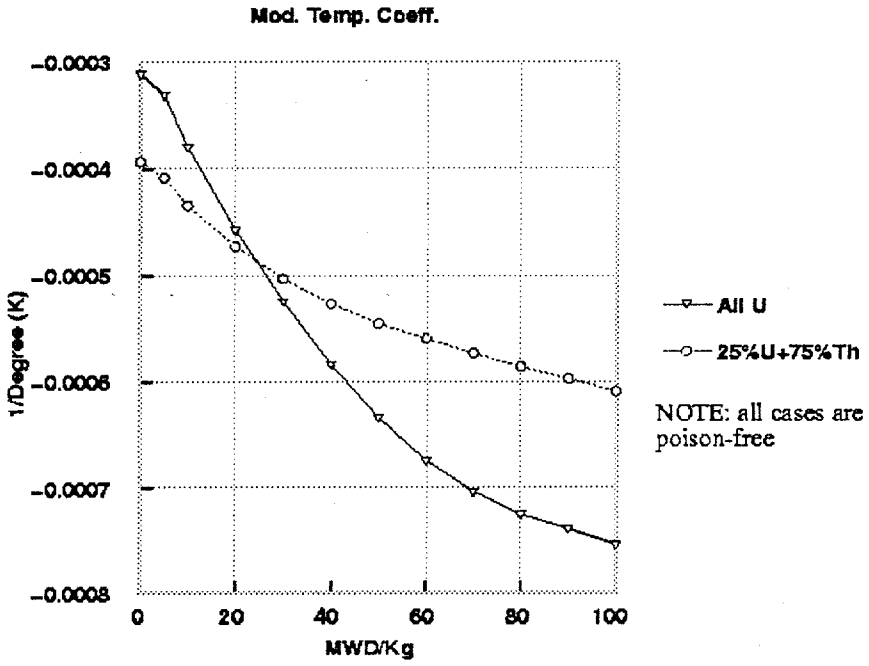

Figure 3.3 Comparison of Moderator Temperature Coefficient for all-U and Thorium-Rich Assembly Burnup Histories

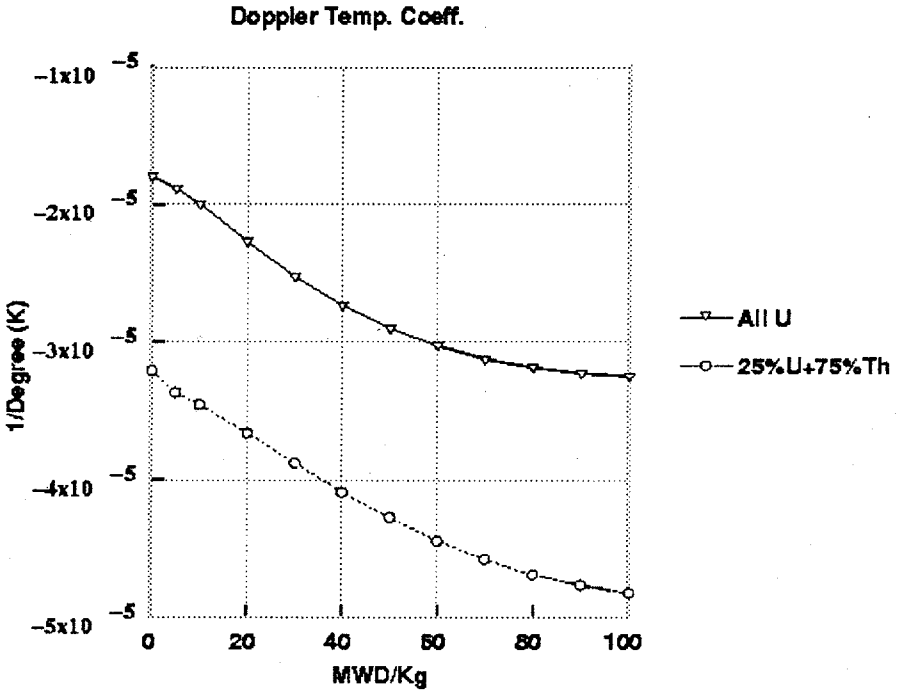

Figure 3.4 Comparison of Doppler Coefficient for all-U and Thorium-Rich Assembly Burnup Histories

3.5 Conclusions

While the virtues of thorium may be enhanced when more realistic comparisons are made, this first order survey does not identify any significant fuel management incentives for the use of thorium in preference to uranium. As it now stands, the case for thorium will have to rest on benefits with respect to better fuel durability at high burnup, enhanced anti-proliferation isotopics, and greater stability under repository conditions.

4. Fuel Performance Modeling

Yun Long, John Meyer, Ronald Ballinger, and Stephen Schultz

4.1 Introduction

The principal accomplishments achieved during the period covered by this report are:

a) Four base cases were defined to provide a framework for examining fuel performance effects.

b) Literature on US reactor Th-U fuelled cores was examined. In particular, information from the Light Water Reactor Breeder Reactor program was assessed.

c) The FRAPCON-3 code [Berna et. al., 1997] was acquired from Battelle Pacific Northwest National Laboratory. The code was installed and tested on a DEC machine (Toolboy). Key results for UO_2 fuels were compared with predictions from the MIT Fuel Lifetime Analysis (FLA) codes.

d) Estimates of the severity of cladding corrosion and fission gas release for ThO_2-UO_2 fuels were developed using the FLA codes.

e) Background investigations were initiated on additional fuel performance effects important in high burnup fuel operation such as the 'rim effect'.

4.2 Base Calculation Cases

To assure a consistent framework for evaluating fuel performance effects, four base case fuel histories were defined for use in this study. These cases, detailed in Table 4.1 and Figure 4.1, were based on [Herring and MacDonald, 1998] with modifications on specific power so that the energy produced per unit volume fuel is same for each case. Core geometry, fuel rod dimensions, and plant conditions are taken from Westinghouse 3411 MWth four loop plant specifications.

Table 4.1 Key parameters for the four base cases

Batch Characteristics					
Case	Uranium Enrichment [w/o U]	Total Uranium [w/o HM]	Calendar Years	Capacity Factor [%]	EFPD
U1	4.5	100	4.5	76	1249
Th1	19.5	14	4.5	76	1249
U2	8.0	100	6	90	1972
Th2	19.5	25	6	90	1972

Table 4.1 (Continued)

Case	Total Core Heavy Metal [Mg]	Total Core Oxide [solid m^3]	Specific Power [kW/kgHM]	EOL Batch Average Burnup [MWd/kgHM]
U1	89.91	9.311	37.94	47.4
Th1	82.84	9.311	41.17	51.4
U2	89.86	9.311	37.96	74.9
Th2	83.66	9.311	40.77	80.4

An envelope pin power history [linear heat deposition rate versus axial position and equivalent full power days] has also been defined for each case. These histories are intended to represent the demand placed on the more severely challenged fuel pins in the core during the batch lifetime. The pin challenges for this study are solely based on considerations of cladding corrosion and fuel fission gas release. The envelope pin for case U1 is the same as developed for the 18-month urania fuelled core in [Delgado et al. 1998].

Since no reactor physics calculations have been completed for the other cases, a simple extrapolation of the case U1 envelope pin has been adopted. In each case, the average linear heat deposition rate versus axial position (pin power) was taken to be constant in each of three equal duration fuel cycles as shown in Figure 4.1.

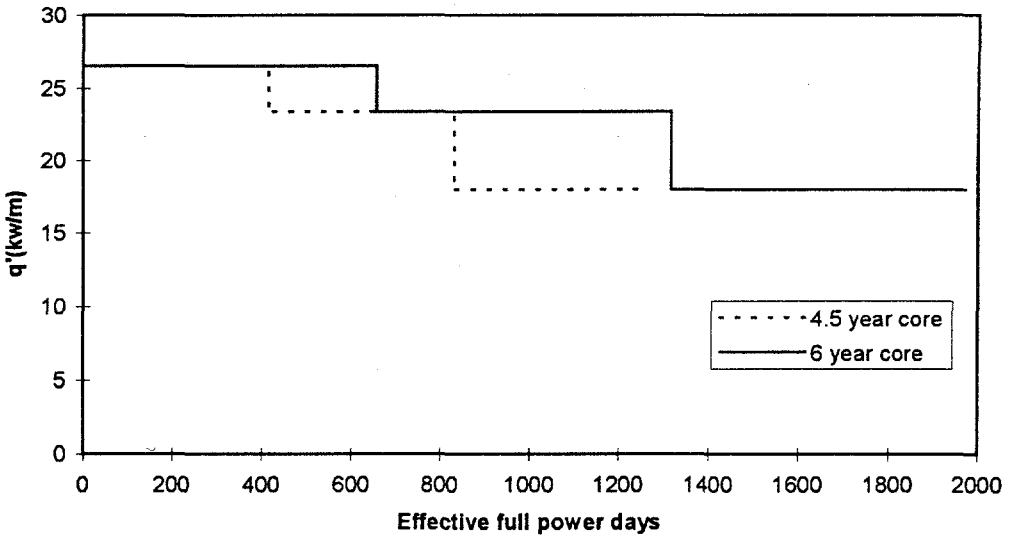

Figure 4.1 Envelope pin axial average heat deposition rate

The pin power in each cycle is the same as for case U1. The 6-year cases were defined by simply increasing the duration of each cycle. The envelope pin average power for each of three cycles are {26.5, 23.4, 18.0} kW/m or {8.08, 7.13, 5.49} kW/ft. For corrosion calculations, the envelope pin local power at z = 0.81 L was used for each of three cycles as {27.5, 24.7, 19.4} kW/m or {8.37,7.52, 5.92} kW/ft. Core average power is 17.82 kW/m (5.43 kW/ft).

Note that all linear heat rates reflect only deposition in the rods (97.4% of the total power); the remaining 2.6% is deposited in the coolant. Note also that EOL envelope pin average burnup values are large (60 MWd/kgHM for case U1 and 95 MWd/kgHM for case U2). The envelope pins are thought to be over-conservative with respect to burnup. Additional work is needed to refine our approach for defining limiting reference design cases so that they may be more representative of anticipated and practical fuel management strategies.

4.3 Historical Experience Search and Evaluation

A literature search was conducted to gather insights about Th-U fuel behavior in past reactors and in past experiments. While many well-documented observations exist, they have been obtained largely from conditions that do not represent present day LWR conditions.

In particular, the early reactor experiences of Elk River and Indian Point 1 were obtained for mostly thorium fuel with only a small fraction (5 w/o) of total uranium (highly enriched). In the current studies the uranium fraction is projected to be at or above 25 w/o. Furthermore, the cladding material was stainless steel rather than Zircaloy.

The Light Water Breeder Reactor used Zircaloy cladding material but the fuel operated at much lower power ratings and somewhat lower coolant temperatures than those planned for the present application. Key parameters of the LWBR core were gathered (mostly from Olson et al, 1998) and are compared with those of current PWR cores (Case U1) in Table 4.2.

In spite of the differences noted in Table 4.2, we expect to find significant uses for many results from the LWBR core application, from associated in-pile irradiations, and from associated technology developments. We expect to find special benefits from the results when obtaining and benchmarking the Th-U system fuel performance models required for the present project.

4.4 Code Acquisition and Verification

4.4.1 Code Introduction

The fuel performance codes employed to date in this study may be described as follows:

a) FLA Codes: FLA is a designation for an MIT package of computer codes for Fuel Lifetime Analysis. The two codes that are available in this package were developed by Luis Garcia-Delgado and are reported in [Delgado et al. 1998].

FLA01 -- Code for Cladding Corrosion– The FLA01 code provides a cladding corrosion model and is taken mostly from [Forsberg et al. 1995]. The code was originally written for modeling the behavior of low-tin Zircaloy-4 and a representative PWR coolant chemistry.

Table 4.2 Parameter comparison between LWBR and a reference case PWR

Assess Information from Light Water Breeder Reactor			
Item	Reference Case U1	LWBR Seed Pin	LWBR Standard Blanket Pin
Geometry			
Active Length (m)	3.66	2.13	2.13
Clad Outside Diameter (mm)	9.50	7.78	14.52
Clad Thickness (mm)	0.57	0.56	0.71
Clad Inside Diameter (mm)	8.36	6.65	13.09
Clad Thinness (Rad.cl.out/cl.thk)	8.33	6.90	10.17
Fuel Outside Diameter (mm)	8.19	6.40 & 6.49	12.97
Radial Gap (micron)	85	126 & 80	64 & 62
Gap Thinness (Rad.cl.in/gap)	49	26 & 42	103 & 105
Pin Arrangement	square	triangular	triangular
As-Built Features			
Clad Material	Zy-4 (low tin)	Zy-4	Zy-4
Clad Condition		RXA	CW + SRA
Helium Cold Pressure (kPa)	About 2800	101	101
Fissile U (w/o U)	4.5 (U235)	0 to 5.2 (U233)	0 to 2.0 (U233)
Total U Content (w/o HM)	100	0 to 5.3	0 to 2.0
Operations			
Calendar Years BOL to EOL	4.5	5.0	
Total EFPD	1249	1210	
Core Average Temperature (C)	310	277 to 266	
System Pressure	15.5	13.8 to 12.5	
Saturation Temperature	345	335 to 328	
Peak Linear Heat Rate (kW/m)	27.5	22.0	
Peak Local Burnup (MWd/kgHM)	63	53	
Peak to Average Burnup	Nearly Uniform	Up to Factor of Two or More	
Coolant Chemistry	Soluble Poison	No Soluble Poison	
Operation Mode	Base Load	204 plant swing load cycles	
Notes:			
a) Notation includes: Zy=Zircaloy; RXA=clad processed using a recrystallized anneal; CW+SRA= clad processed by cold working and a stress relief anneal.			
b) Linear heat rate and burnup for Case U1 from envelope pin.			
c) Linear heat rate and burnup for LWBR seed from pins subjected to post-irradiation examination			

FLA01 (continued): The technique has two chief advantages: it is non-proprietary and it is based on measured oxide thicknesses.

The FLA01 code has also been employed in the present study with provisions for changing the single phase heat transfer coefficient and for modeling standard Zircaloy-4 (not low-tin).

FLA02 – Code for Fission Gas Release – The FLA02 code provides a fission gas release (FGR) model taken from work in [Weisman et al. 1969]. The Weisman model employs parameters developed from experiments using UO_2 fuel rods. For using the model in FLA02, a temperature distribution (temperature versus [r & z] in a single rod) must be supplied as a function of time.

That distribution was originally taken from interpolating in a stored array of pellet outside temperature and pellet center temperature as a function of linear heat rate and burnup. The array was mostly taken from [Maki, 1979]. In work performed for this project, FLA02 has been applied to thoria-urania fuels by using the same arrays adjusted by using thermal conductivities from Belle and Berman 1984 and by using conductivity integral concepts.

b) FRAPCON-3 -- Fuel performance code. This code has been developed by Battelle Pacific Northwest National Laboratory and by INEEL over several years. The FRAPCON source code and documentation have been obtained, installed and tested at MIT as a product of this research. This code contains only UO_2 properties. The first MIT use is in evaluation of the existing FLA01 and FLA02 for UO_2 applications. In subsequent work code models and properties can be modified to perform Thoria fuel analysis.

4.4.2 Code Verification Efforts.

a) FRAPCON-3 Versus FLA01 on Cladding Corrosion

The 4.5 year case (U1) has been used for comparisons between the FRAPCON-3 and FLA01 codes. Standard Zircaloy-4 was used for the cladding corrosion comparison, because FRAPCON-3 does not have coefficients for low-tin Zircaloy-4. The oxide thickness versus EFPD is shown in Figure 4.2. Considering the scatter experimental data and corrosion model uncertainties, these two codes appeared to demonstrate reasonable agreement. However, relatively large, unexpected differences were observed in the temperature calculation result (Figure 4.3). Further investigation uncovered an error in the single phase heat transfer correlation in FLA01. Since this mistake exists in the original reference [Forsberg et al. 1995], it is not clear if the correct correlation was used in the development of the corrosion model. After correcting the correlation in FLA01, the temperatures from the two codes are very close (Figure 4.4), but the oxide thickness from FLA01 is unreasonably large at the last cycle (Figure 4.5). This issue is currently being pursued for clarification with the model developers.

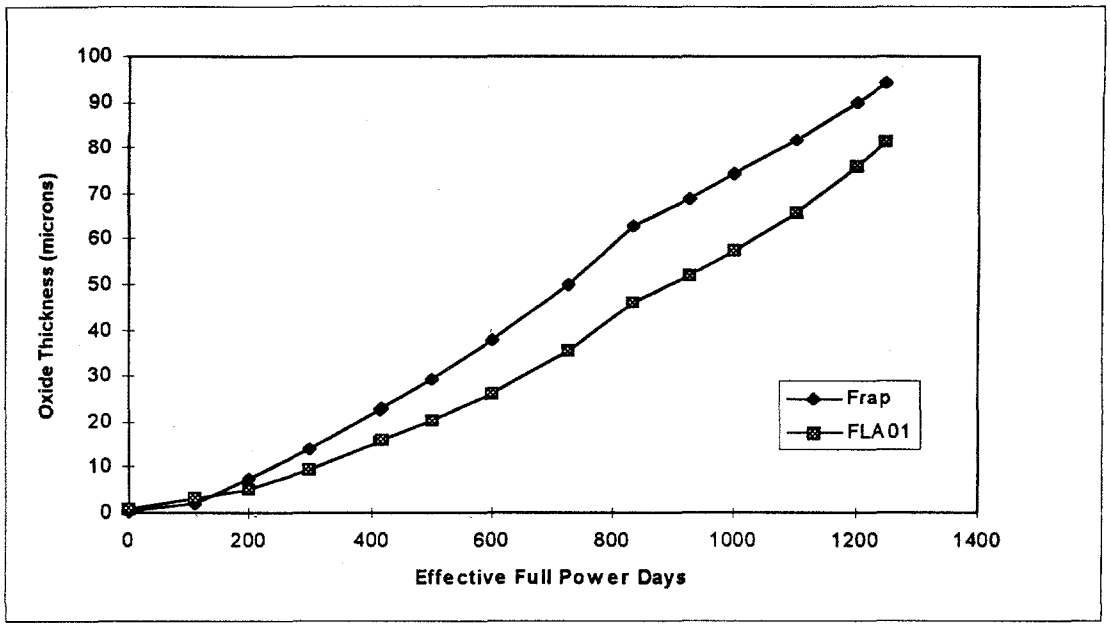

Figure 4.2 Standard Zircaloy-4 cladding corrosion for 4.5 year case
-**before** heat transfer correlation change in FLA01

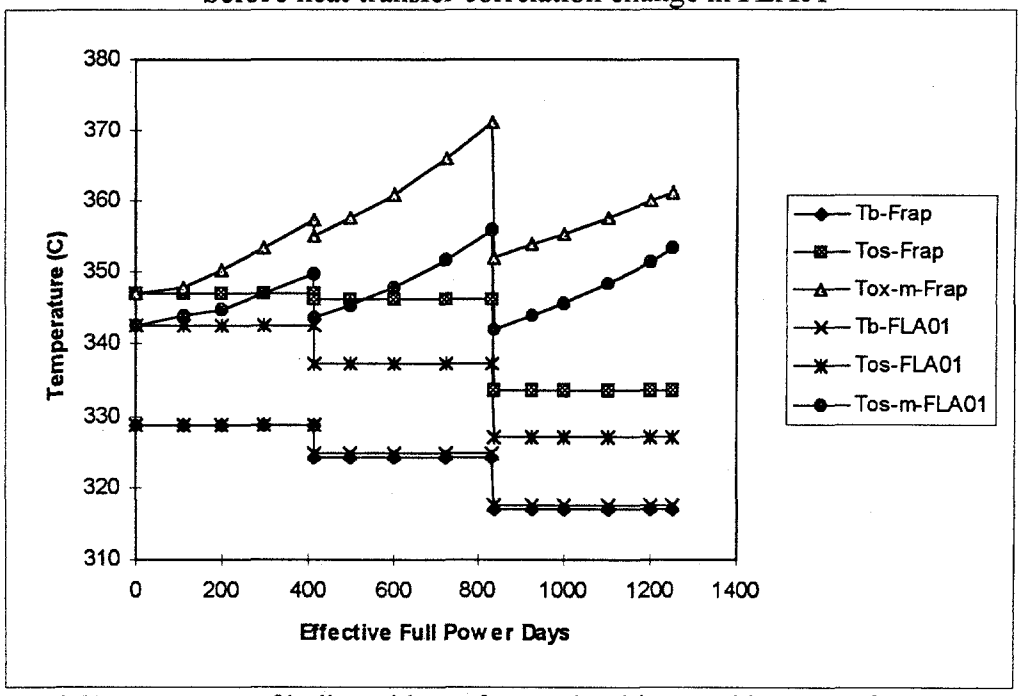

Figure 4.3 Temperature of bulk, oxide surface and oxide-metal interface for 4.5 year case
-**Before** heat transfer correlation change in FLA01

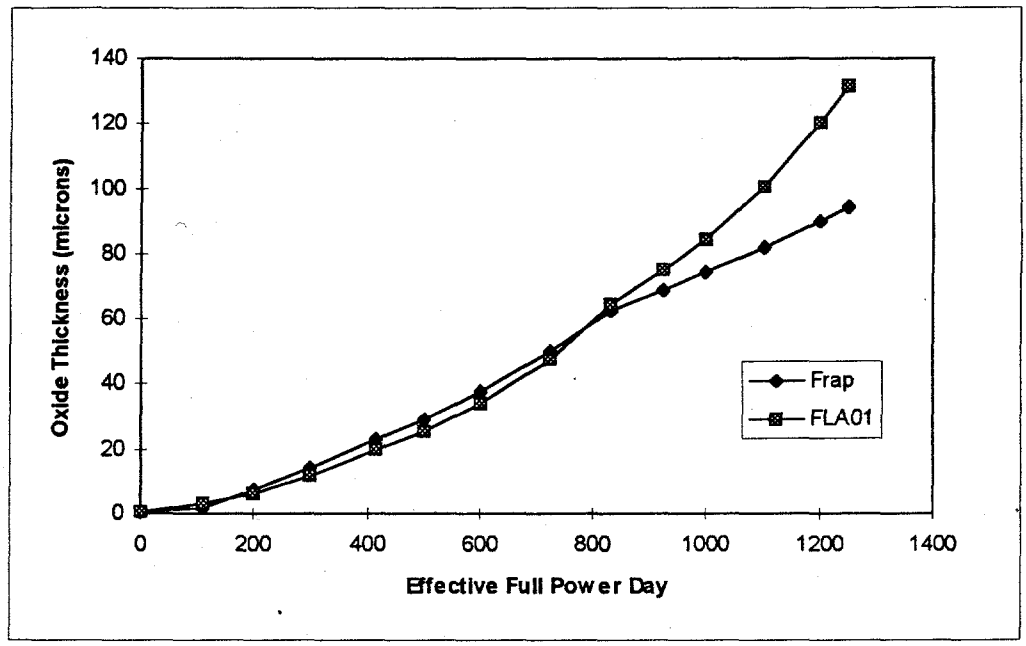

Figure 4.4 Standard Zircaloy-4 cladding corrosion for 4.5 year case
-**After** heat transfer correlation change in FLA01

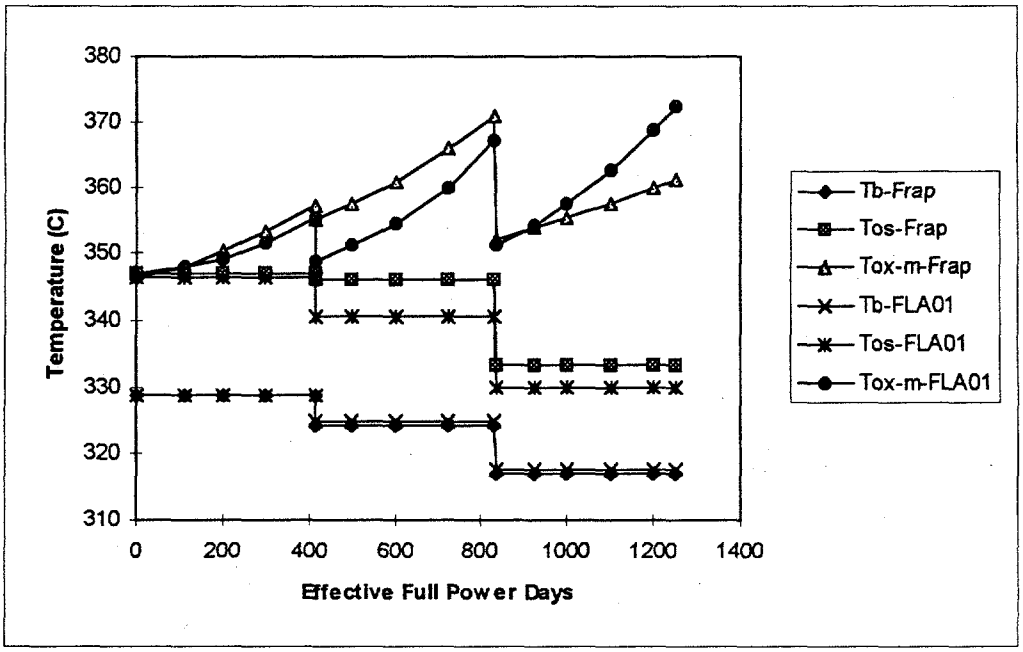

Figure 4.5 Temperature of bulk, oxide surface and oxide-metal interface for 4.5 year case
-**After** heat transfer correlation change in FLA01

b) FRAPCON-3 Versus FLA02 On Fission Gas Release Results

The 4.5 year (U1) case was used for the fission gas release comparison calculation. Fractional gas releases at EOL for the two codes are very close: 2.8% for FLA02 and 2.2% for

FRAPCON-3, considering the large uncertainties in the experimental data and the differences in model theory. The major difference lies in the release mechanism, fuel temperature calculation and power distribution as summarized in Table 4.3.

Table 4.3 Comparison of fission gas release model between FRAPCON-3 and FLA02

Code	Release mechanism	Fuel Temperature	Radial Power Distribution
FRAPCON-3	diffusion theory	direct calculation	bath-tub shape
FLA02	release probability	Extrapolated mostly from [Maki, 1979]	flat

4.5 Preliminary Evaluation on Severity of Cladding Corrosion and Fission Gas Release

To assess the problems that will be faced in fuel performance assessment and modeling in this project, preliminary calculations on cladding corrosion and fission gas release were made for the four calculation cases using the FLA codes.

4.5.1 Cladding Corrosion

The oxidation of Zircaloy was calculated by a 2-stage model. In the first stage (pre-transition oxidation) the oxidation layer is very compact and protective and grows according to a cubic rate equation. In the second oxidation stage (post-transition oxidation) the oxidation rate is significantly increased due to formation of cracks in the oxide that allows oxygen augmented access to the metal surface. The cladding material in this calculation is low-tin (1.3 w/o) Zircaloy-4. The oxide thickness for the fuel in the 6 year core reached 464 μm. This thickness is far beyond the range of aplicability of the FLA02 code. Data from 8 PWRs [Garzarolli et al., 1992] show most of UO2 fuel rods measured develop oxide thickness between 20 and 60 μm, In two reactors the measurements exceeded 100 μm. Oxide thickness values up to 100 μm are probably acceptable. Therefore corrosion clearly appears to be a severe challenge on cladding material design for high burnups.

Cladding development has focused on reducing corrosion and thus reducing the hydrogen content of the cladding and maintaining mechanical properties at high burnup levels. The fuel vendors are moving to advanced zirconium alloys of proprietary compositions and processing techniques. In general, these advanced alloys are low in tin and high in iron and nickel compared to standard Zircaloy-2 and Zircaloy-4. The Zr-Nb alloys are also being evaluated for their ability to maintain ductility at high burnup levels. Small alloy additions to zirconium liners are being investigated to reduce the susceptibility of barrier cladding in boiling water reactors to secondary degradation after a cladding breach. Fuel rods of ZIRLO and three variants of Zircaloy-4 cladding have been irradiated in North Anna Unit1 reactor for two cycles to an assembly average burnup of about 37.8MWd/kgHM. ZIRLO exhibited the best in-reactor corrosion performance with an average axial peak oxide thickness 32% of that formed on conventional Zircaloy-4 [Sabol etal,1994]. The in-reactor corrosion resistance of ZIRLO improved relative to the Zircaloy-4

materials with increasing time. ZIRLO cladding exhibited increased in-reactor dimensional stability with about 80% of the creep and 50% of the growth of Zircaloy-4.

4.5.2 Fission Gas Release

The basic fission gas release mechanisms for uranium fuel and thorium fuel are assumed to be the same at this stage of the investigation. However, the differences in gas production rate, thermal conductivity, and release temperature could greatly affect the relative release fraction between the two fuels.

a) Fission gas production.

For the thoria fuel system the production of more than twice as much Kr per 233U-fission results in a total of approximately 10 percent more total fission gas(Xe plus Kr) production[Belle & Berman, 1984]. In operation, U-233 will build up and U-235 will decrease in Th-U fuel. The fission gas production varies accordingly. Gas production for U-Th fuel and uranium fuel was calculated using the ORIGEN-2.1 code. Figure 4.6 displays the result for the 6-year cases(U2, Th2) showing that the thorium fuel produces more fission gas than Uranium fuel. These results have been derived based on the current ORIGEN-2.1 library. Accuracy could be improved by using the MOCUP code to reach high burnup and accessing the high burnup library for ORIGEN-2.1.

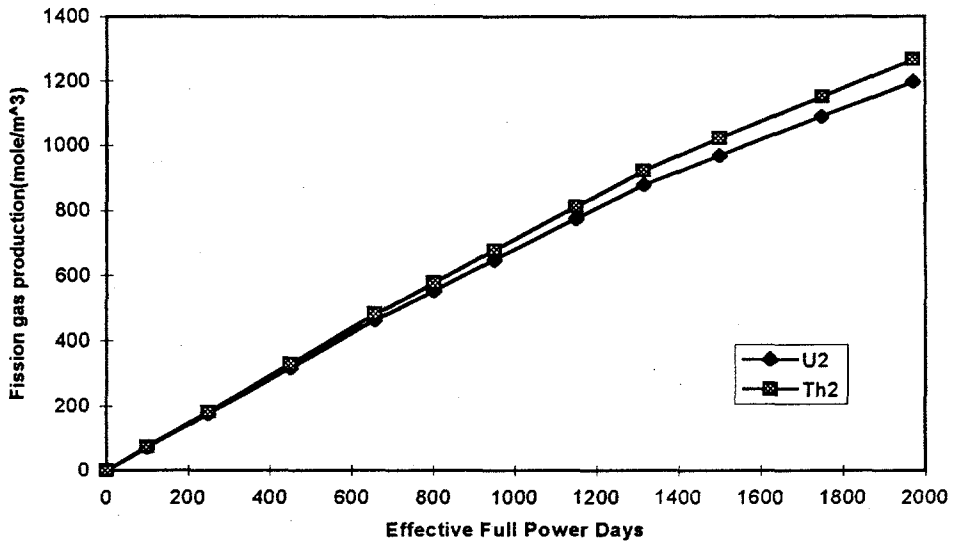

Figure 4.6 Fission gas production - 6 year cases

b) Fuel temperature.

In this calculation assessment for thoria fuel, the surface temperature is the same as for uranium fuel. The fuel centerline temperature is adjusted by solving the following equation:

$$\int_{T_s}^{T_{cU}} k_u(T)dT = \int_{T_s}^{T_{cTh-U}} k_{Th-u}(T)dT = \frac{q'}{4\pi},$$

Because the burnup-dependent thermal conductivity correlation for thoria fuel has not been developed at this time in the project, unirradiated thermal conductivity is also adopted in uranium fuel for a preliminary assessment. In this study, the temperature difference between the fuel centerline and fuel surface is 7.5-10.5% lower for thoria fuel. The cooler thoria fuel is expected to exhibit lower fission gas release.

c) Release temperature.

Thoria fuel has higher dislocation release temperature and grain boundary release temperature. Thus, much less fission gas release is expected for the thoria fuel.

The FLA02 code uses release probability theory instead of release temperature. The release temperature effect should result in a lower release probability in thoria fuel. Unfortunately, the constant coefficients in the release model are not currently available for thoria fuel. Therefore, only the difference in fission gas production and thermal conductivity are considered in this study. Figures 4.7 and 4.8 show the fractional gas release for the 4.5 year core and 6 year core. Thoria fuel has substantially lower fission gas release. The lower release rate expected for thoria fuel would increase the margin.

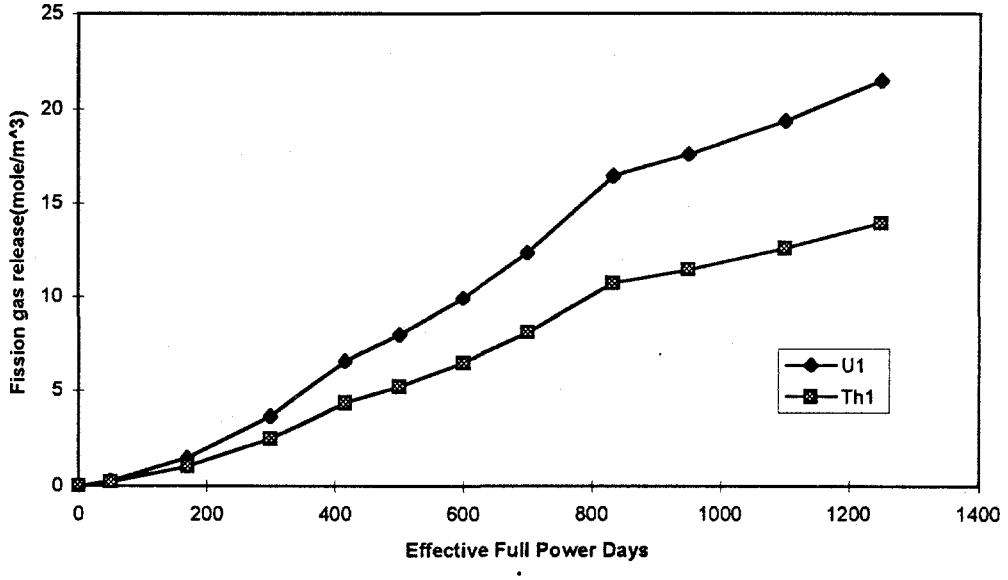

Figure 4.7 Fission gas release-4.5 year cases

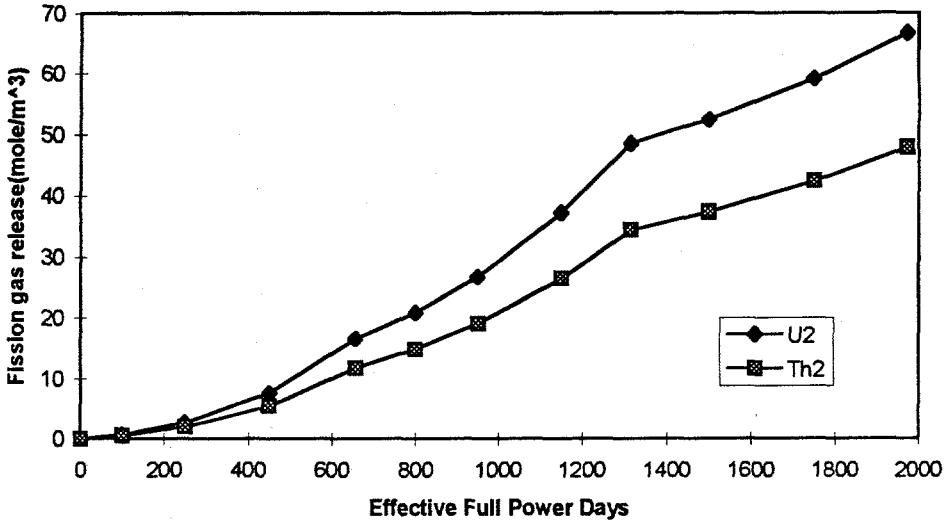

Figure 4.8 Fission gas release-6 year cases

4.6 Additional High Burnup Fuel Performance and Design Considerations

In the course of research in this period, two performance and design features important to high burnup for thoria fuel have been identified. These are the fuel pellet rim effect and annular pellet design.

4.6.1 Fuel Pellet Rim Effect

Recent research on UO2 fuel at high burnup shows that plutonium buildup near the outer edge of fuel pellets distorts the power distribution at high burnup and can affect the fuel microstructure in the outer rim of the fuel. The impact of these changes on fuel temperature, fission gas release, cladding strain, and fuel thermal conductivity are of interest for fuel burnup exceeding 50 MWD/kgHM. The USNRC expects fuel vendors and licensees to assess the impact on fuel performance and fuel-related safety analysis for operation of any UO2 fuel rods in excess of 60 MWD/kgHM. Comparison of the expected performance of thoria fuel with respect to these effects will be important.

4.6.2 Annular Pellet Design

Neutronics of the thoria fuel design can be enhanced with the use of annular pellets. Somewhat higher manufacturing cost may be offset by improved neutron economy and fuel thermal and mechanical performance. FRAPCON-3 is capable of modeling annular pellet designs in UO2 fuel. Fuel modeling studies, in concert with neutronics and fuel management design, are recommended.

4.7 Conclusions

Based on above calculations and findings, the following conclusion could be drawn:

- FRAPCON-3 and FLA02 have demonstrated good agreement on fission gas release for a test UO2 fuel system. More work on thoria modeling needs to be done to extend gas release prediction capability. The FLA01 corrosion and temperature model must be evaluated further and modified to achieve reasonable and consistent results.
- There are significant differences between LWBR core fuel experiences and those anticipated for the U-Th oxide cores of this project. However, it is expected that the LWBR experience will be useful for code and model development and benchmarking.
- The completed severity estimates of corrosion and fission gas release indicate that fuel performance to the six year level could lead to very severe cladding corrosion. More corrosion resistant materials such as ZIRLO should be evaluated.

5. Thermal-hydraulic and Safety R & D considerations

P. Hejzlar, M. Kazimi and P. LaFond

For the homogeneous thoria-urania core, the design of the assembly is essentially that of the present day LWRs. Therefore, the thermal performance is expected to be similar to that of the current LWR fuel assemblies. The main differences, which will impact on safety performance, come from higher decay heat levels, moderately higher thermal conductivity of thoria-based fuels, and differences in power peaking. Since the last feature can be assessed only after the reactor physics calculations provide a core power distribution, only the first two issues will be discussed.

5.1 Impact of decay heat generation

Higher decay heat is the consequence of several factors – higher burnup conditions, different fuel composition, and slightly more fissions required to produce desired core power level. As a result of extended burnup, the amount of fission products in the fuel will be higher increasing the decay power at the end of life. At high burnups, the fuel contains a significant fraction of Pu-238, which is primarily produced from U-235 through three neutron absorptions and two β^- decays. Pu-238 has an α-decay halflife of 87.7 years leading to an appreciable decay rate with considerable thermal power output. The calculations at INEEL [Herring and MacDonald, 1998] showed that the decay heat production of all plutonium isotopes per kg of Pu from the ThO_2-UO_2 fuel is four to five times higher than that of Pu from conventional LWR fuel. However, the total mass of plutonium is almost proportionally lower for the thoria-based fuel so that overall effect of Pu on the total decay heat generation rate is not so significant. The U-233 isotope, which at higher burnups replaces fissile U-235, also exhibits a higher decay heat generation rate compared to U-235. Finally, the smaller energy yield per fission of U233 (191MeV) than that of U235 (193.7MeV) requires slightly more fissions at fixed reactor power, which leads to an increased amount of fission products and thus higher decay power.

To assess the differences in the decay heat generation rates between the U-based and Th-based fuels, decay heat was evaluated using the ORIGEN2.1 computer code [Croff,1980], developed at Oak Ridge National Laboratory (ORNL). The four cases that were studied are summarized in Table 5.1. For both the all-uranium and thorium-uranium dioxide fuels, two cases involving low and high burnups were run. The differences in specific powers are due to the slightly different fuel pellet heavy metal densities of each fuel form.

Table 5.1 Case summary of decay heat calculations

Case	Total Uranium [w/o HM]	Enrichment [w/o U-235]	EFPD	Specific Power [kW/kgHM]	EOL Batch Average Burnup [MWd/kgHM]
Case 1	100%	4.5%	1249	37.94	47.4
Case 2	14%	19.5%	1249	41.17	51.4
Case 3	100%	8.0%	1972	37.96	74.9
Case 4	25%	19.5%	1972	40.77	80.4

The decay heat during the first 72 hours after shutdown is shown in Figure 5.1. A time interval of 72 hours was selected because it is a standard for advanced reactors for the performance of safety systems without human intervention. The decay powers for low and high burnups for both urania and thoria-urania fuels are very close. Hence, the effect of the length of the exposure does not seem to be substantial in this time period for the burnups considered here. Comparing the UO_2 and ThO_2-UO_2 fuel, the latter fuel exhibits slightly higher decay power after 10 seconds. At 500seconds, the difference is about 7%.

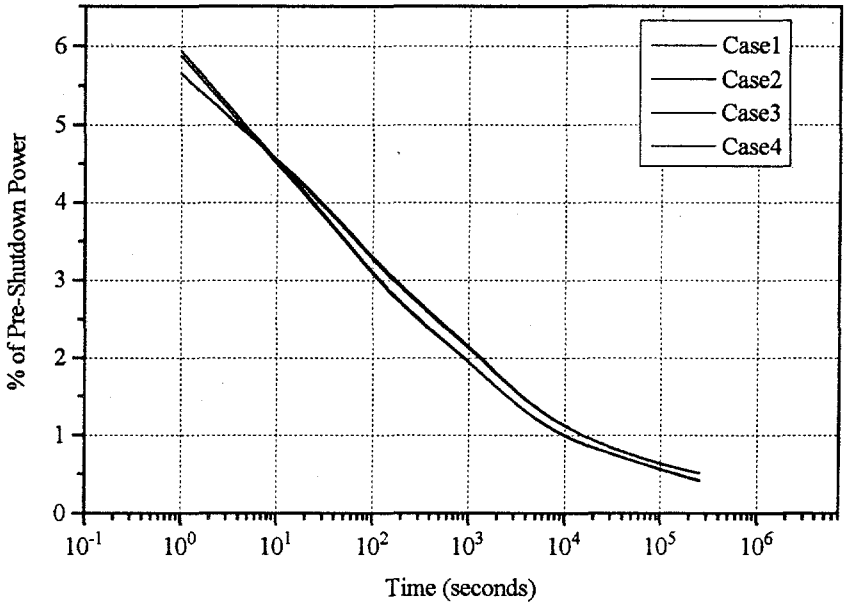

Figure 5.1. Decay heat during first 72 hours after shutdown

Figure 5.2 compares decay power up to 6 months after shutdown. This time interval was chosen to investigate the effect of increased heat generation rate on core cooling systems during extended shutdown. Significantly higher decay heat levels for the ThO_2-UO_2 fuel can be observed at the time interval between 20 to 60 days. This difference stems from the decay of Pa-233, which with its halflife of 27days significantly contributes to the overall heat generation rate. Although the difference in decay heat generation is relatively large during this time period (almost 70% at 30 days), the effect on cooling capability during shutdown will be small, because the cooling systems are designed for much higher power than the levels observed at such a long time after shutdown. After Pa-233 decays, the relative difference in decay power of thoria-based fuel is reduced.

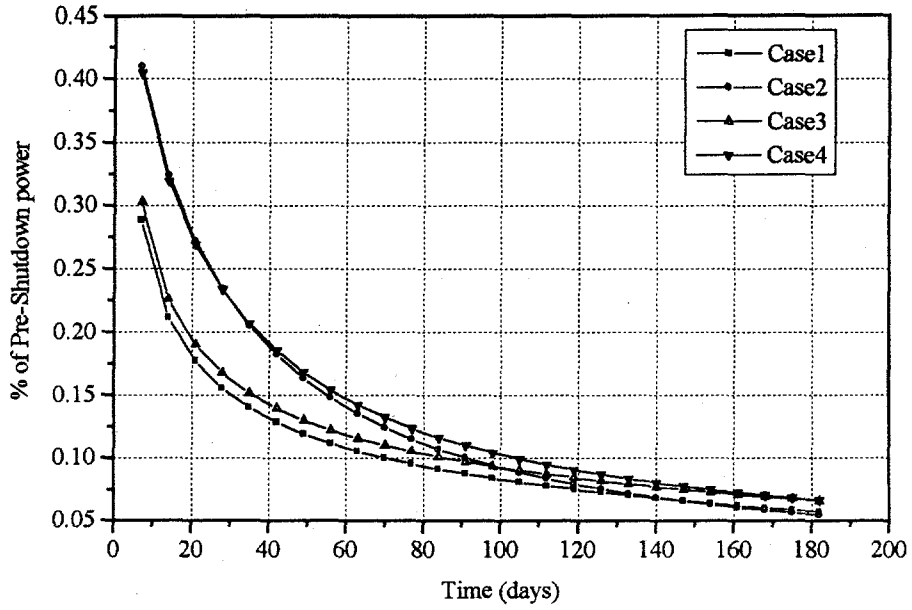

Figure 5.2. Decay heat up to 6 months after shutdown

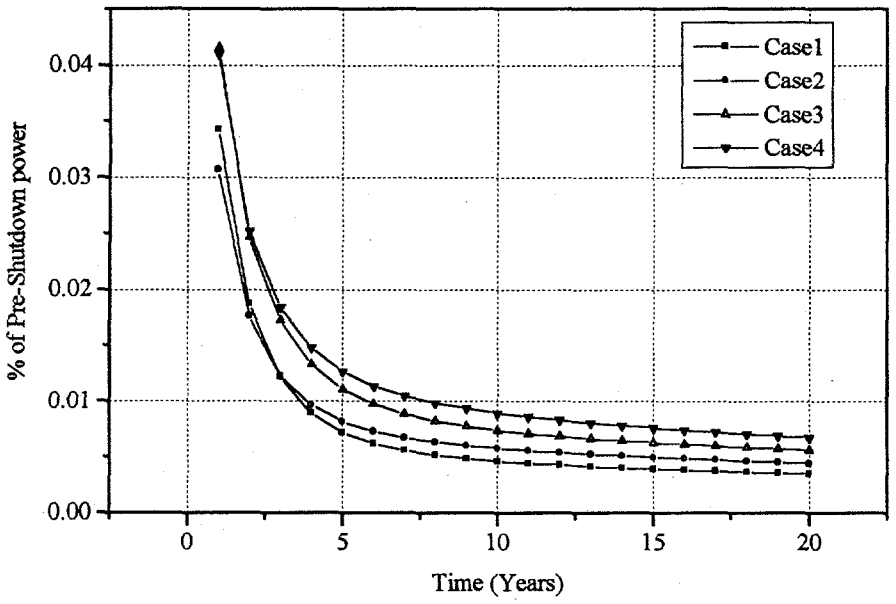

Figure 5.3. Decay heat up to 20 years after shutdown

Finally, decay power generation for the time period of 20 years, i.e., time that the fuel may spend in the fuel pool, affecting the spent fuel pool cooling system, is plotted in Figure 5.3.

The thorium-based fuel exhibits higher decay heat (by approximately 40%) which has to be considered for the spent fuel pool cooling design evaluation.

The decay heat data from the ORIGEN2 calculations confirm the expectations that thoria-urania fuels generate more decay heat than the UO_2 fuel. The differences are not so substantial to significantly affect capabilities of heat removal systems in the currently operating reactors, since the RHR systems are designed to remove power levels higher than those occurring long after shutdown where major differences appear. For the advanced reactors with passive decay heat removal, higher decay heat integrated over a long time may appreciably affect water inventories needed to remove decay heat through passive means. Note that the presented results should be regarded as a preliminary estimate since the Origen2.1 library for ThO_2 fuel does not contain accurate one-group cross sections pertaining to the ThO_2-UO_2 fuel – a deficiency, which will be corrected by future physics calculations.

5.2 Impact of different fuel thermal properties

Thorium oxide exhibits higher thermal conductivity, lower density and slightly smaller specific heat capacity. These parameters will affect stored energy and cladding heat up rate during LOCA scenarios. Better thermal conductivity results in smaller fuel centerline temperature. To assess the overall effects of different thermal properties on fuel performance during LOCA, a computer model of the transient behavior of the fuel rod section with maximum heat load including the gap and the cladding was developed. The one-dimensional heat conduction equation in cylindrical coordinates is solved numerically using the method outlined in the RETRAN computer code manual [McFaden et. al., 1984]. The geometrical and initial data used in the simulation are listed in Table 5.2.

Table 5.2 Data used in the fuel pin transient model

Item	Value
Fuel rod outer diameter (mm)	8.19
Cladding inner diameter (mm)	8.36
Cladding outer diameter (mm)	9.50
Pin linear heat rate prior to disturbance (kW/m)	28.0
Pin linear heat rate after disturbance (kW/m)	1.68(6%)
Coolant heat transfer coefficient prior to disturbance (W/m^2-K)	40000
Gap heat transfer coefficient (W/m^2-K)	4000
Coolant bulk temperature (°C)	317
UO_2 theoretical density (g/cm^3)	10.96
ThO_2 theoretical density (g/cm^3)	10.0
Actual/theoretical density ratio	0.945
Weight percent of Th over total heavy metal in ThO_2-UO_2 mixture	0.75

The temperature-dependent thermal conductivity of the ThO_2-UO_2 fuel was evaluated after the Belle and Berman [1982] approach, where thermal conductivity is expressed as a function of temperature and molar fraction of UO_2. The specific heat capacity of ThO_2-UO_2 was evaluated on a volume fraction basis of urania and thoria constituents, where specific heat capacity of ThO_2 was calculated using the equation recommended by Rand [1975]. Properties of UO_2 fuel were evaluated from MATPRO-A, NUREG/CR-5273.

The transient was initiated at time zero by setting the coolant heat transfer coefficient to zero and simultaneously reducing the power to a decay heat level of 6%. This decay power was maintained constant during the entire transient. Since the total decay heat generation during the first 25 seconds (time of interest for initial response of a fuel rod) after shutdown is about the same for each fuel composition (see Figure 5.1), the same decay power level was used for both the urania and urania-thoria fuels.

Figure 5.4 plots the temperature profile during normal operation. The benefit of higher thermal conductivity of thoria-urania fuel is approximately 150 °C on the fuel centerline temperature. This will result in less fission product gas release and swelling, as discussed in Section 4.5. However, a lower swelling rate will lead to an increased gap between the fuel and the cladding and lower interfacial pressure and thus to a higher thermal resistance of the gap. This will be balanced by lower degradation of gap gas thermal conductivity because of lower fission gas release. Therefore, the overall benefit will be different than that shown in Figure 5.4, where the gap resistance was assumed constant for both fuel cases.

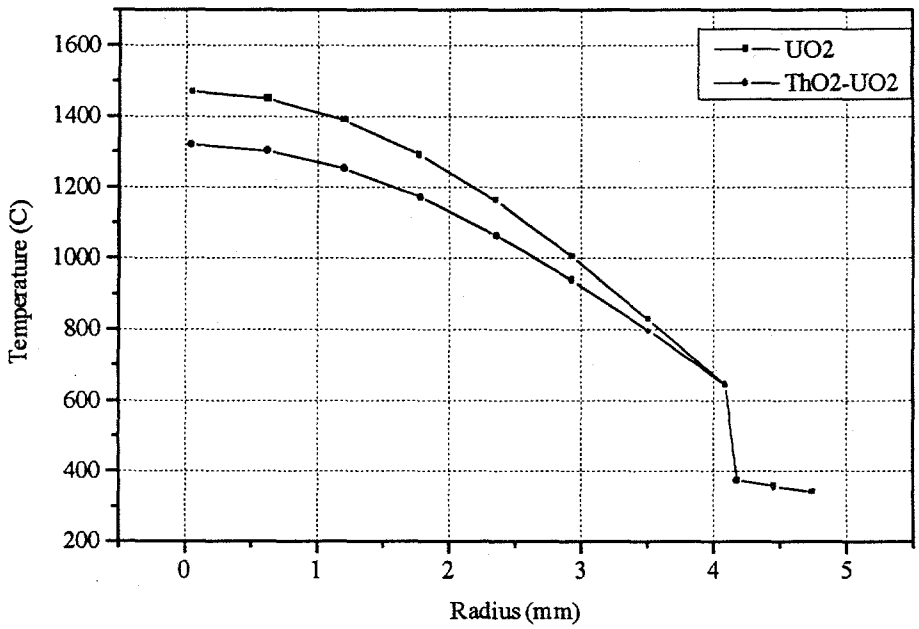

Figure 5.4. Temperature profiles in fuel rod for ThO_2-UO_2 and UO_2 fuel during normal operation

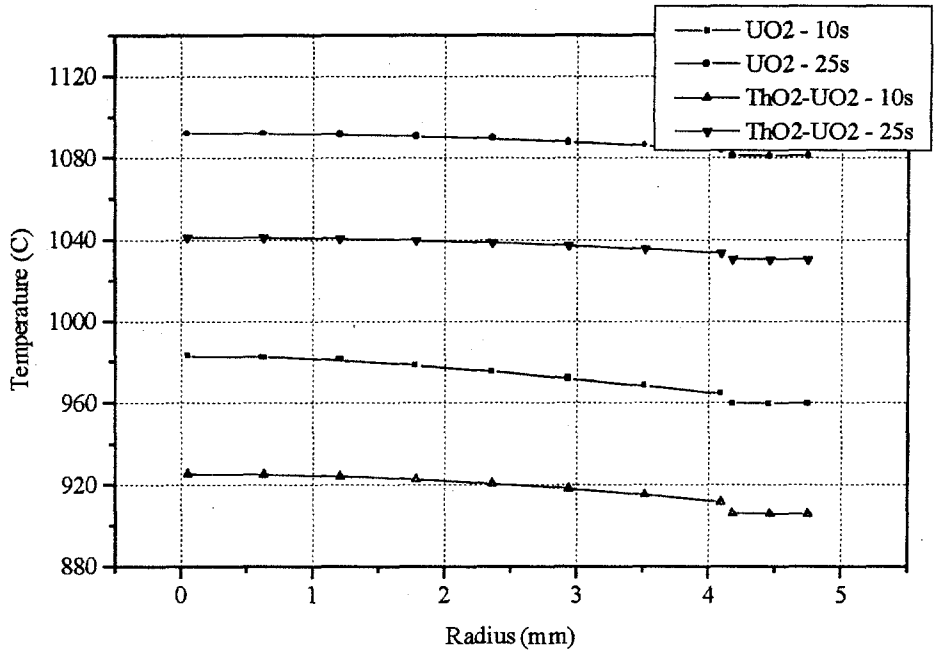

Figure 5.5. Temperature profiles in fuel rod for ThO_2-UO_2 and UO_2 at selected times after LOCA without ECCS

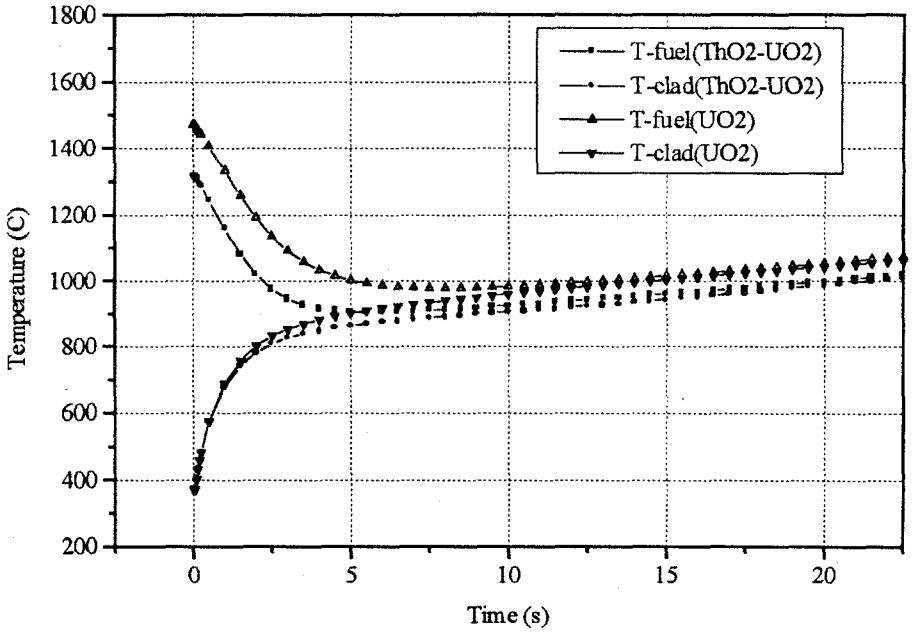

Figure 5.6. Fuel centerline and clad temperature traces following LOCA without ECCS

Figures 5.5 and 5.6 show the transient response of each fuel to the simulated LOCA with no ECCS. The benefit of lower stored energy in ThO_2-UO_2 fuel can be clearly observed. Figure 5.6 indicates that thoria fuel has about five seconds longer time margin to reach the cladding temperature limit of 1200 °C.

5.3 Conclusions

The homogeneous thoria-urania core does not introduce significant changes from the current uranium dioxide core in terms of thermal hydraulic performance*. The main differences having an impact on safety performance were identified to be higher decay heat levels, differences in thermal properties of thoria based fuels, and differences in power peaking. The assessment of the decay heat using the decay heat data from the ORIGEN2 calculations confirm the expectations that thoria-urania fuels generate more decay heat than the UO_2 fuel. The differences are not so substantial to significantly affect capabilities of heat removal systems in the currently operating reactors, since the RHR systems are designed to remove power levels higher than those occurring long after shutdown where major differences appear. For the advanced reactors with passive decay heat removal, higher decay heat integrated over a long time may appreciably affect water inventories needed to remove decay heat through passive means and further evaluation is warranted.

Differences in fuel thermal properties, namely higher thermal conductivity and a slightly lower fuel density and thermal capacity, improve the cladding response in LOCA scenarios due to less stored energy in the fuel pin. The results from an elementary transient model of the fuel pin indicate that cladding of the thoria-urania fuel has approximately 5 seconds longer margin to reach the temperature limit of 1200 °C in a hypothetical event of a total loss of coolant without ECCS than for the case of UO_2 fuel for the same conditions. However, in the future more detailed calculations should include the dynamic effects of lower swelling of ThO_2-UO_2 pellets and lower fission gas release on the heat resistance of the gap between the fuel and the cladding. These effects, which were neglected in this assessment, will improve the prediction of heat resistance of the gap in pretransient conditions and transient analysis.

* Note that thermohydraulic calculations were performed for the beginning of life fuel.

6. Waste Management Issues

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6.1 Introduction

Thorium based fuel has been used in previous reactors. The behavior of the fuel and the resulting waste form has been investigated. While much has been accomplished, a review of the literature indicates the repository behavior of Th requires clarification. Issues which need to be addressed include evaluating isotope production between U and U-Th fuels and determining the behavior of Th based fuel in a repository environment.

Both these evaluations can be related to the recent Yucca Mountain Viability Assessment. In this work, the isotopes ^{99}Tc, ^{129}I, ^{234}U, and ^{237}Np were identified as potential long term radiotoxic hazards. In addition to its proliferation relevance, ^{239}Pu might be an isotope of radiotoxic concern if colloids are found to transport radionuclides from the Yucca Mountain near-field to the far-field. The Yucca Mountain Viability Assessment also targeted colloid formation, secondary phase formation, and waste form solubility as areas of research needs.

In this work, we present some initial studies on the comparison of the formation of actinides and fission products for the two different fuels using the computer program ORIGEN 2.1. The Th containing fuel is found to create more ^{129}I and ^{234}U due to the higher amount of ^{233}U. The U fuel forms more ^{99}Tc, ^{237}Np, and ^{239}Pu from the production of higher actinides. The long term radionuclide inventory for the two different fuels is assessed.

From a chemical standpoint, thermodynamic data on basic oxide, hydroxide, and carbonate species of Th is lacking, In addition, the behavior of Th intrinsic and pseudocolloids is not well understood. Work is presented on estimating the behavior of thorium in the repository near field. A search for existing thermodynamic and speciation data has been initiated. Future research directions and needs are presented.

6.2 Radionuclide Production

Data on the isotopes of concern from a radiological and proliferation standpoint are presented (Table 6.1). These isotopes are produced in differing amount for ThO_2-UO_2 and UO_2 fuel. Naturally, the isotopes ^{99}Tc and ^{129}I are produced from fission. The yields of these fission products from different fissile isotopes are given (Table 6.2). Since the different fuels have different amounts of ^{233}U, ^{235}U, and ^{239}Pu, the concentrations of ^{99}Tc and ^{129}I in the waste will vary. Since the Th containing fuel produces considerably more ^{233}U, the concentration of ^{129}I in the waste will be higher than the U fuel. The U fuel contains more ^{235}U and ^{239}Pu, and will therefore produce more ^{99}Tc.

Table 6.2. Data on the Waste Isotopes of Concern

Isotope	Half-life (years)	Production Route	Daughter
^{99}Tc	2.13×10^5	Fission	^{99}Ru
^{129}I	1.57×10^7	Fission	^{129}Xe
^{234}U	2.46×10^5	^{233}U(n,γ)	^{230}Th
		^{238}Pu decay	
^{237}Np	2.14×10^6	^{237}U decay	^{233}Pa
		^{241}Am decay	
^{239}Pu	2.41×10^4	^{239}Np decay	^{235}U

Table 6.2. Fission Yields (%) for ^{99}Tc and ^{129}I from Different Fissile Isotopes

Fission Product	^{233}U	^{235}U	^{239}Pu
^{99}Tc	4.9	6.1	6.2
^{129}I	1.6	0.54	1.4

The isotope ^{234}U comes from a few routes. The main route is the neutron capture of ^{233}U. While the isotope ^{233}U is fissile with σ_f= 531 b, the neutron capture occurs with σ_γ= 46 b. The other method for the production of ^{234}U is from alpha decay of ^{238}Pu. The ^{238}Pu is formed by:

$$^{237}\text{Np} + \text{n} \rightarrow {}^{238}\text{Np} \xrightarrow{\beta^-} {}^{238}\text{Pu}$$

This route accounts for a much smaller amount of ^{234}U produced since ^{237}Np is in low concentration. For this reason, the Th containing fuel has much more ^{234}U at the end of the time in the reactor. In addition, ^{234}U is also a component of natural U from the decay of ^{238}U by:

$$^{238}\text{U} \xrightarrow{\alpha} {}^{234}\text{Th} \xrightarrow{\beta^-} {}^{234}\text{Pa} \xrightarrow{\beta^-} {}^{234}\text{U}$$

For this reason, the concentration of ^{234}U should noticeably increase with time in the U fuel, but have very little increase in the Th fuel. However, the overall amount of ^{234}U should be greater in the Th fuel.

The two transuranic isotopes in the group, ^{237}Np and ^{239}Pu are produced mainly from the formation of Pu in the fuel. The ^{239}Pu is produced from the neutron capture of ^{238}U, with subsequent ß$^-$ decay of ^{239}U and ^{239}Np. The principal mode for the formation of ^{237}Np is through ^{241}Am by:

$$^{239}\text{Pu} + \text{n} \rightarrow {}^{240}\text{Pu} + \text{n} \rightarrow {}^{241}\text{Pu} \xrightarrow{\beta^-} {}^{241}\text{Am} \xrightarrow{\alpha} {}^{237}\text{Np}$$

From this route, the amount of ^{237}Np in the fuel will increase with time after removal from the reactor due to the decay of ^{241}Am. The ^{237}Np can also be formed from ^{253}U by:

$$^{235}U + n \rightarrow {}^{236}U + n \rightarrow {}^{237}U \xrightarrow{\beta^-} {}^{237}Np$$

As with the neutron capture of ^{233}U, competition from fission will reduce the effectiveness of the ^{237}Np formation from ^{235}U (σ_f= 585 b, σ_γ= 99 b). For both the transuranic isotopes, the concentrations will be greater in the U fuel due to the larger initial amount of ^{238}U.

Table 6.3. Different Fuel Conditions Examined

Case #	Fuel	^{235}U enrichment (%)	Days Run	Specific Power (kW/kg)	Burnup (MWD/kg)
1	UO_2	4.5	1249	37.94	47.4
2	ThO_2-UO_2 (86%Th:14%U)	19.5	1249	41.17	51.4
3	UO_2	8.0	1972	37.96	74.9
4	ThO_2-UO_2 (75%Th:25%U)	19.5	1972	40.77	80.4

For this study, the ThO_2-UO_2 fuel consists of high amount of ^{232}Th and U with 19.5% ^{235}U enrichment. For comparing the amount of considered isotopes formed for the different fuels, four different fuel conditions are considered (Table 6.3). The reason different burnups and specific powers are studied is because cycle energy, not burnup, should be compared. Energy is found by the product of burnup and heavy metal mass. Therefore, density differences between ThO_2 (9.86 g/mL) and UO_2 (10.96 g/mL) must be considered. For 100% U core, the heavy metal density is more than the considered ThO_2-UO_2 mixtures. Thus, this particular reference ThO_2-UO_2 fuel must be driven to a higher burnup to deliver the same energy produced by the fission of standard U fuel. Additionally, to run the core at the same full power level, the specific power should also be increased. The program ORIGEN2 is used to calculate the activity of the examined isotopes as a function of both burnup and cooling time.

For the ORIGEN2 calculations, comparisons can be made between cases 1 and 2 for low burnup and cases 3 and 4 for higher burnup. For the examination of isotopes produced with burnup, the isotope concentrations are expressed in grams/MWD (Figures 6.1 and 6.2). Generally, the last two fuel cases form more radionuclides than the first two cases. The isotopes tend to rapidly accumulate within 10 MWD/kg burnup. The fission products only slightly increase with burnups greater that 10 MWD/kg. For ^{234}U, the Th fuel produces a much greater amount than the U fuel. In the Th fuel case, the amount of ^{234}U increases with burnup. For the U fuel, the concentration of ^{237}Np increases with burnup and is greater when compared to the Th fuel. In all fuel cases the amount of ^{239}Pu decreases with burnup, to a greater extent with the Th fuel. The overall amount of ^{239}Pu is greater in the U fuel.

A comparison of the long-term concentrations of examined radionuclides is presented (Figures 6.3 and 6.4). The radionuclide activities are given in Curies/kg heavy metal. Both fuels produce about the same amount of ^{99}Tc, with a slightly larger level for the U fuel. The differences between the concentration of ^{129}I produced by the different fuel is also small, with a

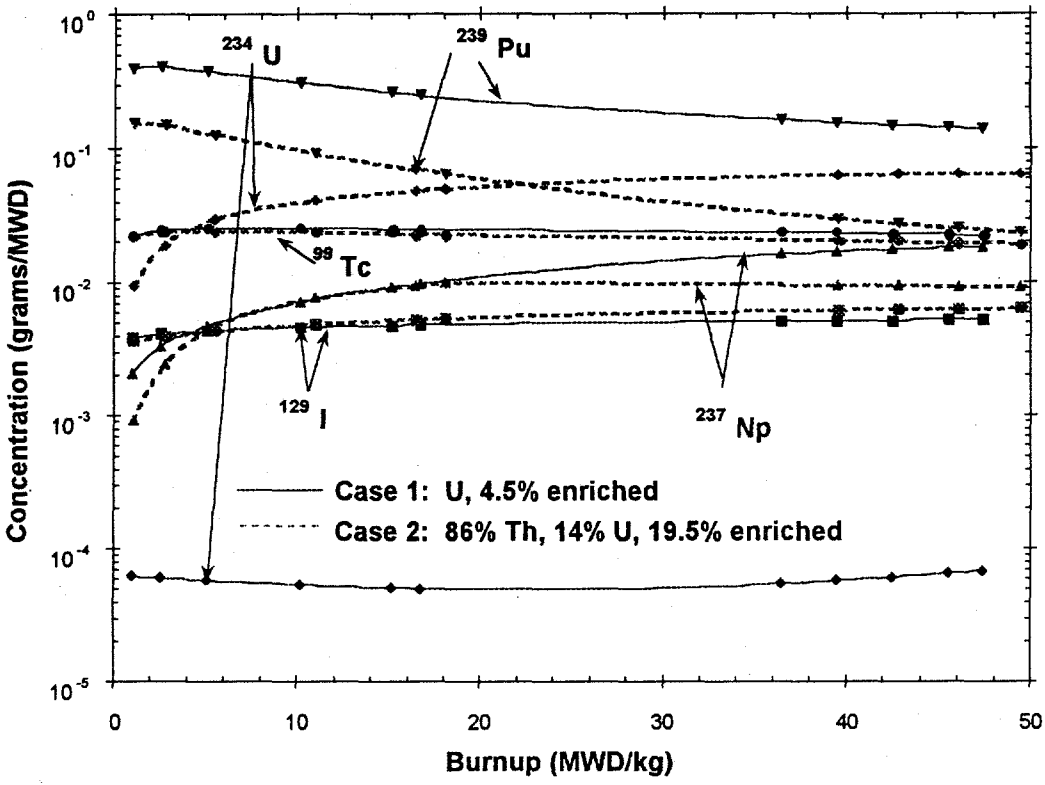

Figure 6.3. Isotopes Produced for Low Burnup Fuel (Cases 1 and 2) at Different Burnups

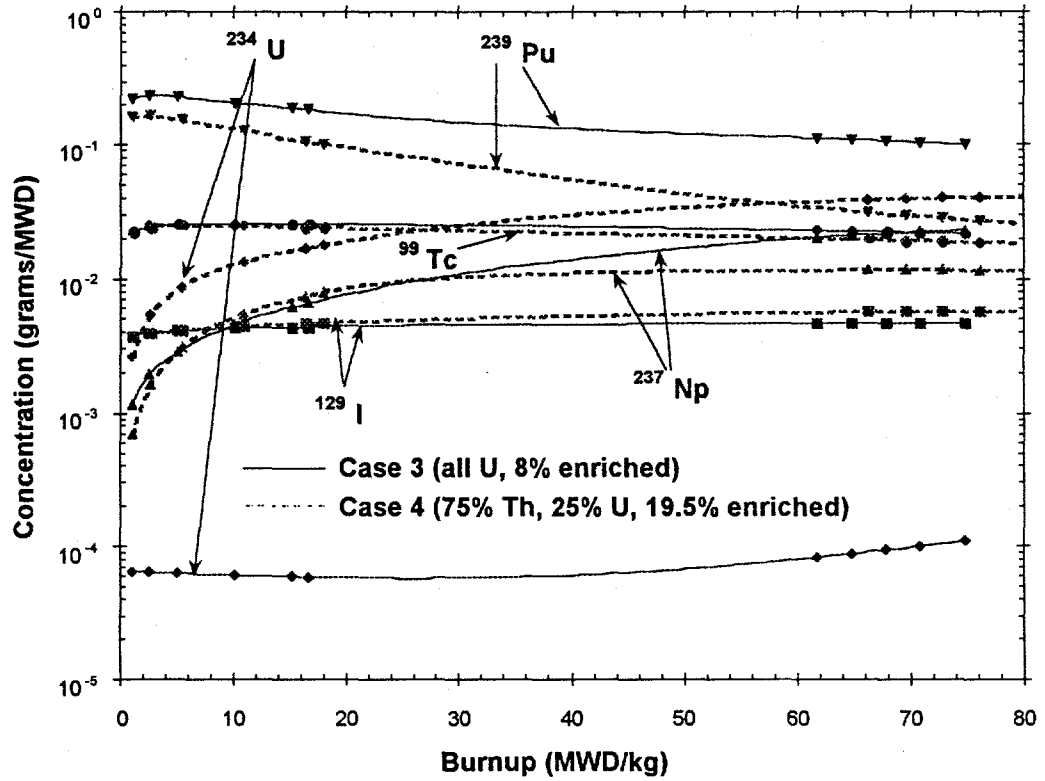

Figure 6.4. Isotopes Produced for High Burnup Fuel (Cases 3 and 4) at Different Burnups

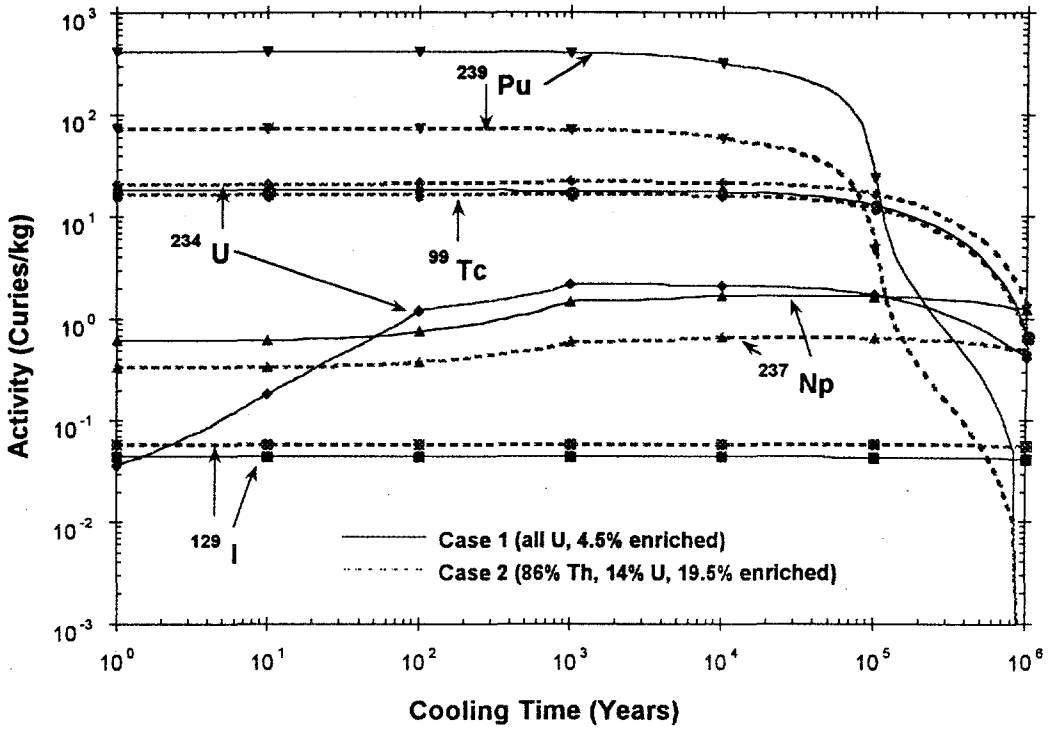

Figure 6.3. Long-Term Radionuclide Concentration of Low Burnup Fuel (Cases 1 and 2)

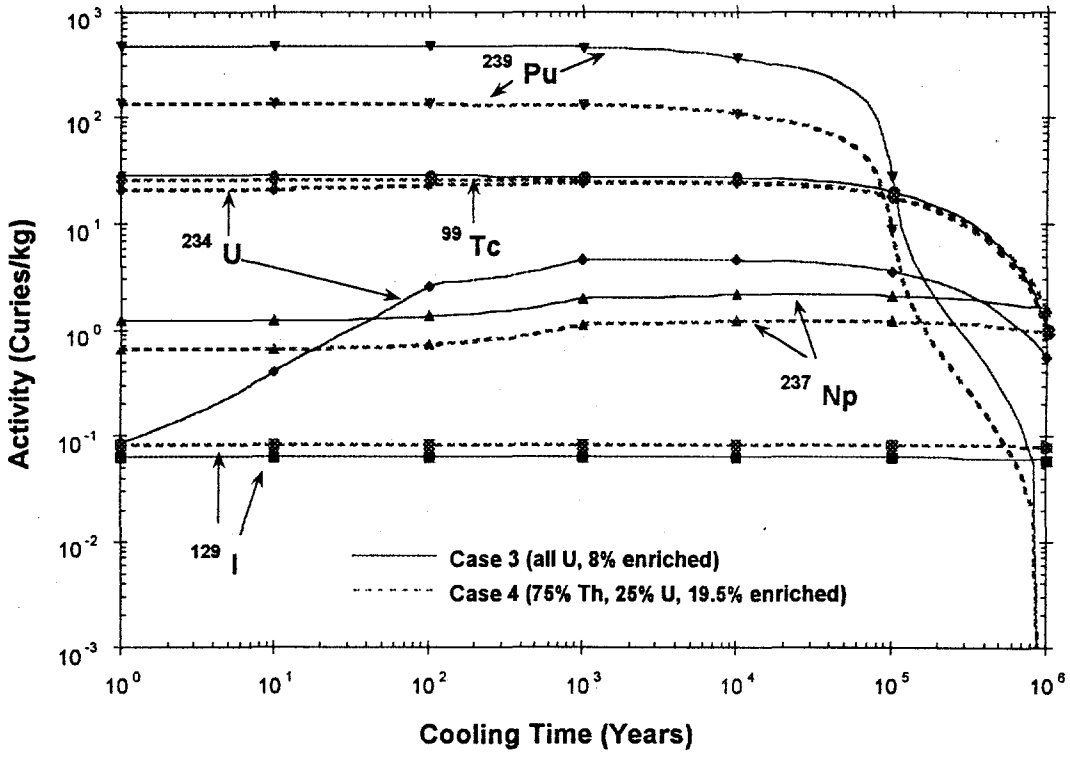

Figure 6.4. Long-Term Radionuclide Concentration of High Burnup Fuel (Cases 3 and 4)

superficially greater amount formed in the Th fuel. However, the activity of ^{129}I is order of magnitudes lower than the other radionuclide up to $6x10^5$ years. A disparity is exhibited in the activity of ^{234}U. Initially, the Th fuel has over 2 orders of magnitude more ^{234}U than the U fuel. However, due to ingrowth from the ^{238}U decay chain, the Th fuel has only around 7 times more ^{234}U at 1000 years. The U fuel has more ^{237}Np, and all cases show an increase of activity after 1000 years. The activity of ^{239}Pu is the largest for all cases up to 10^5 years. The U fuel produces about 5 times of ^{239}Pu under comparable conditions.

6.2.1 Conclusions

Elemental and isotopic composition of the fuel affects critical radionuclide production for the same burnup. The ThO_2-UO_2 fuel produces more ^{129}I and ^{234}U due to the higher amount of ^{233}U. The UO_2 fuel produces more ^{99}Tc, ^{237}Np, and ^{239}Pu. This is related to the greater production of higher actinides from ^{238}U neutron capture. If colloids are found to be a major Pu transport vector, then the use of Th fuel can mitigate this risk. However, further analysis of the Yucca Mountain site is needed to fully address this issue and resolve the hazards posed by the long-lived radioisotopes.

6.3 Solubility of Th under Yucca Mountain Conditions

Since Th is a naturally occurring element, analog studies have been performed. Observations indicate Th migration from a repository should be minimal, and certainly less then oxidized U. Examination of the sorbed Th showed the formation of secondary mineral phases and co-precipitates. This observation indicates the initial dissolution of Th and the formation of new, insoluble minerals in the aqueous phase. This implies the geochemistry of the repository may be important in overall retention of Th. A number of studies showed migration and leaching of Th. In these instances, the presence of colloids was a leading factor in the migration of Th. This observation has powerful implications for the behavior of Th at Yucca Mountain. Recent work has shown the presence of Pu colloids at the Nevada Test Site, which shares geochemistry with the aquifer at Yucca Mountain. If Pu transport is facilitated by colloids, it is reasonable to expect similar behavior with Th. In addition to colloids and secondary mineral phases, geochemical speciation studies show the presence of carbonate and oxyhydroxide species. These studies also indicate the need for evaluating the thermodynamic data for the Th carbonate and oxyhydroxide species.

A great deal of Th thermodynamic data is available. However, species that can be present in environment, particularly colloids, are not sufficiently quantified. Most of the existing data pertains to hydroxide and carbonate quantification and speciation. Solubility experiments have been performed, but thermodynamic data to describe the solubility is not given. The sorption of Th to some surfaces has been examined. However, surfaces which may be encountered in the Yucca Mountain repository environment have not received attention.

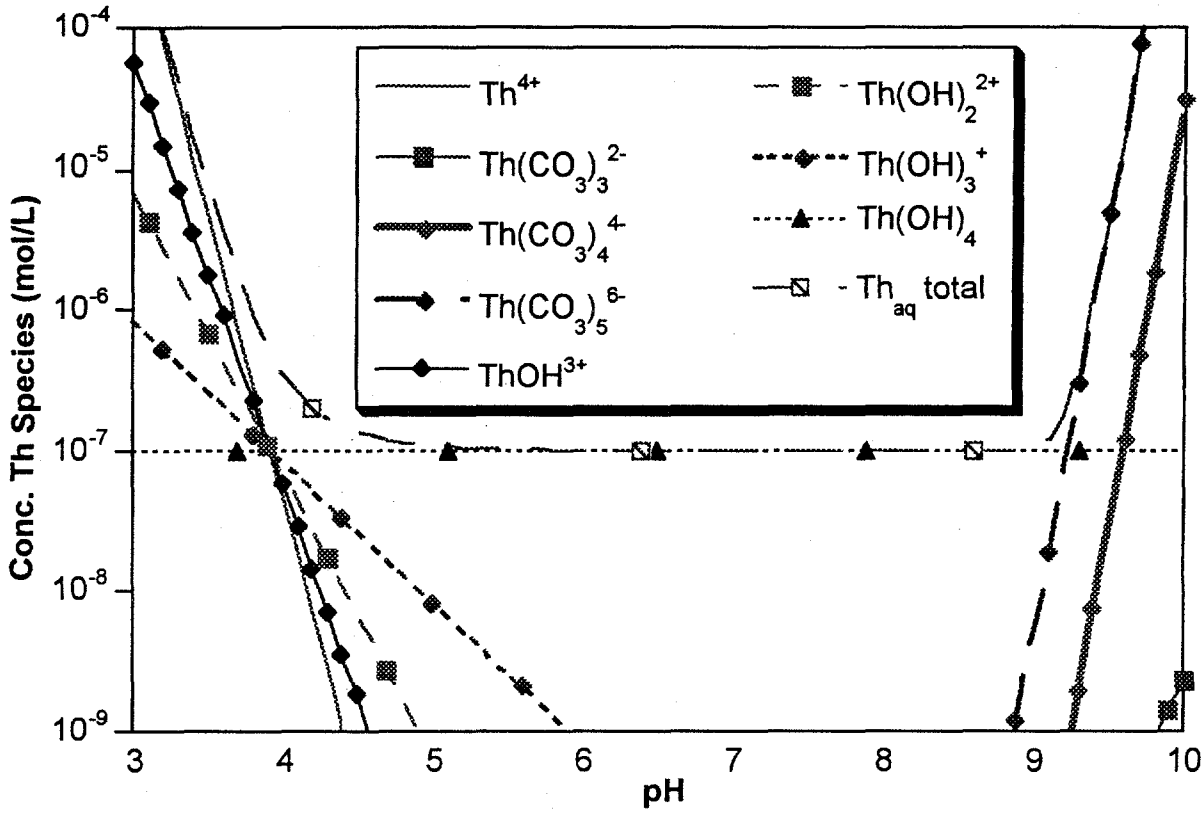

Figure 6-5. Estimated Solution Th species from ThO_2 Under Yucca Mountain Conditions

Table 6.4. Th Thermodynamic Data Used For the Calculations

Th Species	Logß (L/mol)
$ThCO_3^{2+}$	4
$Th(CO_3)_2$	14
$Th(CO_3)_3^{2-}$	21
$Th(CO_3)_4^{4-}$	24
$Th(CO_3)_5^{6-}$	26.2
$ThOH^{3+}$	10
$Th(OH)_2^{2+}$	20
$Th(OH)_3^{+}$	30
$Th(OH)_4$	40.
$Th_2(OH)_2^{6+}$	20
$Th_2(OH)_3^{5+}$	23

In this work, thermodynamic data for Th hydroxides and carbonates is estimated to evaluate the solubility of ThO_2 under Yucca Mountain conditions (Table 6.4). Substituting the estimated constants into the expression for the Th aqueous concentration yields:

$$[Th]_{aq} = [Th^{4+}] + \sum_{x=1}^{4} \beta_{Th(OH)x}[Th^{4+}][OH^-]^x + \sum_{y=1}^{4} \beta_{Th(CO3)y}[Th^{4+}][CO_3^{2-}]^y.$$

The solubility curve can thus be determined (Figure 6-5).

6.3.1 Conclusions

Under Yucca Mountain conditions, ThO_2 solubility should be minimized and U(IV) should oxidize to U(VI). U(VI) is more soluble than Th or U(IV). Dissolution of U can form Th colloids. The ThO_2 may act as a sink for U(VI) and form secondary phases. Additionally, the presence of ThO_2 in the fuel matrix may stabilize the U. Experiments must be conducted in order to obtain Th thermodynamic data for the oxide and mixed uranium species.

7. Outstanding Questions and Future Directions

7.1 Neutronics and Fuel Management

For the area of neutron physics, the following priority items have been identified based upon the findings of the initial scoping studies:

a) CASMO-4 studies of thorium-rich vs. all-uranium assemblies should be made under conditions of heightened realism. In particular, burnable poison and soluble poison should be incorporated. Two cases should be evaluated: Current state of the art burnup at ~ 50 MWd/kg, and advanced capability fuel at ~ 100 MWd/kg. It is expected that thorium performance relative to all-U will improve at the higher burnup. To optimize economic performance and increase fuel integrity margins, the high burnup case should employ annular fuel pellets having 10 vol % central voids. The high burnup case should also be based on n = 4 or 5 batch fuel management (corresponding to 18-24 month cycle length), again to enhance economic performance.

b) Definitive benchmark calculations should be completed on two basic configurations

   1) CASMO-4 vs MOCUP unit cell burnup k_{∞} and isotopic composition for the reference 75% Th, 25% U fuel used in our earlier studies.

   b) CASMO-4 vs MOCUP $1/8^{th}$ assembly burnup k_{∞}, isotopics and pin power comparisons, again for the reference fuel composition.

Our strategy is to validate CASMO-4, which can then be used in conjunction with SIMULATE for whole core calculations in the more distant future.

7.2. Fuel Performance Modeling

The results of the investigations of fuel performance to date demonstrate the value of the following future work:

- The envelope pin input should be upgraded to better represent the actual core operation, especially for the high burnup cases.
- The fission gas release model should be developed to evaluate the Th-U fuel.
- FRAPCON-3 is to be used to validate fuel lifetime analysis (FLA) results and is to be modified to explore improved U-Th oxide fuel calculations.
- Analysis and use of annular fuel pellets should be explored as an option to increase fuel integrity margins and neutron economy.
- Additional physics calculations are required to determined the fission gas production, as well as the power distribution and composition distribution to evaluate the rim effect in thoria fuel.
- Additional review of previous findings regarding U-Th oxide fuels is necessary. This review should include at a minimum more studies of LWBR core results and LWBR-related experiments, and a variety of other topics.

- Continued cognizance of domestic and international research related to these efforts is required.

7.3. Thermal-hydraulic and Safety R & D considerations

In the area of reactor thermal hydraulics, the future effort will be directed at:

- Obtaining core and hot assembly power distribution for the thoria-urania core and performing subchannel analysis to compare DNBR margins with UO2-fueled core. This analysis will be based on the results of neutron physics.
- Performing more detailed calculations that consider changes of thermophysical fuel properties with burnup and the dynamic effects of lower swelling of ThO_2-UO_2 pellets and lower fission gas release on the thermal resistance of the gap between the fuel and the cladding. The alternative of annual fuel pellets will be also investigated.
- Refining decay heat calculations by upgrading the one-group core-average cross section library in ORIGEN2 using MOCUP. This strategy will provide more accurate representation of the thoria-urania mixed core, for which the ORIGEN2 library is not available. Also, the integrated decay energy values will be provided to assess the impact of higher decay power of ThO_2-UO_2 fuel on the magnitude of water inventories in advanced reactors that employ passive means for decay heat removal.

7.4 Waste Management Issues

Future issues to consider are the examination and modeling of the dissolution of Th complexes to understand the long-term behavior of Th containing spent fuel. Such investigations will examine Th oxide, Th hydroxide, and Th carbonate behavior. The oxide is the Th species in the fuel. The carbonate and hydroxide species have importance in environmental systems. Experimental methods include solubility studies, titrations, solvent extraction, UV-Visible spectroscopy, IR spectroscopy, electrochemical methods, and NMR investigations. The interaction of critical radionuclides with Th oxide also merits examination, with attention given to the role of Th containing colloids and the effect of U oxidation on Th colloid formation. Tetravalent metals like Th are known to form colloids in near neutral systems. In addition, Th colloids and precipitates can sorb metal ions, forming both sinks and transport vectors. The goal is to evaluate the behavior of Th-U spent fuel in a repository environment. Results will be described thermodynamically and kinetically to ease incorporation into models and facilitate data comparison. For the models, important terms to be evaluated include solubility constants, stability constants, enthalpy, and entropy. Another subject to consider is the examination of Th-U ceramics. Ceramics are known to be robust and should resist dissolution in the Yucca Mountain repository.

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