R.E. Carter

DRAFT #7

# RESEARCH REACTOR CORE CONVERSION FROM HEU TO LEU FUELS

## SAFETY AND LICENSING GUIDEBOOK

SUMMARY

JUNE 1985



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TECHNICAL COMMITTEE MEETING

Final Review of the Safety and Licensing Guidebook
9 - 11 October 1985
Vienna

AGENDA ITEMS

1. Discussion of Proposed Format

    Comments from absent members

2. Discussion of Summary Volume

    Comments from absent members

3. Discussion of Volumes 1, 2 and 4

4. Discussion of Volume 3, Fuels

    Follow-up report if necessary

5. Miscellaneous Matters

    Schedule
    Second Printing of heavy Water Guidebook
    Distribution and Number of copies
    Petten RERTR Meeting
    Caracas Training Course
    1987 International Conference
    Future Committee Activities

DRAFT #7

June 1985

IAEA-

# RESEARCH REACTOR CORE CONVERSION FROM THE USE OF HIGHLY ENRICHED URANIUM TO THE USE OF LOW ENRICHED URANIUM FUELS

## SAFETY AND LICENSING GUIDEBOOK

## SUMMARY

Prepared by a Consultants' Group,
Coordinated and Edited by the
Safety of Nuclear Institutions Section and Physics Section
International Atomic Energy Agency

Installations

A Technical Document Issued by the
International Atomic Energy Agency, Vienna

# FOREWORD

In view of the proliferation concerns caused by the use of highly enri-'ed uranium (HEU) and in anticipation that the supply of HEU to research and test reactors will be more restricted in the future, this guidebook has been prepared to assist research reactor operators in addressing the safety and licensing issues for conversion of their reactor cores from the use of HEU fuel to the use of low enriched uranium (LEU) fuel.

Two previous guidebooks on research reactor core conversion have been published by the IAEA. The first guidebook (IAEA-TECDOC-233) addressed feasibility studies and fuel development potential for light-water-moderated research reactors and the second guidebook (IAEA-TECDOC-324) addressed these topics for heavy-water-moderated research reactors. This guidebook addresses the effects of changes in the safety-related parameters of mixed cores and the converted core in order to obtain the necessary permission or license for changing the reactor fuel.

This guidebook has been prepared and coordinated by the International Atomic Energy Agency, with contributions volunteered by different organizations. The IAEA is grateful for these contributions and thanks the experts from the various organizations for preparing the detailed investigations and for evaluating and summarizing the results.

## CONTRIBUTING ORGANIZATIONS

| | | |
|---|---|---|
| Argonne National Laboratory | ANL | United States of America |
| Athens Univ. of Agr. Sciences - Physics Lab. | P.L. AUAS | Greece |
| Atomic Energy Research Establishment | HARWELL | United Kingdom |
| Australian Atomic Energy Commission | AAEC | Australia |
| Babcock & Wilcox | B&W | United States of America |
| Chalk River Nuclear Laboratories | CRNL | Canada |
| Comisión Nacional de Energía Atómica | CNEA | Argentina |
| Commissariat à l'Energie Atomique | CEA | France |
| Compagnie Pour l'Étude et la Réalisation de Combustibles Atomiques | CERCA | France |
| Eidg. Institüt für Reaktorforschung | EIR | Switzerland |
| EG&G - Idaho | EG&G | United States of America |
| GA Technologies Inc. | GA | United States of America |
| GEC Energy Systems, Ltd. | GEC | United Kingdom |
| GKSS-Forschungszentrum Geesthacht GmbH | GKSS | Federal Republic of Germany |
| Greek Atomic Energy Commission | GAEC | Greece |
| International Atomic Energy Agency | IAEA | |
| Internationale Atomreaktorbau GmbH | INTERATOM | Federal Republic of Germany |
| Japan Atomic Energy Research Institute | JAERI | Japan |
| Joint Research Centre - Petten Establishment | JRC - Petten | Netherlands |
| Junta de Energía Nuclear | JEN | Spain |
| Kernforschungszentrum Karlsruhe | KfK | Federal Republic of Germany |
| Kyoto University Research Reactor Institute | KURRI | Japan |
| Netherlands Energy Research Foundation | ECN | Netherlands |
| NUKEM GmbH | NUKEM | Federal Republic of Germany |
| Oak Ridge National Laboratory | ORNL | United States of America |
| Österreichisches Forschungszentrum Seibersdorf | ÖFZS | Austria |
| Risø National Laboratory | RISØ | Denmark |
| Safety and Reliability Directorate-UKAEA | SRD-UKAEA | United Kingdom |
| Transnuklear GmbH | TN | Federal Republic of Germany |
| United Kingdom Atomic Energy Authority | UKAEA | United Kingdom |
| University of Michigan - Ford Nuclear Reactor | FNR | United States of America |
| U.S. Department of Energy | USDOE | United States of America |

## PARTICIPATING ORGANIZATIONS

| | | |
|---|---|---|
| Gesellschaft für Reaktorsicherheit | GRS | Federal Republic of Germany |
| Swiss Nuclear Safety Division | HSK | Switzerland |
| United States Nuclear Regulatory Commission | USNRC | United States of America |

# PREFACE

This guidebook has been prepared to assist research reactor operators in addressing the safety and licensing issues for conversion of their reactor cores from the use of highly enriched uranium (HEU) fuel to the use of low enriched uranium (LEU) fuel. It contains a wide variety of information on the analyses that are required to prepare an amendment to a Safety Report, examples of different analyses and licensing documents, the fuels and testing data that are available, and operation of the reactor facility.

Two types of core conversions are considered: (1) conversions where only the fuel and reactor core are changed and (2) conversions where other major modifications are made to accommodate the fuel change.

In most cases, reactor operators will probably choose to convert to LEU fuel without changes in fuel element dimensions or core configurations, thereby minimizing the changes in the safety-related parameters of the facility. Many facilities may operate with an interim core using both HEU and LEU fuel until an equilibrium core with LEU fuel is established. Studies in this guidebook provide assistance in determining the principles and key safety parameters with mixed cores.

The book is organized into a Summary and four Volumes of Appendices. The following table provides a brief overview of the organization.

| Topic | Summary Chapter(s) | Detailed Information Volume | Detailed Information Appendix |
|---|---|---|---|
| 1. Licensing | 1 | - | - |
| 2. Analysis | 2-6 | 1 | A-F |
| 3. Analytical Verification | 7-8 | 2 | G-H |
| 4. Fuels | 9-11 | 3 | I-K |
| 5. Operations | 12-14 | 4 | L-N |
| 6. Safeguards and Physical Protection | 15-16 | - | - |

## 1. Licensing

In special [illegible] S-1

Conversion of a reactor core from the use of HEU fuel to the use of LEU fuel will generally require ~~either~~ a license amendment ~~or~~ a new license. Within this range, there is a wide spectrum of possibilities that depend on the reactor characteristics and on the national ~~situation~~. Reference can be made to IAEA Safety Series No. 35 (1984 Edition) for a description of the technical information to be included in a full Safety Report.

The application for permission to change the reactor fuel should require an amendment to the current Safety Report and/or Safety Specifications addressing only those factors affected by the changes in fuel composition and/or core configuration. The analyses should also include the effect of any changes to the other systems necessitated by the new fuel. If more stringent regulations or requirements have been introduced since the original license was granted, the change to LEU fuel may involve a repetition of the entire licensing procedure.

Chapter 1 of the Summary is intended to assist the reactor operator in preparing an amendment to his Safety Report and/or Safety Specifications for submission to his licensing authority. The scope is restricted to only those parts of a Safety Report which are considered likely to be directly affected by core conversion.

2. Analysis

Chapters 2-6 of the Summary and Appendices A-F (Volume 1) contain example analyses and results showing the differences that can be expected in the core safety parameters and the radiological consequences of hypothetical accidents. Also discussed are methods for preventing loss-of-coolant accidents. There are seven examples of licensing documents related to core conversion and two examples of the methods for determining power limits for Safety Specifications.

3. Analytical Verification

Chapters 7-8 of the Summary and Appendices G-H (Volume 2) contain the results of a safety-related benchmark problem and comparisons of calculated and measured data. Both of these approaches are very useful in ensuring that the calculational methods employed in the preparation of a Safety Report are accurate. As a first step, it is recommended that reactor operators/physicists use their own methods and codes to calculate this benchmark problem, and to compare the results of calculations with measurements in their own reactor or in one of the reactor for which measured data is available in Appendix H.

4. Fuels

The information and test data on reduced enrichment fuels that are available as of October 1985 are summarized in Chapters 9-11. Detailed data on the fuel materials, irradiation tests, and post-irradiation examinations (PIE) can be found in Appendices I and J of Volume 3. Appendix K of this volume contains detailed examples of fuel specifications and inspection procedures that are very useful in procuring new fuel.

5. Operations

Chapters 12-14 of the Summary and Appendices L-N (Volume 4) contain useful information and recommendations for startup procedures and experiments with reduced enrichment fuels, and discussions of the experiences of several reactor operators with mixed cores, ~~composed of different types of~~ fuel. Also included is information on the transportation of fresh and spent fuel, spent fuel storage, and reprocessing.

6. Safeguards and Physical Protection

Chapters 15 and 16 of the Summary contain information prepared by the IAEA on research reactor safeguards and physical protection with various uranium enrichments and fissile inventories.

7. IAEA Assistance

The IAEA can be contacted, through official channels, to provide coordinating assistance between reactor organizations and those laboratories which have offered technical assistance for core conversion studies on specific reactors. If necessary, the IAEA can also provide fellowships to visit those laboratories for joint studies on core conversion.

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For simplicity, the following definitions have been adopted only for this publication. The legal definition of highly enriched uranium is uranium with equal to or greater than 20 wt% $^{235}U$.

HEU - Highly Enriched Uranium ($\geq$ 70 wt% $^{235}U$)

MEU - Medium Enriched Uranium (45 wt% $^{235}U$)

LEU - Low Enriched Uranium (< 20 wt% $^{235}U$)

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APPENDICES

VOLUME 1

ANALYSIS

(Cont.)

# APPENDICES

## VOLUME 2

## ANALYTICAL VERIFICATION



*Self-Limiting Transients in Heavy Water Moderated Reactors

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# APPENDICES

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## FUELS
(Cont.)

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## VOLUME 3

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(Cont.)

# APPENDICES

## VOLUME 4

## OPERATIONS

SUMMARY

# CHAPTER 1

## TOPICS TO BE ADDRESSED IN A SAFETY REPORT AMENDMENT FOR CORE CONVERSION

### Introduction

Conversion of a reactor core from the use of highly enriched uranium fuel to the use of low enriched uranium fuel will generally require ~~either~~ a license amendment, or a new license. ~~Within this range, there is a wide spectrum of possibilities~~. The actual requirements for core conversion depend on the reactor characteristics and on the national ~~situation~~. Figure 1 shows two possibile approaches, applicable in different circumstances.

In this regard, the facility operator would be required to submit an amendment to, or a revision of, the Safety Report. In either case, the report should contain technical information proving that the converted facility could continue to be operated without undue risk to public health and safety. The report should demonstrate that the degree of safety in the new reactor core is equivalent to that of the previous core. The information should permit a determination of the adequacy of the evaluations; that is, assurance that the evaluations included are correct and complete and that all the necessary evaluations have been made.

In compiling the report, the various guidelines and relevant regulations should be taken into account. Different regulations and requirements exist in the various countries, but attention is drawn to IAEA Safety Series No. 35 - "Safe Operation of Research Reactors and Critical Assemblies, Code of Practice and Annexes" (1984 Edition).

This guidebook on "Safety Aspects of Research Reactor Core Conversion" is restricted to only those parts of a Safety Report which are considered likely to be directly affected by core conversion.

This chapter is intended to assist the operator in preparing a Safety Report amendment for submission to his licensing authority. The format adopted follows that presented in Annex A of IAEA Safety Series No. 35 (1984 Edition) and all sections of the Safety Report are listed with the same numbering system. When an amendment of the section is considered likely because of the conversion, the relevant subsections are also listed and discussed. Since not all the subsections are quoted, the numbers of those quoted are not always consecutive.

*Comments in italics indicate the scope of possible amendments.*

The attachment to this chapter contains a brief description of the tasks which must be performed to prepare the information required for the Safety Report amendment. It is divided into two parts, with the first part covering the tasks related to analysis and planning and the second part covering the tasks related to the properties and qualification of the fuel.

Depending upon the extent of the modifications being made, amendments to the following supporting documents for the Safety Report may also be required:

- supplementary plans, drawings and descriptions of the plant and its components,
- information on the safeguards and physical ~~security~~ of the plant,
- safety or technical specifications ~~not included in the Safety Report~~,
- information on the environmental impact,
- information on the provisions for spent fuel disposal (including storage, transfer, reprocessing, burial or other).

| | | |
|---|---|---|
| Kind of Alteration | Core Conversion Only | Core Conversion with Modifications* |
| Possible Hardware Alterations | New fuel, new core including control rods, new reflector | Major modifications of systems and components in addition to new fuel including control rods, new reflector |
| Documents for Application | Amendment to Safety Report, revision of relevant chapters including fuel specification and fuel qualification | Completely revised Safety Report including fuel specification and fuel qualification |
| Licensing ++ Procedure | - Application for a license amendment<br>- Examination of licensing documents by an expert body<br>- Approval of Amendment to Safety Report | - Application for a new license<br>- Examination of the documents by all relevant bodies<br>- Approval of licensing documents by the authority |

*These modifications may be required by (a) a wish to upgrade the facility, (b) special facility features which require modifications in order to implement the conversion, (c) inadequacy of current safety features, which would require modifications even without core conversion. Such inadequacy might be revealed by a review of the reactor safety considered appropriate in view of the national requirements, the age of the facility, the period of time since the previous review, the extent of changes to the facility since that review, the state of existing documentation and the operating license.

Fig. 1. Example of Possible Licensing Steps

++ In some cases a license amendment might [illegible]

CHAPTERS OF A SAFETY REPORT INDICATING THOSE LIKELY TO REQUIRE AMENDMENT FOR CORE CONVERSION

## 1. INTRODUCTION AND GENERAL DESCRIPTION OF THE FACILITY

The first chapter of the Safety Report should present an introduction to the report and a general description of the facility.

*The general description need only indicate the extent of changes to the facility and the reasons for the changes. References should be given to relevant supporting documents, drawings, etc. for the modified plant. The legislative and other requirements relevant to the reactors should be identified.*

## 2. SITE CHARACTERISTICS

*Site characteristics are not expected to be affected by core conversion.*

## 3. SAFETY PRINCIPLES AND GENERAL DESIGN CRITERIA

This chapter of the Safety Report should identify, describe, and discuss the safety principles of the architectural and engineering design of the structures, components, equipment, and systems important to safety.

*The safety principles, design criteria, and mechanical design methods are not expected to be affected by core conversion.*

## 4. BUILDINGS AND STRUCTURES

*Generally, there is no effect expected here for core conversion.*

## 5. REACTOR

In this chapter of the Safety Report, the applicant should provide an evaluation and supporting information to establish the capability of the reactor to perform its safety functions throughout its design lifetime with the new core under all normal operational modes (including both transient and steady state) and accident conditions. This chapter should also include information to support the analyses presented in Chapter 16, Safety Analyses.

### 5.1 Summary Description

A summary description of the mechanical, nuclear, thermal and hydraulic designs of the various reactor components, including the fuel, reactor vessel internals, and reactivity control systems, should be given. The description should indicate the independent and interrelated performance and safety functions of each component. A summary table of the important design and performance characteristics should be included. A tabulation of analysis techniques used, load conditions considered and names of verified computer codes should be provided.

*Directly affected by core conversion will be the density and enrichment of the uranium and, possibly, the chemical composition of the fuel meat. Also, in some cases, fuel element design, control rod design, and reflector element design will be affected.*

### 5.2 Fuel System Design

The design bases for the mechanical, chemical, and thermal design of the fuel system that can affect or limit the safe operation of the facility should be presented. The description of the fuel system mechanical design should include the following aspects: (a) mechanical design limits such as those for allowable stresses, deflection, cycling, and fatigue; (b) capacity for fuel fission gas inventory and pressure; (c) a listing of material properties; and (d) considerations for radiation damage, materials selection, and normal operational vibration.

The chemical design should consider all possible fuel/cladding/coolant interactions. The description of the thermal design should include such items as maximum fuel and cladding temperatures and fuel cladding integrity criteria. Details of fuel qualification should be included.

The selection of design bases, actual design description, design evaluation, and the proposed fuel testing and inspection plan should be discussed.

Detailed specifications on mechanical, chemical, and thermal design could be presented in additional reports. The Safety Report should only include principal details necessary for understanding nuclear design and safety analysis. Of special interest are experimentally verified limitations for the chosen fuel element design.

*The technical description of the fuel elements will be changed owing to the lower enrichment and the probable higher fissile content of each fuel element. The physical properties of the fuel plates, e.g. heat capacity and thermal conductivity will be changed even if the geometrical shape and cladding thickness are preserved.*

### 5.3 Nuclear Design

The design bases, design description, and analysis for the nuclear design of the fuel and reactivity control systems should be provided and discussed, including nuclear and reactivity control limits such as excess reactivity, control rod insertions, fuel burnup, negative reactivity feedback, core design lifetime, fuel replacement philosophy, reactivity coefficients, stability criteria, maximum controlled reactivity insertion rates, control of power distribution, shutdown margins, stuck rod criteria, rod speeds, chemical and mechanical shim control, burnable poison requirements, and backup and emergency shutdown provisions.

*A comparison of the old and new nuclear design should be made in addition to the required new calculations for presentation of the main changes to the licensing authorities. The main purpose for this comparison is to gain better understanding of the safety analysis.*

margins

*Core conversion may require re-consideration of the fissile loading required in initial and subsequent cores and re-statement of reactivity values and shutdown ~~reactivity~~, power peaking factors and burnup data for the various operational states during the life of the core. Reflector changes should be discussed because, in some cases, they may influence core-size, burnup, form-factors, etc. Plutonium produced in the fuel may also need to be considered, even though the quantities are expected to be small.*

5.4 Thermal and Hydraulic Design

The design bases, design description, and analysis for the thermal and hydraulic design of the reactor and core coolant system should be provided, including such items as maximum fuel and clad temperatures, critical heat flux ratio (at rated power, at design overpower, and during transients), flow velocities and distribution control, coolant and moderator voids, hydraulic stability, transient limits, fuel cladding integrity criteria, and fuel assembly integrity criteria.

There should be a discussion of the testing and verification techniques to be used to ensure that the planned thermal and hydraulic design characteristics of the core and the reactor coolant system have been provided and will remain within required limits throughout core lifetime.

*If there are no changes in fuel element geometry, core-size or power peaking factors, only the thermal design needs discussion; otherwise, both thermal and hydraulic design should be discussed. A comparison with the old design is recommended.*

5.5 Reactor Materials

*A list of the materials and their specifications for each component of the control rod system and for reactor internals which have undergone changes should be presented. The effects of changed irradiation conditions on the materials should be discussed if significant.*

5.6 Mechanical Design of Reactivity Control Systems

Information should be presented to establish that the control rods (dimensions and materials) and the control rod drive system, which includes the essential ancillary equipment and systems, are designed and installed to provide the required functional performance and are properly isolated from other equipment. Additionally, information should be presented to establish the bases for assessing the combined functional performance of all the reactivity control systems (including insertion times) to mitigate the consequences of anticipated transients and postulated accidents.

*If the control rod design has changed, information should be presented to establish that the control rod drive system, which includes the essential ancillary equipment and hydraulic systems, is still able to provide the required functional performance (including insertion times). Details of the testing programme and the results of the tests should be given.*

6. REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

This chapter of the Safety Report should provide information regarding the reactor coolant system and systems connected to it. Evaluations, together with the necessary supporting material, should be submitted to show that the reactor coolant system is adequate to accomplish its objective and to maintain its integrity under conditions imposed by all foreseeable reactor behaviour, either normal or accident conditions. The information should permit a determination of the adequacy of the evaluations. Evaluations included in other chapters which have a bearing on the reactor coolant system should be referenced.

*If the geometrical shape of the fuel elements is changed as in Section 5.4, the primary cooling circuit performance may be altered and a new set of normal operation characteristics could result. However, if there are no drastic changes in fuel element geometry or reduction of margins to the safety limits due to higher power peaking factors, no significant changes in the reactor coolant system will occur for the same nominal reactor power level.*

## 7. ENGINEERED SAFETY FEATURES OR BARRIERS

Engineered safety features may be provided to mitigate the consequences of postulated accidents in spite of the fact that these accidents are very unlikely. This chapter of the Safety Report should present information in sufficient detail to permit an adequate evaluation of the performance capability of these features. A listing of the information that should be included is contained in IAEA Safety Series No. 35 (1984 Edition), Annex A, p. 37.

*Normally, the requirement for an Emergency Core Cooling System (ECCS) or the requirements to be met by an existing ECCS will not be affected by core conversion if there are only minor changes in the decay heat levels.*

## 8. INSTRUMENTATION AND CONTROLS

This chapter of the Safety Report should provide information regarding the instrumentation and control systems, including the power regulating systems for reactor control, the reactor protection system, and other engineered safety systems instrumentation. The information provided should emphasize those instruments and associated equipment which constitute the reactor safety system.

*If there are no significant changes in the nuclear and thermal/hydraulic charactersitics of the core (Sections 5.3 and 5.4), it is not expected that core conversion will affect the nuclear instrumentation and control system other than possible re-calibration at initial start-up after conversion.*

*If there are significant changes in the nuclear and thermal/hydraulic characteristics of the core, new trip settings may have to be determined.*

## 9. ELECTRIC POWER

The electric power system is the source of power for the reactor coolant pumps and other auxiliaries during normal operation and for the safety system and engineered safety features during abnormal and accident conditions. The information in this chapter of the Safety Report should establish the functional adequacy of the safety-related electric power systems and ensure that these systems have adequate redundancy, independence, and testability in conformance with current criteria.

*It is not expected that the electric power system will be affected by core conversion unless major plant changes are required.*

## 10. AUXILIARY SYSTEMS

This chapter of the Safety Report should provide information concerning the auxiliary systems included in the facility. Those systems that are essential for the safe shutdown of the reactor or the protection of the health and safety of the public should be identified. The description of each system,

Estimates of the release of radioactive materials (by radionuclide) from each source identified and the subsequent transport mechanism and release path should be provided. Identify planned operations, including experiments and anticipated operational occurrences, that may result in release of radioactive materials to the environment. Consider leakage rates and concentrations of radioactive materials for both expected and design conditions. The bases for all values used should be provided. Describe changes from previous designs that may affect the release of radioactive materials to the environment.

*Changes in core material ~~will~~ may require re-calculation of source terms arising from the fuel elements.*

## 13. RADIOLOGICAL PROTECTION

This chapter of the Safety Report should provide information on methods for radiological protection as required by the IAEA Code of Practice (Safety Series No. 35, 1984 Edition, Chapter 13, p. 16), including estimated occupational radiation exposures to operating and construction personnel and to the public during normal operation and anticipated operational occurrences. It should provide information on facility and equipment design, the planning and procedures programmes, and the techniques and practices employed by the applicant in meeting the standards for protection against radiation.

*It is expected that the only effect would be on the radiation design features.*

### 13.3 Radiation Design Features

In this section, equipment and facility design features such as shielding, ventilation, and area and airborne radioactivity monitoring instrumentation which are intended to ensure that radiation exposures are within the specified requirements and as low as reasonably achievable (ALARA) should be described.

*The radiation level outside water or concrete will only be marginally influenced by changes in core size or reflector. Therefore, the radiation level is not expected to be affected by core conversion unless major changes to the plant are required.*

*Depending upon the detailed reactor design, the activity in the primary circuit components may be very sensitive to delay times of coolant on exit from the core. If the coolant flow rate has been increased at all, the adequacy of the delay system may require reassessment.*

## 14. CONDUCT OF OPERATIONS

This chapter of the Safety Report should provide information relating to the preparations and plans for operation of the facility. Its purpose is to provide assurance that the applicant will establish and maintain a staff of adequate size and technical competence and the operating plans to be followed by the licensee are adequate to protect public health and safety.

*The need to revise procedures and re-train staff should be considered if major changes to the plant are required.*

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*The need to revise procedures and re-train staff should be considered if major changes to the plant are required.*

### 14.6 Security

This section of the Safety Report should describe the plans for physical protection of the facility. It is a general practice in many countries to have these plans described in a separate document (see IAEA Safety Series No. 35, 1984 Edition, Code of Practice, Section 15, p. 19).

*In the separate document, a distinction should be made between physical protection of the plant and of the fuel. The physical protection for the plant (~~fission product inventory,~~ MCA) is not influenced by core conversion. The physical protection requirements of the LEU fuel will be lower in some cases. See Chapter ~~12~~ 16 in the Summary of this Guidebook.*

## 15. COMMISSIONING OR TEST PROGRAMME FOR CORE CONVERSION

This chapter of the Safety Report should provide information on the test programme for structures, systems, components, and design features for the facility. The information provided should address major phases of the test programme, including pre-operational tests, initial fuel loading and initial criticality, low-power tests, and power-ascension tests, if applicable.

*~~Three~~ Some possible methods for core conversion are: ~~recognized:~~*

*1. Gradual conversion over a number of refuelling cycles.*

*2. Complete core change at one shutdown.*

*3. Major plant changes, in combination with core conversion*

*The type of test programme necessary will be different in each case.*

*A start-up programme for verifying the calculations and for educating the reactor operators is necessary. Such a programme is required, especially if mixed cores are used, in order to assure that adequate thermal-hydraulic safety margins and shutdown margins are maintained.*

*Hydraulic and reactor physics tests on a new fuel element design must be performed prior to full-power operation. The performance of all reactor ~~heat removal~~ systems must be verified.*

## 16. SAFETY ANALYSES

The evaluation of the safety of a research reactor should include analyses of the response of the reactor to postulated disturbances in process variables and to postulated malfunctions, failures of equipment, or operator errors. Such safety analyses provide a significant contribution to the selection of limiting conditions for operation, limiting safety system settings, and design specifications for components and systems from the standpoint of public health and safety.

The situations analyzed should include anticipated operational occurrences, off-design transients that induce fuel failures, and postulated accidents of low probability (e.g., the sudden loss of integrity of a major component). The analyses should include an assessment of the consequences of an assumed fission product release that would result in potential hazards not exceeded by those from any accident considered credible.

*Besides fuel element design, core design, and fuel reliability, the accident analysis for core conversion is most important. An adequate safety margin must be demonstrated and ~~in some cases, only a marginal reduction of the existing margins could be accepted~~. In many cases the conversion of the fuel only may not require new licensing procedures for the conversion. The accident analysis must be discussed very carefully for the LEU fuel to ensure that current safety requirements are satisfied. If possible, the analyses for the LEU and HEU designs should be compared in order to have a clear final resume of all changes~~, and the overall increase or decrease of the risk~~.*

*Reactivity transients, loss-of-coolant and loss-of-flow accidents must be reconsidered and may need to be reanalyzed owing to possible changes in fuel element design and the response of the fuel to accident conditions and possible changes in temperature feedback coefficients, absorber worths, and thermal-hydraulic characteristics. The ~~probability~~ of fission product release must be revaluated.*

*The scope of the work depends on the magnitude of the changes to the core and the system.*

potential

*Examples of ~~representative~~ accidents~~, the analysis of which~~ may be affected by the change to LEU fuel, are listed in Table 1. Re-analysis of these accidents should take into account the changes in the reactor characteristics after conversion to LEU fuel. The nature and extent expected for some of these changes are summarized in Chapters 2 to 4 and are described in detail in Appendices A to D.*

## 17. SAFETY SPECIFICATIONS

Each Safety Report may contain or refer to Safety Specifications that set forth safety limits and safety system settings, limiting conditions for safe operation, surveillance requirements, and administrative and organizational requirements. These are imposed on facility operation for, among other purposes, the protection of the health and safety of the public. The safety specifications should be derived from and be consistent with the safety analysis in Chapter 16.

may

*Revisions ~~must~~ be made to the conditions referring to fuel element loading, temperature and cooling. The conditions on absorber worths may also need to be re-evaluated. Trip limits referring to the core thermal-hydraulics and reactor physics parameters may require revision.*

## 18. QUALITY ASSURANCE

*A quality assurance system and specifications for the new fuel, ~~acceptable to the licensing body, should be agreed~~ between the operator ~~and the fuel fabricator~~.*

*In the case where modifications to the reactor are necessary, the quality assurance in the design, production and installation of the modified systems should be discussed.*

## 19. DECOMMISSIONING

In some Member States, there is a requirement to include in the Safety Report plans for decommissioning the reactor.

*The scope of work depends on the magnitude of the changes to the core and system.*

## TABLE 1

Potential

Be

## EXAMPLES OF ~~REPRESENTATIVE~~ ACCIDENTS THAT MAY ~~NEED TO BE~~ RE-ANALYZED ~~IN A SAFETY REPORT~~ AMENDMENT FOR CORE CONVERSION*

[illegible]

DECREASE IN HEAT REMOVAL BY THE REACTOR COOLING SYSTEM

- Primary pumps failure and flow coastdown
- Flow blockage to coolant channels

REACTIVITY INSERTIONS AND POWER DISTRIBUTION ANOMALIES

- Startup accident giving ramp insertion of reactivity
- Cold water insertion
- Control rod and control rod follower failure
- Fuel loading error
- Flooding or voiding of experimental beam ports, loops, or thimbles
- Failure or withdrawal of an in-core experiment
- $H_2O$ insertion in a $D_2O$ system or vice versa, or loss of $H_2O$ coolant where other moderation is used
- Criticality during fuel handling
- Control system runaway

CHANGES IN INVENTORY OR PRESSURE OF REACTOR COOLANT

- Whole core loss-of-coolant accident (LOCA)

RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

- Local failure/melting of a few fuel plates or rods in core
- Fuel element cladding failure in core
- Fuel element failure during handling incident

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*See also IAEA Safety Series No. 35 (1984 Edition), Annex A, p. 48

**ATTACHMENT**

## SUMMARY OF REQUIRED TASKS

This section briefly describes the tasks that must be performed to provide the information required for a Safety Report amendment for core conversion. The tasks are listed in an order in which they could be logically performed and are divided into two parts. The first part (General Considerations) includes mostly the analytical and planning tasks described in Volume 1 of this guidebook. The second part (Fuel Considerations) includes tasks related to the properties and qualification of the fuels described in Volume 2 of this guidebook.

### A.1 GENERAL CONSIDERATIONS

#### A.1.1 General Design Features

Summary description of reactor design in comparison with the former reactor design.

#### A.1.2 General Design Rules

Statement of the main rules and regulations taken into account in designing modified systems and components.

#### A.1.3 Fuel Management

Summary description of the burnup cycle and the plutonium production. Statement of burnup data (discharge burnup of initial core, subsequent cores, fully-converted cores, and equilibrium core).

#### A.1.4 Power Distribution

Summary description and explanation of the selected fuel element arrangements and modifications; diagram of a representative power density distribution over the core cross-section with explanation, statement of the macroscopic power density distribution and the local power density peaks.

#### A.1.5 Reactivity Balance

Description of the compensation of excess reactivity. Description of the compensation of reactivity changes (e.g. burnable and Xenon poisoning). Statement of representative reactivity equivalents (e.g. for stuck control absorber) and of the maximum reactivity change rate.

Statement of reactivity values of the reactor, of shutdown reactivity for different operational states (cold, zero power, full power) and burnup states.

#### A.1.6 Reactivity Coefficients

Definition of the reactivity coefficients. Statement of the reactivity coefficients of the fuel temperature (Doppler coefficient), of the moderator temperature, moderator density and voids and power. Representation of the dependences (e.g. on operational state, burnup) in diagrams.

A.1.7 Reactor Protection System

Discussion of possible changes in the control rod design and worth, in the calibration of the linear control and safety channels, and in the trip safety settings.

A.1.8 Radiation and Shielding

Summary description of the radiation sources in the reactor core, i.e., gamma radiation and neutron radiation in different energy groups. Statement of the radiation flows in the individual shielding media and dose rates (neutrons, gamma) in radial and axial direction.

Statement of the concentrations of radioactive materials in the reactor coolant; statement of the equilibrium activities (reference values in stationary operation for planning and safety analyses) for the isotopes of the fission products (noble gases, iodine, solid matter) and of the activation products (gases, corrosion products).

A.1.9 Thermohydraulic Design Principles

Statement of the parameters and limits considered in the design (e.g. critical heat flux ratio, thermodydraulic stability, power density in the fuel, power distribution in the reactor core).

Description of the flow circuit, statement of operating pressure, coolant throughput (main and bypass flow), pressure losses, coolant temperatures, cladding temperatures, admissible fuel element power or heat flux density.

A.1.10 Hot-Channel Factors/Peaking Factors

Definition of the factors. Statement of the expected values and limits of acceptability.

A.1.11 Critical Heat Flux Ratio

Definition of the critical heat flux ratio. Statement of its expected value in the hot channel under different operating conditions and the limits of acceptability including diagram and explanation.

Description of the determination of the critical heat flux ratio during regular operation.

A.1.12 Reactor Cooling System

If modifications are necessary (e.g. due to increased core pressure drop), the following should be provided (1) a summary functional description of the individual components, (2) a statement of significant characteristics (e.g. pressure, temperature and throughput), (3) a summary description of the function and design of the recirculating pumps (e.g. speed, capacity, discharge flow), and (4) summary descriptions of material and certification tests and in-service inspections.

A.1.13 Accident Analysis

Calculations (including e.g. assumptions, physical models and mathematical methods, description of the accident course and effects of the accident) of reactor behaviour during accidents identified in Table 1 of of this chapter.

Further specification of the events to be considered (e.g. primary pumps failure, flow blockage, startup accident, different kinds of LOCA etc.) has to be considered on a case by case basis.

A.1.14 Emergency Core Cooling System

If modifications are necessary to an existing ECCS, a description should be provided of the residual heat production, the functions and operation of the ECCS design (e.g. capacity redundancy, spatial separation of the system), and the test possibilities (e.g. functional tests at certain intervals during operation or during a refuelling operation).

A.1.15 Mixed Cores Operation

Where there will be a transitional period with mixed cores of HEU and LEU fuel, the scheme to be adopted for changing the core from HEU to LEU should be described. Safety aspects to be considered include power distribution and peaking, prediction of burnup, and reactivity effects including the worths of control absorbers. Where other aspects such as the hydraulic characteristics of the fuel have changed, the effects of these should also be considered.

A.1.16 Startup Procedures

Description of the pre-startup tests, startup, zero power and power range tests including e.g. chemical and radiochemical measurements, measurements of the radiation level, measurements of the shutdown reactivity and reactivity coefficients, power calibrations.

A.1.17 Operational Procedures

Summary description of changed operating measures if such changes are necessary due to the modifications. Procedures for startup, power operation and calibration, normal and emergency shutdown, decay heat removal, handling and emergency procedures will need revision to the extent that modifications have been necessary.

A.1.18 Handling and Storing of Fuel Elements

If modifications are needed, description should be provided of the storage provisions for both fresh and spent fuel elements, their position and their capacity. Criticality safety considerations should also be described and explained. Description should also be given of the provisions against crash of heavy loads (e.g. fuel element transport cask) and for spent elements, the provisions to detect and monitor leaks.

## A.2 FUEL CONSIDERATIONS

### A.2.1 Maximum Burnup Levels

Discuss the maximum burnup levels with the new fuel. Compare this data with that experienced for the present fuel.

### A.2.2 Thermal Power Density

Discuss the maximum power density expected with the new fuel and compare with the present fuel.

### A.2.3 Geometry

Discuss any geometry differences that may exist when using the new fuel both in standard fuel elements and in control elements.

### A.2.4 Thermal Characteristics

Discuss the thermal conductivity of the fuel, the maximum fuel and clad temperatures, the maximum surface heat flux, the maximum coolant velocity, etc., expected using the new fuel and compare with the present fuel.

### A.2.5 Manufacturing Data

Describe the manufacturing process of fuel and include all necessary data to support the conclusion that the fuel will perform safely.

### A.2.6 Failure History

Discuss the average or projected rejection rate for the new fuel and any reactor failure history or estimates using the new fuel. Compare these values with statistical/historical data for the present fuel.

### A.2.7 Fuel Swelling or Blistering

Discuss the degree of dimensional stability as a function of specific power, burnup, and fuel temperature. Those parameters considered to be design limits should be included as technical specifications and compared with similar values using the present fuel.

### A.2.8 Corrosion Behaviour

Discuss corrosion rates for the fuel cladding under projected typical water chemistry conditions using the new fuel and compare with similar data for the present fuel. Include the basis for and any changes required in water chemistry and surveillance specifications.

### A.2.9 Quality Assurance

Describe the quality assurance procedures to be followed in the design and production of the modified systems and components.

# CHAPTER 2

# SAFETY ANALYSES

## 2.1 Introduction

This summary is based on the work presented in Appendices A and B. Appendicies A-1 and A-2 present the results of safety analyses performed by INTERATOM (FRG) and the Argonne National Laboratory (USA), respectively, for the generic 10 MW reactor based on replacement of the HEU plate-type fuel with LEU plate-type fuel. Appendix A-3 on the other hand presents the results of safety analyses performed by GA Technologies (USA) for replacement of the HEU plate-type fuel with TRIGA LEU rodded-type fuel. Appendix B presents a methodology for probibilistic accident analysis contributed by GEC (UK) and the UKAEA-SRD (UK), although the information does not contain a comparative study of HEU and LEU fuels.

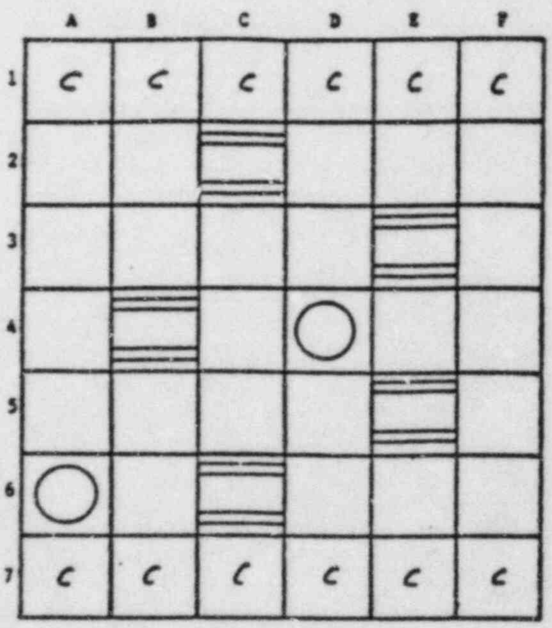
For the calculations in Appendix A, unlike the benchmark studies (Chapter 6), the contributors were free to select boundary conditions for the hypothetical accidents consistent with their own regulatory requirements. For comparison of methods of calculation, reference can be made to Chapter 6 and Appendix F.

## 2.2 Plate-Type Fuels

### 2.2.1 Equilibrium Cores

In Appendices A-1 and A-2, the same HEU fuel element with 23 plates and 280 g $^{235}U$ is studied. For the LEU core, both contributions examine the safety parameters of an equilibrium core and each core of a gradual transition from HEU to LEU fuel. One case uses an LEU fuel element with 20 plates, 1.0 mm-thick fuel meat, and a fissile loading of 446 g and the other uses an LEU fuel element with 390 g $^{235}U$ and the same geometry as the HEU element.

Fig. 2.1 10 MW Reactor Core




The 5 × 6 element core (Fig. 2.1) contains 23 MTR-type fuel elements and 5 control fuel elements. The core is reflected by graphite on two opposite faces and is surrounded water. One flux trap is located near the center of the core and another near an edge.

The first step in the calculations was to compare the operating parameters and safety margins of the HEU and LEU equilibrium cores to ensure that these characteristics were satisfactory before beginning the HEU-to-LEU transition core analysis.

The data include cycle lengths, average $^{235}U$ discharge burnups, nuclear power peaking factors, steady-state thermal-hydraulic safety margins, control rod worths, and shutdown margins.

The cycle lengths are considerably longer in the LEU cores than in the HEU core because the fissile loading of the LEU elements is much larger. The percentage of $^{235}U$ burned in the discharged elements is about the same in the HEU and LEU cases. The thermal-hydraulic safety margins and the shutdown margins in all of the cores are shown to be perfectly adequate to guarantee the safety of the facility.

### 2.2.2 Transient Analyses

The basic kinetics parameters that were computed for the HEU and LEU equilibrium cores are shown below:

| | HEU | LEU |
|---|---|---|
| Prompt Neutron Generation Time $\Lambda$, $\mu s$ | 55 | 40-42 |
| Effective Delayed Neutron Fraction, $\beta_{eff}$ | 0.0076 | 0.0073 |

The reactivity feedback coefficients for the combined effects of moderator temperature and density are nearly the same in the HEU and LEU cores. However, the LEU cores have much larger Doppler coefficients and larger void coefficients as well. The latter two coefficients play an important role in distinguishing LEU fuel from HEU fuel in some of the transient analyses.

Two types of transients are analyzed in both Appendices A-1 and A-2. These are:

- Loss-of-Flow Transients
- Slow Reactivity Insertion Transients

#### Loss-of-Flow Transients

It was assumed that the reactor is operating at its maximum overpower level when the loss-of-flow occurred. The coastdown of the primary flow rate was approximated by an exponential function with a time constant of 1.0s. The trip setting was at 85% of the nominal flow, with a time delay of 200 ms before the shutdown reactivity insertion. The calculations were terminated at a relative flow rate of 15% because then the natural circulation flaps are assumed to open automatically causing a flow reversal.

The results of calculations taken from Appendix A-1 are shown in Table 2.1. Comparison of the HEU and LEU cases shows that the peak clad surface temperature, the coolant outlet temperature and the safety margin against flow instability are almost identical. The only exception is the peak temperature in the fuel meat, which is about 20°C higher for the LEU fuel because of the lower thermal conductivity of this fuel. Similar results were obtained in Appendix A-2, where the peak temperatures at the surface of the clad were 114°C and 113°C in the HEU and LEU cases, respectively.

All of the peak clad surface temperatures computed for this transient are far below the melting temperature of the cladding, which is about 600°C but depends on the specific composition of the alloy cladding material that is used.

Table 2.1 Fast Loss of Flow Transient

| Fuel | HEU | LEU |
|---|---|---|
| Initial Power, MW | 11.5 | 11.5 |
| Initial Flow Rate, $m^3/h$ | 1000 | 1000 |
| Time Constant for Flow Decay, s | 1.0 | 1.0 |
| Flow Trip Point, % | 85 | 85 |
| Time Delay, s | 0.2 | 0.2 |
| Power Level at Scram, % | 107.4 (0.363)* | 106.1 (0.363) |
| Peak Fuel Temperature, °C | 144.1 (0.363) | 167.4 (0.363) |
| Peak Clad Temperature, °C | 141.1 (0.363) | 141.9 (0.363) |
| Peak Outlet Temperature, °C | 80.1 (0.42) | 80.1 (0.42) |
| Min. Bubble Detachment Parameter, $cm^3K/Ws$ | 75.4 (0.38) | 77.3 (0.38) |
| At ~ 15 % Relative Flow: | | |
| Fuel Temperature, °C | 66.7 | 67.5 |
| Clad Temperature, °C | 66.5 | 66.0 |
| Outlet Temperature, °C | 51.1 | 51.2 |

Table 2.2 Control Rod Withdrawal Accident (Power Range)

| Fuel | HEU | LEU |
|---|---|---|
| Reactivity Insertion Rate, c/s | 18.9 | 19.8 |
| Initial Power; MW | 10 | 10 |
| Trip Point, MW | 11.5 (0.775)* | 11.5 (0.765) |
| Flow Rate, $m^3/h$ | 1000 | 1000 |
| Time Delay, s | 0.2 | 0.2 |
| Peak Power, MW | 11.93 (0.975) | 11.81 (0.965) |
| Total Energy Release to Time of Peak Power, Ws | $1.065 \cdot 10^7$ | $1.051 \cdot 10^7$ |
| Total Energy Release beyond 11.5 MW, Ws | $2.338 \cdot 10^6$ | $2.319 \cdot 10^6$ |
| Peak Fuel Temperature, °C | 133.3 (0.975) | 130.5 (0.965) |
| Peak Clad Temperature, °C | 130.0 (0.975) | 158.4 (0.965) |
| Peak Outlet Temperature, °C | 52.9 (1.0) | 52.8 (0.965) |
| Min. Bubble Detachment Parameter, $cm^3$ K/Ws | 114.3 (0.975) | 117.2 (0.965) |

*) Quantities in parentheses indicate time (in seconds) at which values occur

Slow Reactivity Insertion Transient

It was postulated that all control rods are withdrawn from the core with the nominal control rod drive speed (~0.21 cm/s) while the pumps are in operation. This slow reactivity insertion due to inadvertent control rod withdrawal may occur during reactor startup. Cases were analyzed for reactor powers in the startup range (~1 W) and in the power range (~10 MW).

The specific assumptions for the ramp rates, trip settings, and time delays before shutdown reactivity insertion were slightly different in Appendices A-1 and A-2. For example, in Appendix A-1, the ramp reactivity insertion rates were assumed to be 0.189 $/s in the HEU case and 0.198 $/s in the LEU case. In Appendix A-2, the corresponding ramp rates were computed to be 0.16 $/s in the HEU core and 0.14 $/s in the LEU core. However, the conclusions in both Appendices are the same.

In Appendix A-1, the peak temperatures reached at the surface of the cladding were 130°C in the HEU core and 158°C in the LEU core if the inadvertent control rod withdrawal were to occur in the power range (see Table 2.2). In Appendix A-2, the corresponding values were 102°C and 101°C in the HEU and LEU cases, respectively. All of these values are far below the temperature needed to initiate melting of the cladding.

Fast Reactivity Insertion Transient

In Appendix A-2, results are also provided for the fast reactivity insertions necessary to initiate melting of the cladding. No initiating mechanisms for the reactivity insertions are postulated. Validation of the PARET code and the methods used can be found in Chapter 6 and Appendix F-1, where calculations are compared with measurements in the SPERT I series of experiments. Comparisons with the SPERT I experiments have been traditionally used in Safety Reports for research reactors in the U.S.

The calculations were done for step reactivity insertions and ramp reactivity insertions in 0.5s from a power level of 1 W with and without scram at 12 MW. A time delay of 25 ms was assumed before the shutdown reactivity insertion for the cases with scram. The results are shown in Table 2.3.

Table 2.3 Summary of Limiting Reactivity Insertions from a Power Level of 1 W to Initiate Melting of 6061 Alloy Cladding at a Surface Temperature of 582°C for HEU and LEU Equilibrium Cores.

| | Limiting Reactivity Insertion, $ | |
|---|---|---|
| Scram | HEU | LEU |
| | Step Insertions, $ | |
| Yes | 2.3 | 2.9 |
| No | 2.3 | 2.9 |
| | Ramp Insertions, $/0.5s | |
| Yes | 3.3 | 8.1 |
| No | 2.8 | 7.9 |

All of the limiting reactivity insertions are larger in the LEU equilibrium core because of its significant prompt Doppler coefficient and larger void coefficient. Results are also shown in Appendix A-2 for the limiting reactivity insertions for the HEU and LEU cores as a function of the prompt neutron generation time and as a function of the thermal conductivity of the fuel meat.

2.2.3 HEU-to-LEU Transition Cores

Most research reactor operators are planning to convert their cores from HEU to LEU fuel by gradually replacing their HEU elements with LEU elements. Over the years, many reactors have been safely operated with numerous mixed cores composed of elements with different geometries, different fissile loadings, different enrichments, or a combination of these. The same principles and safety considerations apply to the current conversions from HEU to LEU fuel.

The most important safety parameters in plate-type reactors are the shutdown margin, the margin to onset of nucleate boiling (ONB), and the margin to onset of an excursive flow instability. Generally, ONB will occur before an excursive flow instability. The larger the difference in the fissile content of the HEU and LEU elements, the more care must be exercised.

Since the HEU elements in Appendices A-1 and A-2 contain 280 g $^{235}U$ and the LEU elements contain either 446 g or 390 g $^{235}U$, nuclear power peaking will be larger in mixed cores of these elements than in the individual equilibrium cores and the thermal-hydraulic safety margins will be smaller as a result. Reactors which currently operate near the limits of their heat removal systems need to carefully examine the nuclear power peaking in mixed cores if the increase in the fissile content of the LEU replacement elements is as large as those considered here. Shutdown margins will also be smaller in both the mixed cores and in the LEU equilibrium core because the neutron spectrum is harder in the much more highly loaded LEU elements.

Before beginning the neutronics calculations, it is prudent to determine the maximum total nuclear power peaking factor that will yield an acceptable margin to ONB. Since the calculations for the mixed cores are performed sequentially, the adequacy of the margin to ONB and the limiting shutdown margin must be checked after each cycle. If one choice of LEU element positions does not satisfy the safety criteria, others must be tried until a successful solution is found.

Appendices A-1 and A-2 present detailed results for the key operational and safety parameters for each step of a gradual transition from HEU to LEU fuel. In Appendix A-1, either five or six HEU elements are replaced per cycle and the cycle length increases each cycle along with the $^{235}U$ content of the core. The conversion from HEU to LEU fuel is completed at the beginning of the 5th cycle, i.e. after a transition phase of 351 full power days of operation.

Appendix A-2 presents transition core results based on replacement of two HEU elements per cycle. The conversion was completed at the beginning of the 14th cycle after 324 full power days of operation. Only two cycle lengths were used in this analysis. Eight of the first nine mixed cores were run using the cycle length of the HEU equilibrium core and the remaining five cycles were run using the cycle length of the LEU equilibrium core. The core was operated for a longer time during the 14th cycle in order to run down the excess reactivity to a value near that expected for the LEU equilibrium core and to maximize the burnup in two LEU elements that need to be replaced for the next cycle.

Appendices A-1 and A-2 should be consulted for detailed data. All of the safety criteria are shown to be fully satisfactory in each case.

## 2.3 Rod-Type Fuel

Appendix A-3 presents safety analysis results for a 10 MW TRIGA-LEU reactor which uses GA's 16-rod UZrH fuel (45 wt% U) cluster. Figures 2.2 and 2.3 show the general layouts of the core and the fuel cluster.

The 6 × 6 core arrangement contains 30 fuel clusters, either 4 or 5 control rods and either 1 or 2 water-filled flux traps. The coolant and reflector are light water. When fresh, each LEU fuel cluster contains 877 g $^{235}U$ and about 0.8 wt% erbium which serves as a burnable absorber. The desired average $^{235}U$ burnup in fuel clusters discharged from the core is >40%. New fuel clusters are introduced into the center of the core when they are needed.

Data are provided describing the nuclear design characteristics of the core, power peaking, the prompt negative temperature coefficient, and the heat transfer analysis.

Two accident scenarios are analyzed and discussed.

### Loss-of-Flow Accident

When the reactor is shut down from power under normal operation, the main coolant pumps will continue to be operated for a short time until the fuel temperature declines to a near-ambient value. Should the pumps fail or be shut off because of an emergency during full-power operation, the reactor would scram on a loss-of-flow signal. Experiments conducted on other force-flow-cooled TRIGA reactors show that the flow coast-down takes several seconds and then the flow reverses direction to the natural convection mode very quickly and smoothly, with essentially no interruption in the fuel temperature decay rate. Thus, the afterheat from the shut-down reactor will be removed by natural convection following pump failure or emergency shutdown. Data on a flow coast-down in a 14 MW TRIGA reactor are provided that confirm this conclusion.

### Reactivity Accident

In this hypothetical startup accident, the entire rod bank was assumed to withdraw from the full-in position at the normal rate of 10 cm/min constant velocity until a scram occurs. A variety of conditions were also assumed that ensure that the results of the calculations are conservative. These include: (1) flow characteristics representative of natural convection flow induced by removal of accident-generated heat, (2) a fuel temperature scram at a temperature about 40°C above the normal operation temperature of 640°C, (3) a redundant power scram at 12 MW, (4) a 0.2s delay between reaching the scram point and initiation of scram rod movement, (5) the maximum-worth rod does not scram, and (6) the period scram does not operate.

The calculational results show that the peak power reaches about 11.6 MW, but that the reactivity insertion is slow enough that sizeable fuel temperatures are generated before the reactivity insertion is completed. Since the peak power does not exceed the assumed scram point, the ramp continues until the fuel temperature at the thermocouple reaches 680°C. Although there would be bulk boiling in the channel, the fuel temperature remains low enough to preclude damage because a temperature scram would limit the fuel temperature to about 700°C. This is about 240°C below the fuel temperature safety limit (when fuel and clad are at the same temperature) of 940°C.

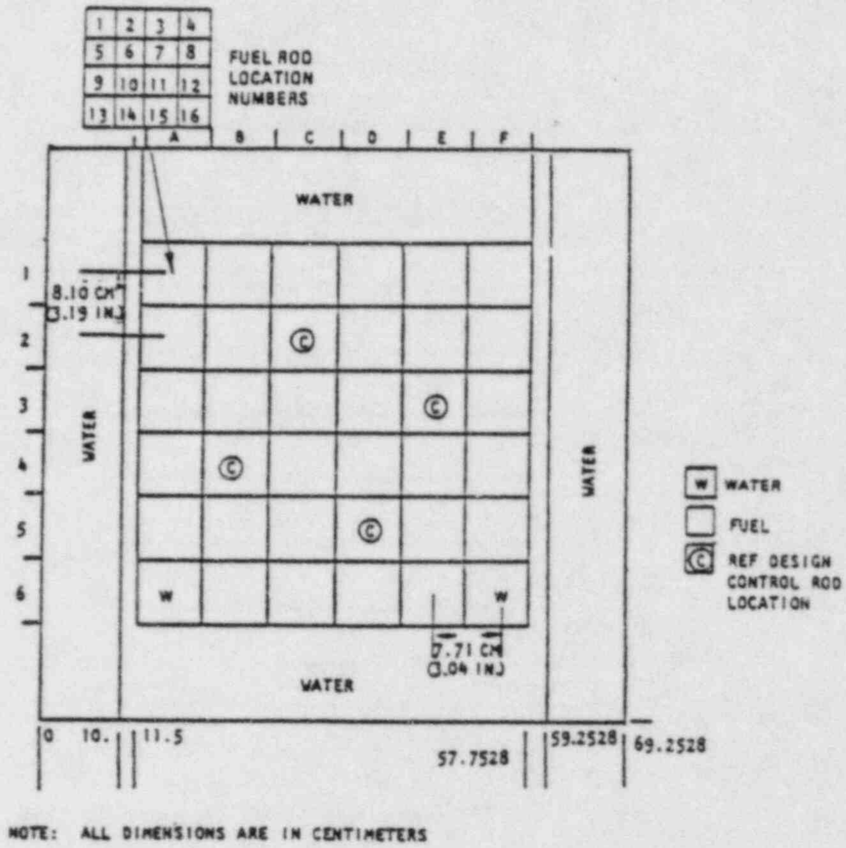



Fig. 2.2 Core Arrangements and Typical Dimensions for 10 MW TRIGA Geometry

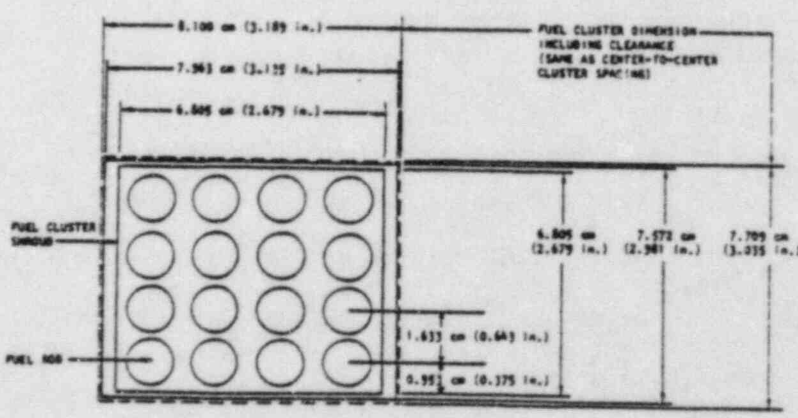



Fig. 2.3 General Layout of 16-Rod Fuel Cluster

## 2.4 Probabilistic Methods for Accident Analysis

Appendices B-0, B-1, and B-2, although not referring in any way to the differences between HEU and LEU, are presented as a guide to probabilistic safety analysis methods.

The use of Probabilistic Risk Assessment (PRA) is recommended for power reactors by several leading licensing authorities (such as the USNRC, the UKNII and some other European authorities), as a supplement to the more traditional, deterministic approaches.

There are in general no mandatory requirements for such studies, and for research reactors the use of PRA techniques (in licensing) has not been widely applied.

The demands on both applicant and licensing authority in terms of total effort and the level of skill are substantial. The decision to use PRA in the licensing process should not be made lightly, particularly when resources and previous experience are limited and perhaps only one research reactor is involved.

Even for a single research reactor, the effort required for a full PRA based treatment can be comparable to that for a major power reactor (see Appendix B-0) while the benefits may be relatively small.

Nevertheless, the value of limited application of probabilistic assessment methods at appropriate stages of design and operation can be substantial in the context of evaluating design options, design changes, and testing and maintenance strategies.

Appendix B-0 gives a brief general review of the background related to the use of PRA in design and licensing and discusses very briefly the requirements for a comprehensive PRA approach to safety assessment and review for licensing purposes. For detailed guidance to these requirements and modelling methods, reference to NUREG CR-2300 is recommended.

Appendices B-1 and B-2 provide illustrative examples of how, with relatively modest effort, supplementary quantitative ideas about system performance can be generated during redesign and/or re-licensing of a reactor.

Appendix B-1 exemplifies a study which would be suitable at an early design stage, or as a brief comparative review at a later evolutionary stage of design. It demonstrates how useful feedback about the design concept of a system can be obtained.

Appendix B2 demonstrates how a rather more detailed approach to the study of the performance of a system may be set within the general context of an overall risk-based scheme. A still more detailed analysis than that presented in this example would be required for a complete, in-depth study.

# CHAPTER 3

## METHODS FOR PREVENTING LOCA

A complete loss-of-coolant accident (LOCA) in some higher power reactors can result in partial melting of the fuel from decay heat and release of fission products into the reactor building. For reactors operating even at relatively low power, the consequences of a complete LOCA must be examined. Even at low powers, the loss of biological shielding from the water must be considered.

In nearly all research reactors a partial uncovering of the fuel (e.g. by leakage through horizontal beam tubes) may be possible and this must be studied separately. In some reactors operating initially at powers of a few MW, a partial LOCA could result in boiling at the immersed part of the fuel. Conduction and steam cooling may be adequate to prevent melting.

The probability of a LOCA is normally reduced to very small values by engineered safety features such as:

1. Elevation of primary pipework (in pool or pool wall) above the core.
2. Elevation of pool system pipework (in pool or pool wall) above the core.
3. Antisyphon devices.
4. Valves on reactor vessel above and below core to admit pool water for natural convection cooling.
5. Emergency spray system from water storage tanks.
6. Pool-liner continued through beam tube.
7. Beam tubes sealed and terminated outside core box walls.
8. Sealed protective covers on beam tube ends in pool walls.
9. Slide-valves in pool walls at beam tube penetrations.
10. Beam tube shutters (released automatically).
11. Passive flap valves in beam tubes.

The engineered safety features appropriate to a particular reactor are dependent on its design. In each case a careful study will be necessary for every sequence of events that could lead to a loss-of-coolant, the effect of engineered safety features, and the adequacy of any heat removal system that is required to keep the reactor in a safe state until fission product decay reduces heating to an acceptable level.

Special attention should be given to the fact that some measures taken against LOCA could, by malfunction, interrupt the coolant flow rate. From this, a meltdown on operation could result. For example, automatic butterfly valves in the main pipe need careful design of the automatic control system.

Where the features protecting against LOCA may be less satisfactory, provision of an emergency core cooling system (ECCS) should be given additional consideration. The effect of earthquakes, of a magnitude appropriate to the site geological conditions, should be considered.

New designs should make the probability of a LOCA as low as possible.

Appendix C considers the protective measures taken in several reactors of very different designs: two light water swimming-pool-type reactors (DEMOCRITOS and SAPHIR), a light water tank-type reactor (HFR-Petten), and two heavy water tank-type reactors (DIDO and PLUTO).

Appendix C-1 describes the engineered safety features against LOCA for the DEMOCRITOS reactor, which is a swimming-pool-type reactor originally operated at 1 MW, when no special features were provided against LOCA. Following uprating to 5 MW, one safety concern was a LOCA due to rupture of either a pipe of the primary cooling system or of an experimental beam tube. Several alternative methods of protection were considered, but the solution that was chosen was installation of automatic valves in the primary pipework and sealed protective covers on the beam tube ends. Further consideration is now being given to provision of an anti-syphon device (Fig. 1).

Appendix C-2 describes the engineered safety features against LOCA for another swimming-pool-type reactor, SAPHIR, in which the power was increased from 5 MW to 10 MW in 1983. In SAPHIR, the pipework layout is elevated above the core and an anti-syphon arrangement is provided above the core level such that a complete LOCA cannot occur (Fig. 2). In order to prevent a partial LOCA from a beam tube failure, three measures have been taken: each collimator is sealed by an aluminum cover, each collimator is provided with a set of shutters against radiation that can be closed by actuation of an electric motor, and a slide-valve that can be manually actuated by a hydraulic system is installed in the pool wall of each beam tube end (Fig. 3).

Appendix C-3 describes the protective measures taken against both a total or partial LOCA in the 45 MW Petten HFR tank-type design. The reactor core with its adjacent devices is contained in a closed vessel at low pressure which is immersed in a pool. The protective measures against a complete uncovering of the core are a high elevation and U-type design (with a vacuum break anti-syphon system) of the primary cooling pipes (Fig. 4). For emergency cooling, manually activated valves are provided on the reactor vessel, above and below the core, to admit pool water for natural convection cooling in the event of loss of primary cooling.

In the original design, beam tubes above and below the core centerline were welded on the vessel and core box walls to prevent leakage from the vessel through the tubes. The improvements (Fig. 5) made to protect against partial uncovering of the core in the new vessel designed for 60 MW are: (1) the beam tube ends are not welded to the reactor vessel but are sealed and terminated at about 5 mm from the core box walls, (2) an aluminum protective cover is bolted at the external pool wall side of each beam tube as a second barrier to leakage of pool water through the tube, and (3) the beam tube bellows are removed, thus improving the integrity of the tubes.

Appendix C-4 deals with the heavy water tank-type reactors DIDO and PLUTO. The original spray system has been replaced following power uprating from 10 MW to 25 MW. Probability study (c.f. Appendix B-2 of this guidebook) showed that there was risk of failure of the smaller pipes of the primary systeml or more serious, of the bosses where these small pipes join the large primary pipework. The chosen solution was to isolate the primary pipework from the reactor tank by rapidly closing powered valves. Water from the sump is then returned through a cooler to the tank, and will overflow back to the sump. An additional backup light water injection system is provided, which also gives protection against seismic events. The reactor design and the system which has been installed are illusrated in Figs. 6 and 7.

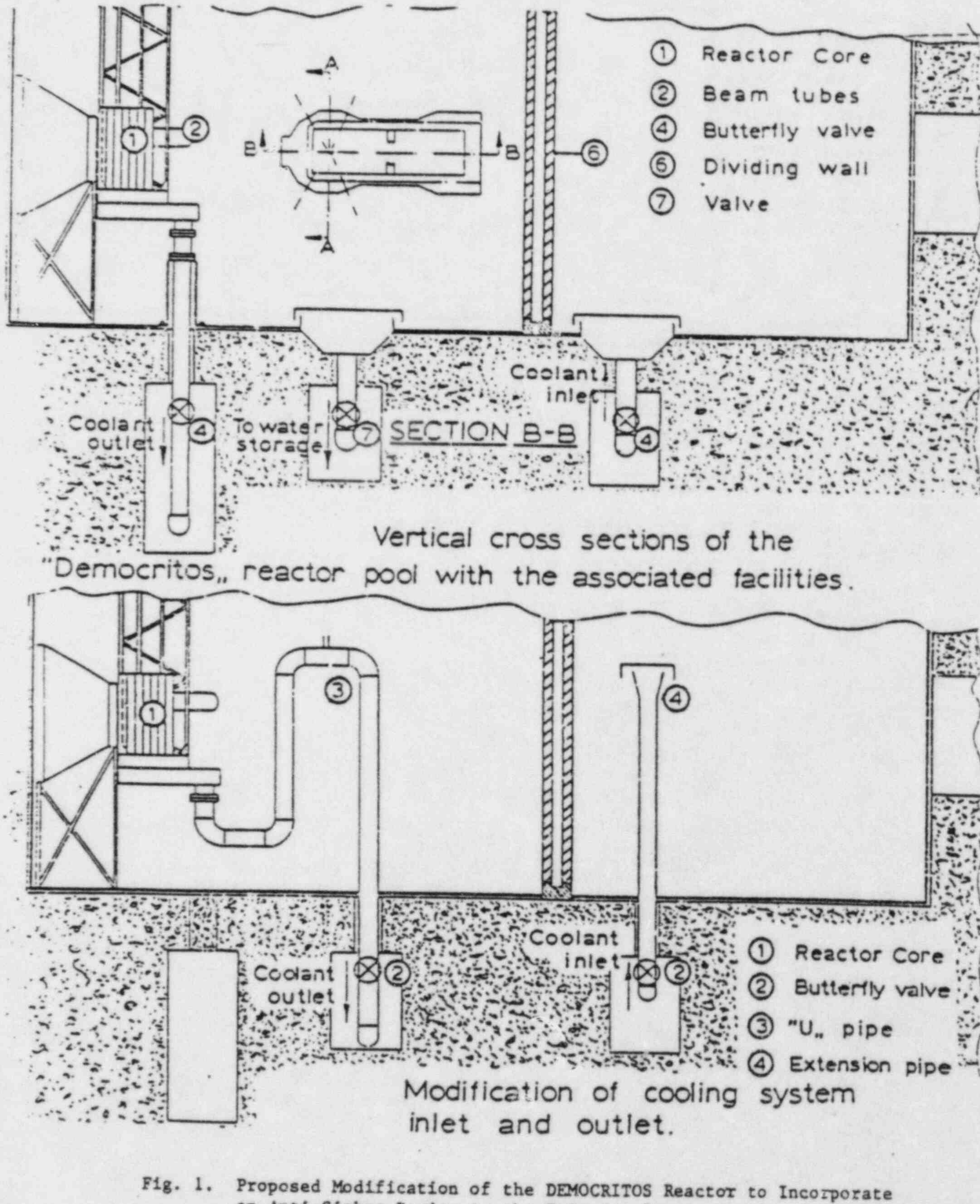



Fig. 1. Proposed Modification of the DEMOCRITOS Reactor to Incorporate an Anti-Siphon Device to the Existing System.

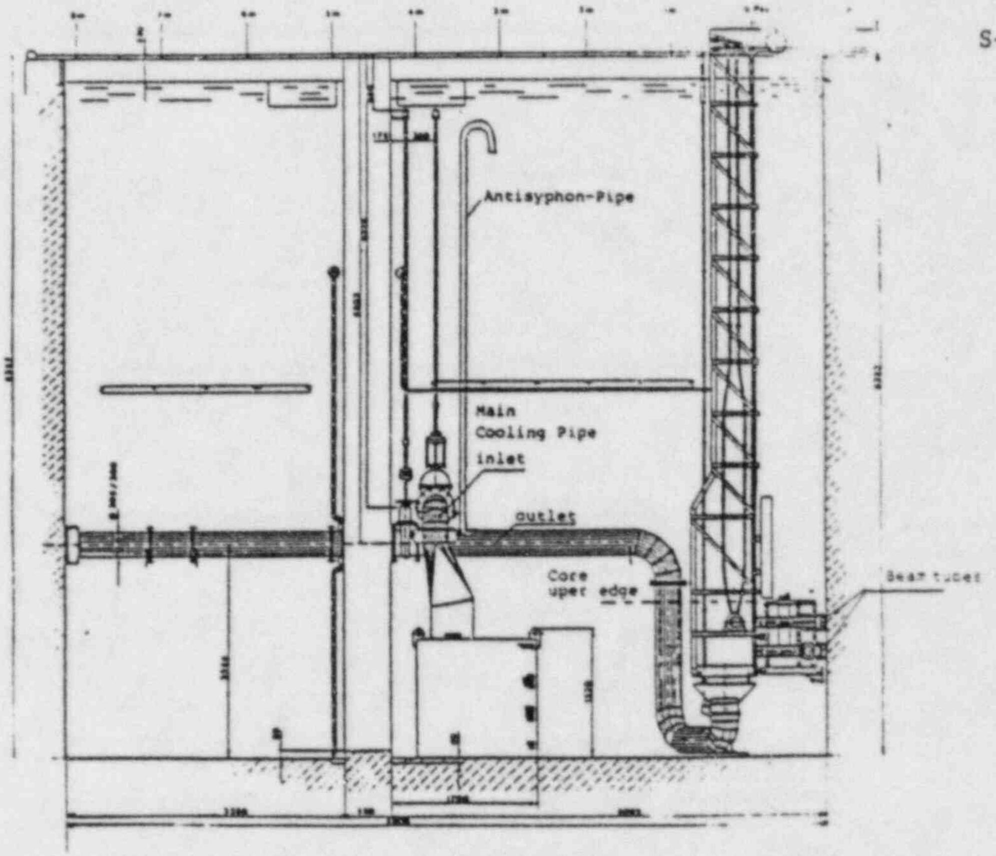



Fig. 2: Pool Longitudinal Section for the SAPHIR-Reactor

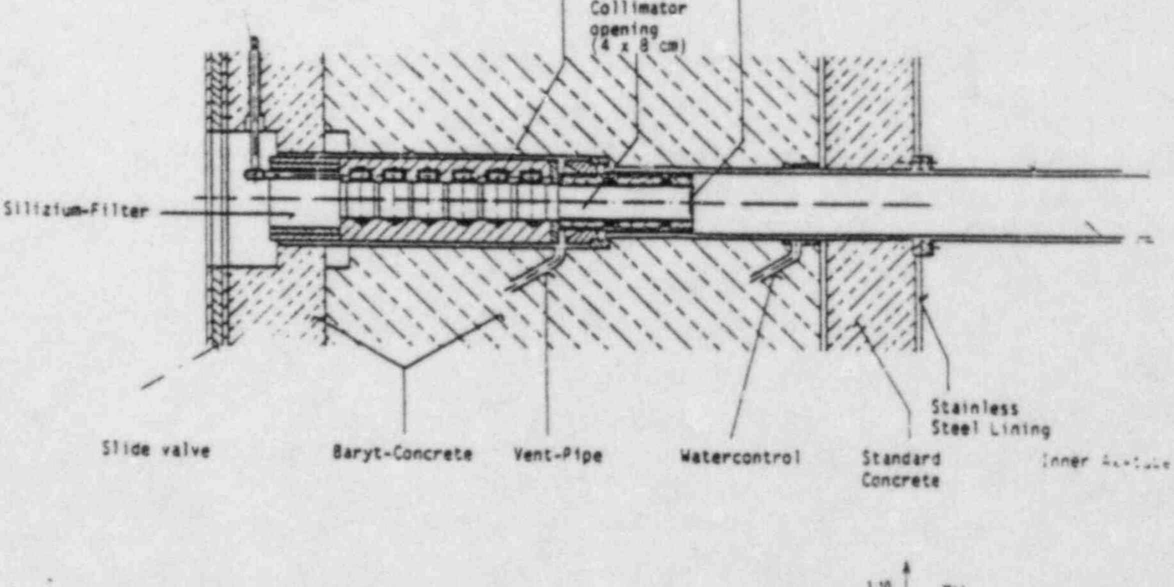



Fig. 3: Radial beam tube with shutter plug in the SAPHIR-Reactor

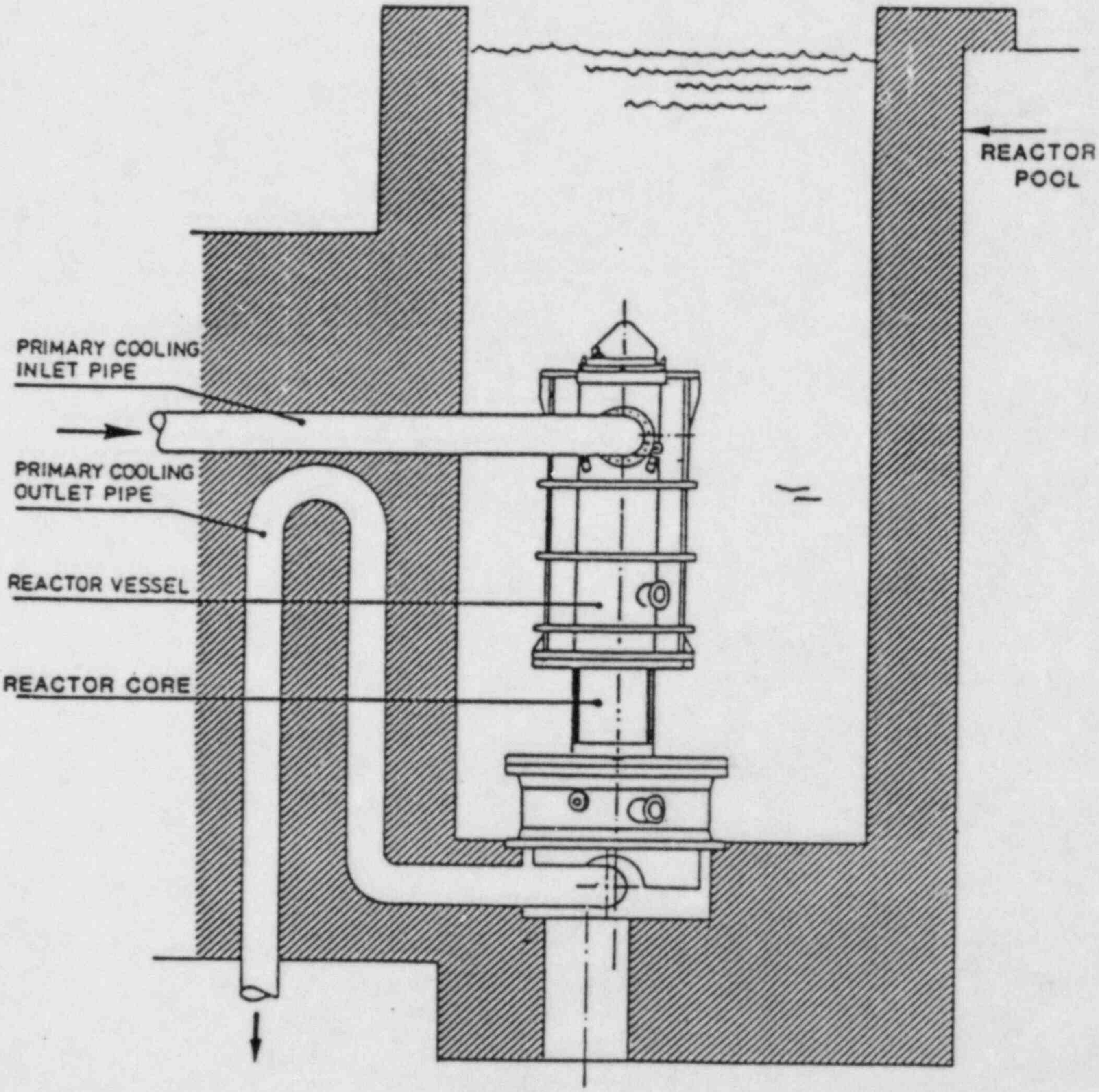
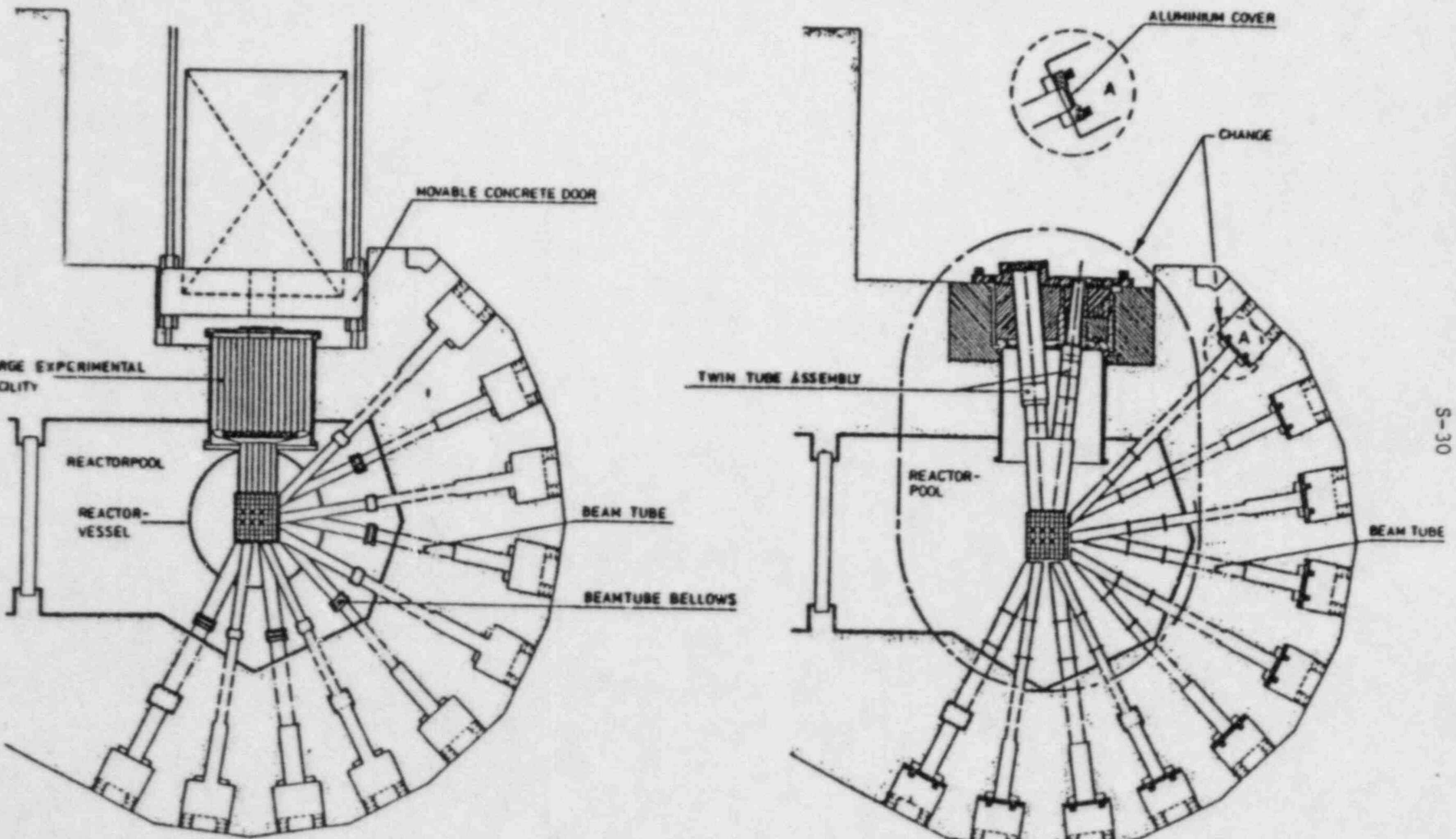



Fig. 4. Cross Section Through HFR Petten Reactor Vessel and Pool Walls.




(a)

Original Design

(b)

Improved Design

Fig. 5. Schematic Horizontal Cross Section Through HFR Petten Reactor Vessel and Beam Tubes.

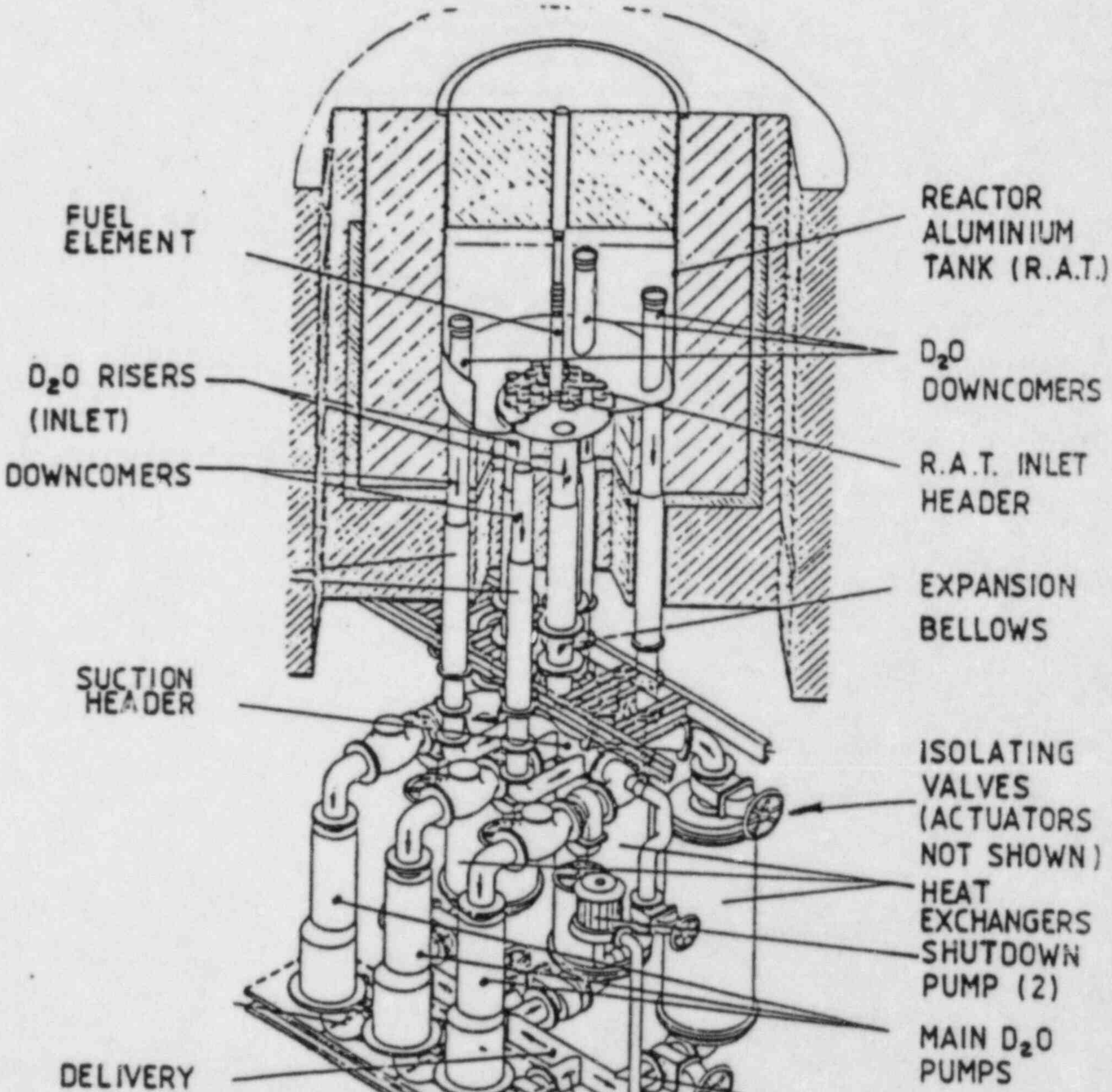



Fig. 6. $D_2O$ System for the DIDO and PLUTO Reactors.

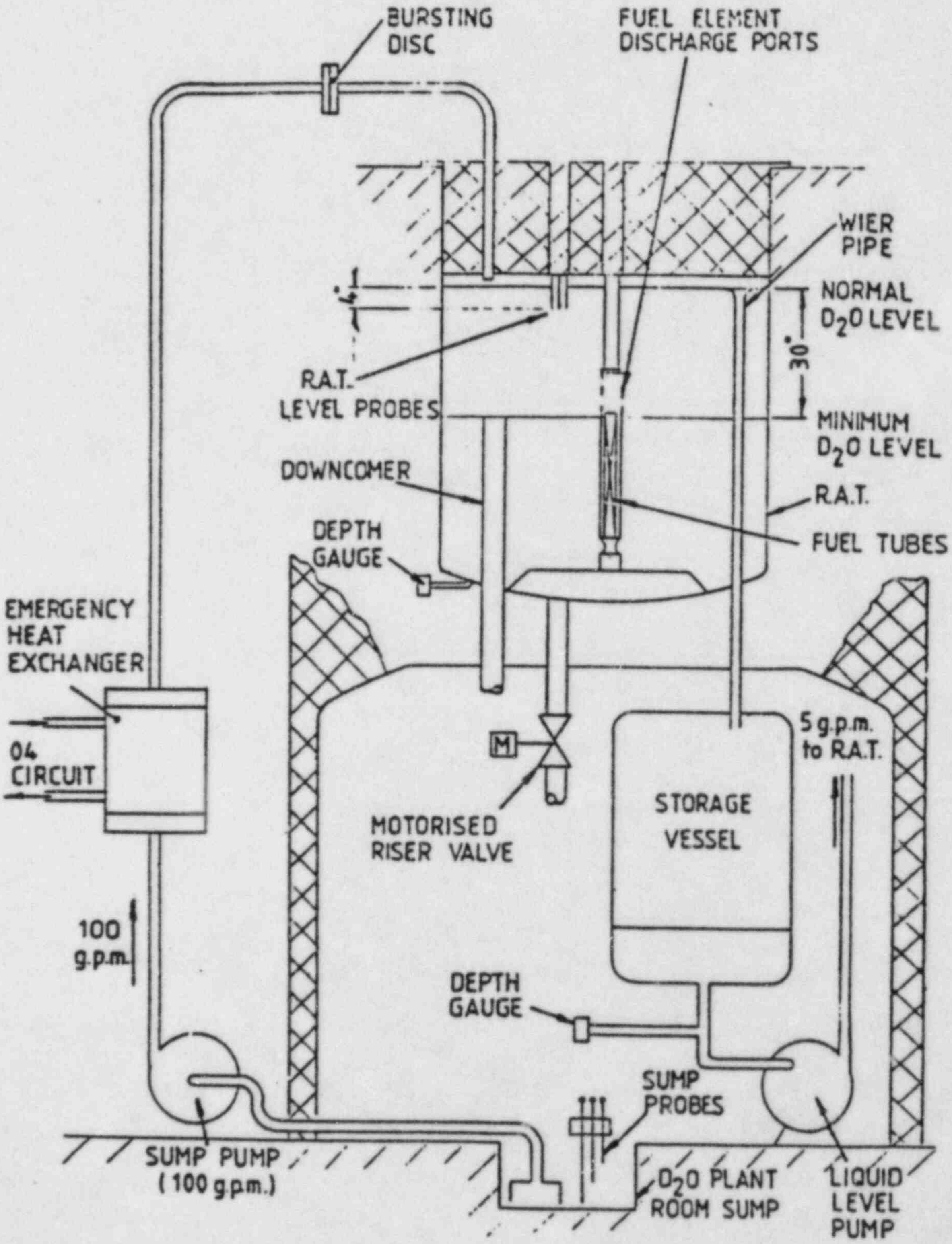



Fig. 7. Diagramatic Arrangement of Emergency Cooling System for the DIDO and PLUTO Reactors.

# CHAPTER 4

## RADIOLOGICAL CONSEQUENCES

A common approach in Safety Reports for research and test reactors is to assume that a ~~hypothetical~~ accident results in the release of some portion of the inventory of radioactive materials from the fuel to the containment/reactor building and, eventually, in the release of a portion of these materials to the atmosphere. The consequences to the surrounding population are usually evaluated in terms of estimated radiological doses from the materials released. The conversion of a research reactor from a highly enriched core to lower enrichment will generally require a review of the impact of the conversion on these previously determined radiological consequences. In most cases this impact will be relatively small.

The main factors which should be considered in performing the radiological consequence assessment include the following:

1. Core fission product and actinide inventory.

2. Fraction of the core involved in the postulated event.

3. Fractional release from the fuel elements involved in the postulated event.

4. Reactor design features affecting the release from the fuel to the containment/building.

5. Passive chemical and physical factors within the containment/building which can influence the quantity of material available for release.

6. Containment/building design features influencing the material available for release to the atmosphere.

7. Radioactive decay factors.

8. Atmospheric dispersion factors including the effect of site topography.

9. Demography of the site surroundings.

10. Estimation of individual and population exposures and risks.

Numerous methods are available for evaluating these factors. Simple hand calculations based on fission product yield table inventories, conservative fission product fractional release data, and non-site specific atmospheric dispersion data may be adequate in many cases. However a more detailed evaluation of the above factors using a sophisticated computer code will often result in substantial reductions in the dose estimates. Table 4.1 from Appendix D-1 shows a comparison of the doses calculated by the two techniques for the generic 10 MW reactor discussed in Chapter 2 and Appendix A.

Table 4.1. Comparison of Doses with Inventories from ORIGEN Code vs. Yield Tables

| | | Dose at 500 m Site Boundary, rem* | | | | |
|---|---|---|---|---|---|---|
| | | Inhalation | | | Whole Body | |
| Case | Time | Bone | Lung | Thyroid | Internal (Inhalation) | External (Immersion) |
| 100 FPD | 2 h | 0.1384 | 0.1987 | 4.419 | 1.549-02 | 5.305-02 |
| ORIGEN | 30 d | 1.305 | 1.600 | 26.38 | 0.1171 | 0.1812 |
| 300 FPD | 2 h | 0.3121 | 0.2726 | 4.305 | 2.217-02 | 5.117-02 |
| ORIGEN | 30 d | 3.065 | 2.338 | 25.74 | 0.1849 | 0.1770 |
| Cumulative Yields | 2 h | 1.712 | 0.4859 | 4.463 | 5.230-02 | 5.730-02 |
| Infinite | 30 d | 17.47 | 4.541 | 26.70 | 0.4943 | 0.2030 |
| Cumulative Yields | 2 h | 0.1373 | 0.1999 | 4.463 | 1.567-02 | 5.718-02 |
| 100 d | 30 d | 1.309 | 1.632 | 26.70 | 0.1195 | 0.2015 |

*Using peak element, 0.44342 MW, in 10 MW HEU generic reactor with a 1.0%/d leak rate and the release of 100% of Noble gases, 25% of halogens, and 1% of all other to containment.

Appendix D includes radiological consequence methods and calculations relevant to conversion from a highly enriched core to lower enrichment for several different types of research reactors and according to various national practices. Practices used in the United States, the United Kingdom, Greece, Canada, and the Federal Republic of Germany are described in the appendices.

Appendix D-1 describes a model for estimating the radiological consequences from a hypothetical accident in HEU and LEU fueled research and test reactors based on U.S. Nuclear Regulatory Commission requirements. The method incorporates fission product inventories and dose conversion data to calculate doses. The model accounts for containment/building leakage, decay of fission products, and the dispersion of airborne material by diffusion factors based on release height, wind velocity, atmospheric stability, and diffusion parameters.

This analysis shows that the LEU fuel gives essentially the same doses as HEU fuel. Table 4.2 gives a comparison of doses for LEU and HEU fuel for the generic 10 MW reactor at a 500 m site boundary. The analysis also shows that the plutonium buildup in the LEU fuel does not significantly increase the radiological consequences. Table 4.3 shows the plutonium buildup within the fuel with irradiation time. Figure 4.1 shows the variation in isotopic contribution to bone dose with irradiation time and burnup for LEU fuel. While the fractional contributon to bone dose from the plutonium isotopes increases with time, the bone dose is still substantially below that for the thyroid.

Table 4.2. Doses at 500 m Site Boundary for 10 MW Generic Reactor 100 FPD Peak Element With HEU and LEU Fuel*

| | HEU | LEU |
|---|---|---|
| Bone Dose, rem | | |
| 2 h | 1.384-1 | 1.504-1 |
| 30 d | 1.305 | 1.425 |
| Lung Dose, rem | | |
| 2 h | 1.987-1 | 2.031-1 |
| 30 d | 1.600 | 1.634 |
| Thyroid Dose, rem | | |
| 2 h | 4.419 | 4.519 |
| 30 d | 26.38 | 26.98 |
| Whole Body (internal), rem | | |
| 2 h | 1.549-2 | 1.604-2 |
| 30 d | 1.171-1 | 1.217-1 |
| Whole Body (external), rem | | |
| 2 h | 5.305-2 | 5.461-2 |
| 30 d | 1.812-1 | 1.874-1 |
| Burnup, MWD | 44.34 | 45.11 |

*Assuming 100% of noble gases, 25% of halogens, and 1% of other are available for release from the containment, and a leakage rate from the containment of 1%/day (using Regulatory Guide 1.4 $\chi/Q$ values).

Table 4.3. 10 MW LEU Generic Reactor - Plutonium Buildup and Dose

| Irrad. Time, d | Atom % Burnup $^{235}U$ | 390 g $^{235}U$ LEU Peak Power (0.4511 MW) Element: Mass, g | | | | Dose, rem at 2 h (30 d) | | |
|---|---|---|---|---|---|---|---|---|
| | | Pu-238 | Pu-239 | Pu-240 | Pu-241 | Bone | Lung | Thyroid |
| 100 | 14 | - | 3.61 | 0.18 | 0.02 | 0.150 (1.42) | 0.203 (1.63) | 4.52 (27.0) |
| 200 | 28 | 0.01 | 6.71 | 0.66 | 0.14 | 0.233 (2.25) | 0.250 (2.10) | 4.53 (27.1) |
| 300 | 41 | 0.04 | 9.19 | 1.36 | 0.46 | 0.319 (3.14) | 0.278 (2.37) | 4.48 (26.8) |
| 400 | 54 | 0.10 | 11.0 | 2.20 | 1.03 | 0.440 (4.37) | 0.300 (2.60) | 4.48 (26.8) |
| 500 | 66 | 0.23 | 12.3 | 3.802 | 1.84 | 0.613 (6.15) | 0.319 (2.80) | 4.43 (26.7) |

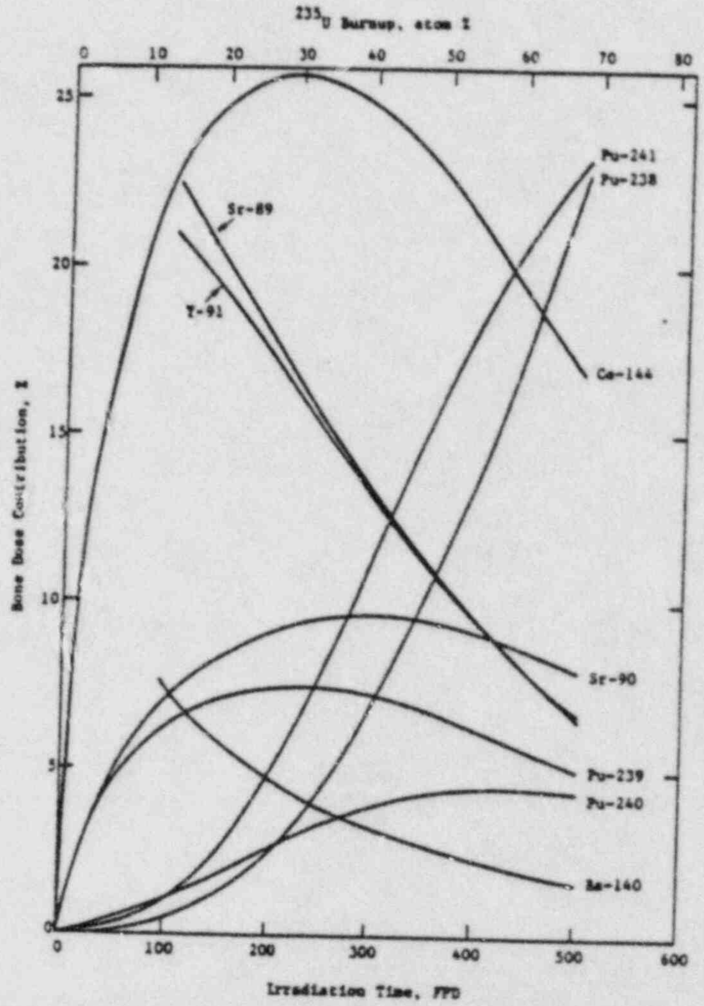



Fig. 4.1 Variation in Isotopic Contribution to Bone Dose With Irradiation Time for LEU Fuel

Appendix D-2 describes methods used by the Safety and Reliability Directorate of the UKAEA for conducting a radiological consequence analysis. For the assessment of accidental releases of radioactive material to the atmosphere, it would be normal in the UK to use the Tirion of Weerie suites of computer codes. These contain models of all the important physical processes between the release from the containment and the dose commitment to the individual. Tirion is a Gaussian Plume constant weather code containing the following models: Release Model, Dispersion Model, Dry Deposition, Radioactive Decay, Building Wakes, Lift-Off, Plume-Rise, Plume-Breakup, Inversion Lids, Inhalation Dose, External Radiation Dose, Consequence of Absorbed Doses, Number of Casualties, and Deposited Activity.

This appendix also discusses a simpler and quicker method of obtaining public doses by the use of preprepared solutions to the diffusion equation plus dose conversion factors. An example of this type of simple calculation is given where a child thyroid dose of 127 mrem is calculated for a 10 Ci, 2 hour, I-131 release from a 50 m high stack at a distance of 500 m under worst weather conditions.

Appendix D-3 describes a model for calculating radiological consequences appropriate for the Democritos research reactor in Athens, Greece. The model covers all steps of the chain: Source Term, Air Concentration of Radioactivity, and Adsorbed Dose, and is purposely simple for use when additional computer codes are not available. In addition to describing simple methods for calculating the source term and the release rates for radioactive effluents, this appendix gives simple formulae for calculating the radiological dose from all relevant exposure pathways.

The pathways considered are the external whole body dose due to submersion in the exhaust air plume, the external whole body dose due to the activity deposited on the ground, the internal irradiation originating from radionuclides inhaled with the air, resulting in both critical organ and whole body doses, external beta radiation from the exhaust air plume and the internal irradiation due to consumption of contaminated food. In order to determine the total dose of the whole body or of a certain critical organ, the contributions of all relevant radionuclides via the exposure pathways have to be summed for the individual receptor.

Appendix D-4 describes some of the factors to be considered in a radiological consequence anlaysis for a high power Canadian Research Reactor. The methods of determining the inventory of fission products and some of the factors affecting the fission product release including the chemical behaviour of the radioiodines is considered. Atmospheric dispersion factors including meteorological conditions, stack heights, ground contours, and building wake effects are also described. Simple formulae are given for calculating whole body external doses for gamma radiation from a cloud of noble gas fission products. Correction factors are given to account for the finite size of the noble gas plume when calculating both the individual dose on the plume centreline and the dose to a population distributed above the plume centreline.

Appendix D-5 describes a fundamental calculation model for the determination of the radiological effects, inside and outside a research reactor, after hypothetical accidents, with release of high amounts of fission products from the core as performed by Interatom of the Federal Republic of Germany. Conservative calculation methods are used to solve the problem. The reference reactor for the model assumption is a pool type (light water) reactor.

Models are presented for the following processes: source term, containment and activity enclosure, time dependent activity behaviour in the reactor building, radiation exposure in the reactor plant, and radiological exposure in the environment.

In Appendix D-6, GA Technologies Inc. presents methods and sample calculations for a radiological consequence analysis for U-ZrH fuel failure in a TRIGA reactor. The analysis summarizes the calculations in tabular form for four failure modes: single-rod failure in air and three 16-rod cluster failures in water using both experimentally based fission product release fractions and conservative design basis release fractions. Fission product inventories and release fractions, building release and downwind doses are described and compared to design basis dose criteria.

One of the critical parameters in any radiological consequence analysis is the assumed fractional release of fission products from the fuel to the reactor building. Several of the appendices give typical values which are used in their jurisdiction and these are summarized in Table 4.4. For noble gases, it is common to assume a release fraction of 1.0. Radioiodine release fractions used in safety analyses vary widely depending on the degree of conservatism used and the type of accident being considered. For example, it is common practice in some jurisdictions to assume an effective radioiodine release fraction of 0.25 from the portion of the fuel that has failed. On the other hand, for a release under water, most of the radioiodine will remain in the liquid phase and values ranging from $10^{-2}$ to $10^{-4}$ are commonly used. A similar situation exists for particulates where a release fraction of 0.01 is commonly assumed. In one case this is modified to $10^{-5}$ due to the presence of the pool water.

Table 4.4. Fission Product Release Fractions - Fuel to Reactor Building (Fraction of Core Inventory).

| | ANL (App. D-1) | GAEC (App. D-3) | INTERATOM (App. D-5) | GA (a) (App. D-6) |
|---|---|---|---|---|
| Noble Gas | 1.0 | 1.0 | 1.0 | 1.0 |
| Radioiodines | | | | |
| From Fuel to Pool | | | 0.25 | $0.25 \times 6.3 \times 10^{-4}$ (c) |
| From Pool to Building | | | | |
| Elemental | | | $10^{-4}$ | $10^{-2}$ (90%) |
| Organic | | | $10^{-2}$ | 1.0 (10%) |
| Total to Building | | | | |
| Elemental | | | $2.5 \times 10^{-5}$ | |
| Organic | | | $2.5 \times 10^{-3}$ | |
| Total | | | | 0.25 (d) |
| | 0.25 (b) | 0.25 | | $2.725 \times 10^{-2}$ (e) |
| | | | | $1.72 \times 10^{-5}$ (c) |
| Particulates | 0.01 | 0.01 | $10^{-5}$ | |

(a) Values given are fraction of gap inventory.

(b) Assumed 50% release from fuel and 50% of that available in the building.

(c) Experimental release value.

(d) Release in air.

(e) Design basis release value.

# CHAPTER 5

## EXAMPLES OF AMENDMENTS TO SAFETY REPORTS

Appendix E contains several documents intended to illustrate what work has been required, or is expected to be required, in preparing a Safety Report amendment for a licensing authority in order to obtain a licence for core conversion.

The documents present a spectrum of reactor types from critical facilities to research reactors with power levels between 2 MW and 70 MW. The changes which are addressed range from the testing of prototype elements, to full core replacement. One reactor required changes to associated plant as well.

In most cases, the documents are summaries indicating the work required. That for the FNR (Appendix E-3) is the actual Safety Report amendment, and that for GRR-1 (Appendix E-6) is an example document illustrating the format set out in this Guidebook (and also IAEA Safety Series No. 35, 1984 Edition) using GRR-1 as an example. With the exception of GRR-1, these documents describe changes which have been approved and in some cases already successfully implemented.

In all cases, reactor physics parameters were recalculated with the new fuel. In the case of reactors of significant thermal powers, power distributions were recalculated and the effects on thermal-hydraulic behaviour considered. Fuel material and cladding behaviour was discussed in detail, including fuel failure, reference being made to experimental investigation and fuel testing programmes. Where other significant changes had been made, their effects were also considered. In some cases the opportunity was taken to incorporate recently developed requirements into the safety analysis. In some cases it was also necessary to re-evaluate fission product inventories, fission product releases and radiological consequences.

It will be seen from the variations between the documents presented that core conversion to MEU or LEU can encompass a wide range, from changes to the fuel material only to changes to the core and some associated plant changes. The extent of the Safety Report amendments required varies from case to case over a similarly wide range.

Table 5.1 gives an overview of Appendices E-1 to E-7 in order to help the reader to find adequate examples for his individual requirements.

Table 5.1

Characteristics and Conversion Status of Reactors
with Example SAR Amendments in Appendix E

| | | | | Fuel Conversion | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | From | | | To | | | | | | |
| Appendix | Reactor | Country | Power, MW | Enr., % | Chem. Comp. | U Dens. $g/cm^3$ | Enr., % | Chem. Comp. | U Dens. $g/cm^3$ | Core Alteration | Design Change | Start of Operation | Comment |
| E-1 | KUCA* | Japan | $< 10^{-6}$ | 93 | Alloy | 0.75 | 45 | $UAl_x$ | 1.7 | Full | No | 5/81 | RERTR Demonstration |
| E-2 | JMTRC* | Japan | $< 10^{-6}$ | 93 | Alloy | 0.75 | 45 | $UAl_x$ | 1.7 | Full | No | 9/83 | RERTR Demonstration |
| E-3 | FNR | USA | 2 | 93 | $UAl_x$ | 0.44 | 20 | $UAl_x$ | 1.7-1.8 | Full | No | 12/81 | RERTR Demonstration |
| E-4 | OSIRIS | France | 70 | 93 | $UAl_x$ | 1.0 | 7 | $UO_2$ | 9.5 | Full | Yes | 12/79 | Normal Operation |
| E-5 | DIDO PLUTO | UK | 25.5 | 70 | Alloy | 0.65 | 20 | $U_3O_8$ | 2.3 | Full | No | | Feasibility Study |
| E-6 | GRR-1 | Greece | 5 | 93 | $UAl_x$ | 0.6 | 20 | $U_3O_8$ | 2.2 | Full | No | | Example SAR |
| E-7 | FRG-2 | FRG | 15 | 93 | $UAl_x$ | 0.44 | 45 | $UAl_x$ | 1.5 | Partial | No | 2/82 | Fuel Test |
| | | | | | | | 20 | $U_3O_8$ | 3.1 | Partial | No | 9/83 | Fuel Test |
| | | | | | | | 20 | $U_3Si_2$ | 3.7 | Partial | No | 4/84 | Fuel Test |

*Critical Facility

# CHAPTER 6

## SAFETY SPECIFICATIONS

Appendix F contains two contributions dealing with determination of the power limits of a reactor facility with reduced enrichment fuel compared with highly enriched fuel.

Generally, there will be a strong incentive to retain the primary circuit without modification. As core conversion requires an increase of uranium contents, sometimes realized not only by increasing the uranium density in the fuel meat but also by increasing the fuel meat volume (and thus the thickness of the fuel plates), the consequences of such geometrical changes have to be investigated thoroughly, regardless of any changes in the primary circuit.

In Appendix F-1, a general procedure is described for determination of the new power limits. An adequate criterion is formulated in which not only the geometrical parameters of the fuel elements change, but also the new coolant channel flow which can be derived from the new circuit resistance and the pump characteristics.

In Appendix F-2, the consequences of core conversion are investigated in a specific reactor (HFR at Petten) for three specific types of LEU elements. The LEU elements considered are elements with 15, 16 and 18 relatively thick plates. These are compared with the present 23 plate HEU elements. In this comparison, the present primary circuit has been considered as a boundary condition.

The limiting power is determined by the bubble detachment criterion. From this criterion, it follows that the power limit is proportional to the number of fuel plates (heated area), dependent on the coolant channel thickness, and proportional to the coolant velocity in the channels. This last quantity is determined by the primary coolant flow, which is derived from the pump characteristics and the primary circuit resistance. The changes in this resistance have been calculated.

It is seen that for the cases studied, the loss in heated area (fewer plates) is compensated for more than 50% by an increase in coolant velocity in the elements. This increase is due to the circumstance that in this type of reactor the total flow is influenced weakly by the reduced flow area of the considered LEU fuel elements.

CHAPTER 7

# BENCHMARK CALCULATIONS

## 7.1 INTRODUCTION

A safety-related benchmark problem for an idealized light-water, pool-type reactor was specified in order to compare calculational methods used in various research centres and institutions. The specifications of this problem are given in Appendix G-0 and the detailed results contributed by five organizations (ANL, INTERATOM, JAERI, EIR, and JEN) are provided in Appendices G-1 through G-5. The reactor and loading specifications are identical to those of the 10 MW neutronics benchmark problem defined in IAEA-TECDOC-233 (August 1980) except for the central flux trap.

For heavy-water-moderated research reactors, a separate benchmark problem for both the neutronics and safety-related parameters was defined and a summary and the detailed results contributed by five organizations (ANL, HARWELL, AAEC, JAERI, and RISØ) are provided in IAEA-TECDOC-324 (January 1985). In the interest of completeness, selected transient calculations contributed by the AAEC for a reactor very similar to the heavy water benchmark reactor are included in this summary and in Appendix G-6.

Since the purpose of these benchmark problems is to compare calculational methods, the reactor configurations were idealized and simplified. Thus, these calculations may not correspond to realistic reactor conditions, and only limited conclusions about actual reactor performance and safety should be drawn from them, even though some results are very similar to the results of the generic studies for light water reactors and the specific studies for heavy water reactors.

The main parameters which have been calculated for the light-water reactor benchmark problem with HEU and LEU fuels are:

- Prompt Neutron Generation Times
- Delayed Neutron Fractions
- Isothermal Temperature and Void Reactivity Coefficients
- Radial and Local Power Peaking Factors
- Control Rod Reactivity Worths
- Power and Temperature Responses to Loss of Flow
- Power and Temperature Responses to Ramp Reactivity Insertions

In the following sections, the main results of the calculations are summarized and compared.

## 7.2 RESULTS OF STATIC CALCULATIONS

### 7.2.1 Basic Kinetic Parameters

The results of the calculations of the basic kinetic parameters, namely prompt neutron generation time ($\Lambda$) and delayed neutron fraction ($\beta_{eff}$) are presented in Table 7.1.

The values show about a 23% shorter generation time $\Lambda$ and a 3-5% lower $\beta_{eff}$ value for the LEU case compared to the HEU case. This is mainly due to the harder neutron-spectrum in the LEU case.

Table 7.1 - Basic Kinetic Parameters

| Parameter | Fuel | ANL | INTERATOM | JAERI | EIR | JEN |
|---|---|---|---|---|---|---|
| $\Lambda$ | HEU | 56.0 | 54.5 | 57.6 | 58.8 | 51.7 |
| (μs) | LEU | 43.7 | 42.2 | 44.4 | 44.8 | 38.0 |
| $\beta_{eff}$ | HEU | 0.761 | 0.762 | 0.744 | 0.778 | 0.736 |
| (%) | LEU | 0.728 | 0.732 | 0.722 | 0.736 | 0.713 |

### 7.2.2 Isothermal Reactivity Coefficients

Table 7.2 shows the average values of the temperature coefficients of reactivity for the HEU and LEU cases over the temperature ranges 20-38°C, 38-50°C, and 50-100°C for change of water temperature only, change of water density only, and change of fuel temperature only. Least-squares fits of the data in Appendices G-1 to G-5 were performed to obtain the values shown in Table 7.2 for those cases in which calculations were not done at the specified points. Also shown in this table is whole-core void coefficient of reactivity for a change in water density from 0.958 - 0.90 g/cm$^3$. Figure 7.1 gives as an example the calculated isothermal reactivity differences as functions of temperature and uranium enrichment.

The dependence of these coefficients on the enrichment of the fuel are described below:

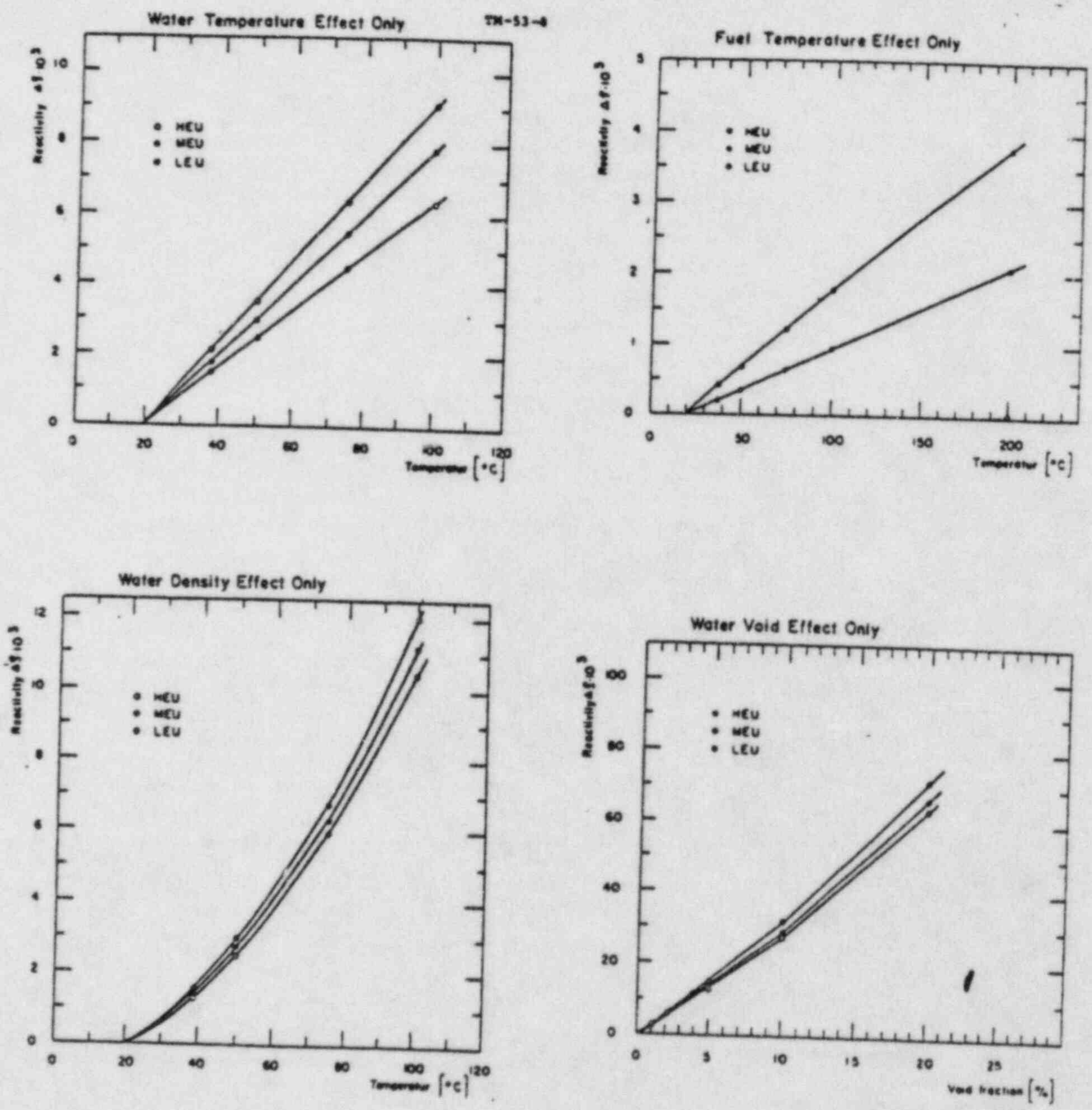
- The density component of the water reactivity coefficient increases and the temperature component decreases when changing from HEU to LEU fuel. When the water temperature and density coefficients are combined, the values for the LEU core are smaller by ~9% over the temperature range 38-50°C and smaller by ~5% over the temperature range 50-100%.

  It should be mentioned that the water density coefficient is nearly linear over the density range 0.998-0.958 g/cm$^3$ (20-100°C). When plotted against temperature, the reactivity loss due to decreasing water density is non-linear because the dependence of water density

Table 7.2 - Isothermal Reactivity Coefficients

| Effect | Fuel | ANL | INTER-ATOM | JAERI | EIR | JEN |
|---|---|---|---|---|---|---|
| | | Temperature Range: 20-38°C ($-\Delta\rho/°C \times 10^5$) | | | | |
| Water Temp. Only: $\alpha_{T_w}$ | HEU | 11.9 | 10.4 | 9.6 | 12.0 | 9.0 |
| | LEU | 8.2 | 7.9 | 9.6 | 8.5 | 7.1 |
| Water Density Only: $\alpha_{D_w}$ | HEU | 7.1 | 6.8 | 5.7 | 7.3 | 12.0 |
| | LEU | 8.3 | 7.9 | 6.3 | 8.5 | 13.6 |
| $\alpha_{T_w} + \alpha_{D_w}$ | HEU | 19.0 | 17.2 | 15.3 | 19.3 | 21.0 |
| | LEU | 16.5 | 15.8 | 15.9 | 17.0 | 20.7 |
| Fuel Temp. Only: $\alpha_{T_f}$ | HEU | 0.058 | 0.045 | 0.113 | 0.02 | 0.020 |
| | LEU | 2.63 | 2.19 | 1.94 | 2.37 | 3.15 |
| | | Temperature Range: 38-50°C ($-\Delta\rho/°C \times 10^5$) | | | | |
| Water Temp. Only: $\alpha_{T_w}$ | HEU | 11.9 | 10.8 | 10.3 | 11.6 | 8.7 |
| | LEU | 8.1 | 7.7 | 9.2 | 8.2 | 6.8 |
| Water Density Only: $\alpha_{D_w}$ | HEU | 10.4 | 10.0 | 7.9 | 10.4 | 17.5 |
| | LEU | 12.3 | 11.2 | 9.7 | 11.7 | 19.6 |
| $\alpha_{T_w} + \alpha_{D_w}$ | HEU | 22.3 | 20.8 | 18.2 | 22.0 | 26.2 |
| | LEU | 20.4 | 18.9 | 18.9 | 19.9 | 26.4 |
| Fuel Temp. Only: $\alpha_{T_f}$ | HEU | 0.055 | 0.044 | 0.104 | 0.02 | 0.020 |
| | LEU | 2.58 | 2.17 | 1.92 | 2.16 | 3.08 |
| | | Temperature Range: 50-100°C ($-\Delta\rho/°C \times 10^5$) | | | | |
| Water Temp. Only: $\alpha_{T_w}$ | HEU | 11.6 | 11.4 | 11.8 | 11.2 | 8.0 |
| | LEU | 7.8 | 7.5 | 8.2 | 7.8 | 6.2 |
| Water Density Only: $\alpha_{D_w}$ | HEU | 15.7 | 14.5 | 12.0 | 15.9 | 26.7 |
| | LEU | 18.6 | 17.1 | 14.3 | 18.1 | 29.8 |
| $\alpha_{T_w} + \alpha_{D_w}$ | HEU | 27.3 | 25.9 | 23.8 | 27.1 | 34.7 |
| | LEU | 26.4 | 24.6 | 22.5 | 25.9 | 36.0 |
| Fuel Temp. Only: $\alpha_{T_f}$ | HEU | 0.034 | 0.042 | 0.087 | 0.02 | 0.019 |
| | LEU | 2.52 | 2.12 | 1.89 | 2.19 | 2.94 |
| | | Water Density Range: 0.958-0.90 g/cm$^3$ ($-\Delta\rho/\Delta\rho_w$) | | | | |
| Voids or Water Density: $\alpha_V$ | HEU | 0.296 | 0.278 | 0.222 | 0.300 | 0.466 |
| | LEU | 0.344 | 0.316 | 0.232 | 0.337 | 0.513 |
| | | Water Density Range: 0.998-0.958 g/cm$^3$ ($-\Delta\rho/\Delta\rho_w$) | | | | |
| | HEU | 0.258 | 0.239 | 0.199 | 0.261 | 0.442 |
| | LEU | 0.305 | 0.280 | 0.237 | 0.299 | 0.490 |

Fig. 7.1 Isothermal Reactivity Feedback Data Corresponding to Changes in Water Temperature Only, Water Density Only, Fuel Temperature Only, and Water Void Fraction Only for the HEU, MEU, and LEU Cores



on water temperature is non-linear. The data at the bottom of Table 7.2 show that the water density coefficient is more negative in the LEU core than in the HEU core by about 15-20% over the density range 0.998-0.958 g/cm$^3$.

- A safety benefit in changing from HEU to LEU fuel is the significant increase in the fuel temperature coefficient due to the Doppler effect. This coefficient is almost zero in HEU fuel. The reactivity feedback due to the Doppler effect is virtually instantaneous as the temperature of the LEU fuel meat increases, while the reactivity feedback due to increasing water temperature and decreasing water density is delayed because heat generated in the fuel meat must be first transferred to the cladding and then to the water. The Doppler component also has more weight than the water component because temperature differences from nominal conditions are generally larger in the fuel meat than in the water.

- The whole-core void coefficient is more negative by about 10-15% for LEU fuel than for the HEU fuel over the water density range 0.958-0.90 g/cm$^3$. This increased void coefficient in the LEU case is very significant in the analysis of certain extreme hypothetical accidents.

The changes in the kinetic parameters and reactivity feedback coefficients discussed above are not meaningful by themselves, but have significance only when they are combined in the analyses of transients and operational reactivity swings. The results of these analyses are discussed in sections which follow.

### 7.2.3 Power Defect of Reactivity

For reactor operation, the power defect of reactivity (or the cold-to-hot reactivity swing) is an important parameter, defined as the total of all reactivity effects induced by bringing the reactor (at full flow) from cold zero-power conditions to normal operating conditions. That is:

$$\Delta\rho_{power} = (\alpha_{T_w} + \alpha_{D_w})\,\overline{\Delta T_w} + \alpha_{T_f}\,\overline{\Delta T_f}$$

where $\alpha_{T_w}$, $\alpha_{D_w}$, and $\alpha_{T_f}$ are the temperature coefficients of reactivity defined in Table 7.2 and $\overline{\Delta T_w}$ and $\overline{\Delta T_f}$ are the mean temperature differences in the water and in the fuel from cold zero-power conditions to normal operating conditions. Some steady-state thermal-hydraulic data calculated for the average channel in the 10 MW benchmark core with HEU and LEU fuels are shown in Table 7.3.

Table 7.3. Steady-State Thermal-Hydraulic Data for the Average Channel in the Benchmark Core.

| Fuel | Inlet Temp., °C | Flow Rate, m$^3$/h | Mean Water Temp., °C | $\overline{\Delta T_w}$, °C | Mean Fuel Temp., °C | $\overline{\Delta T_f}$, °C | Water Outlet Temp., °C | Peak Fuel Temp., °C |
|---|---|---|---|---|---|---|---|---|
| HEU | 38 | 1000 | 42.5 | 4.5 | 54.7 | 16.7 | 47.1 | 61.7 |
| LEU | 38 | 1000 | 42.5 | 4.5 | 54.9 | 16.9 | 47.1 | 62.0 |
| HEU | 20 | 1000 | 24.5 | 4.5 | 39.3 | 19.3 | 29.1 | 47.6 |
| HEU | 20 | 800 | 25.8 | 5.8 | 43.3 | 23.3 | 31.5 | 53.3 |

For the HEU and LEU cases in Table 7.3 with an inlet temperature of 38°C and a flow rate of 1000 $m^3/h$, the mean temperature differences between zero power and full power would be about 4.5°C in the water and about 16.8°C in the fuel meat.

Table 7.4 shows the water, fuel, and total reactivity differences between zero and full power computed using the isothermal reactivity coefficients in Table 7.2 for the temperature range 38-50°C. The difference in the power defect of reactivity between the HEU and LEU cores in this example is about 4-6 ¢.

Table 7.4 Power Defect of Reactivity: $\Delta\rho \times 10^3$

$\overline{\Delta T_w} = 4.5°C$; $\overline{\Delta T_f} = 16.8°C$

| Effect | Fuel | ANL | INTER-ATOM | JAERI | EIR | JEN |
|---|---|---|---|---|---|---|
| Water Temp. | HEU | 1.004 | 0.936 | 0.819 | 0.990 | 1.179 |
| + Density | LEU | 0.918 | 0.851 | 0.851 | 0.900 | 1.188 |
| Fuel Temp. | HEU | 0.009 | 0.007 | 0.018 | 0.003 | 0.003 |
| | LEU | 0.433 | 0.365 | 0.323 | 0.363 | 0.517 |
| $\Delta\rho_{power}$ | HEU | 1.013 | 0.943 | 0.837 | 0.993 | 1.182 |
| | LEU | 1.351 | 1.216 | 1.174 | 1.263 | 1.705 |
| $\beta_{eff}$, % | HEU | 0.761 | 0.762 | 0.744 | 0.778 | 0.736 |
| | LEU | 0.728 | 0.732 | 0.722 | 0.736 | 0.713 |
| $\Delta\rho_{power}$, ¢ | HEU | 13.3 | 12.4 | 11.3 | 12.8 | 16.1 |
| | LEU | 18.6 | 16.6 | 16.3 | 17.2 | 23.9 |

### 7.2.4 Power Peaking Factors

The results of the specified 2D calculations of the radial and local power peaking factors when selected fuel elements in the HEU and LEU BOC cores (with equilibrium fission product concentrations) were replaced with elements having fresh fuel are shown in Table 7.5. The radial power peaking factor is defined as the ratio of the average midplane power in a specific element to the average midplane power in the core. The local power peaking factor is defined as the ratio of the maximum midplane power to the average midplane power in the specified element that was substituted.

The parameter which is most significant in Table 7.5 is the product of the radial and local factors. For the HEU core with HEU element substitutions and the LEU core with LEU element substitutions, the limiting radial × local power peaking factors (in CFE-1 or SFE-1) calculated by each contributor are about the same in both cores, though there are some significant differences.

For an initial mixed core (one LEU element in a HEU core) the radial × local peaking factors are larger than in the HEU core by about 16 - 20% because the fissile content of the LEU elements is larger by a factor of 1.39.

Table 7.5. Power Peaking Factors

| Core | Fresh Element | Element Substituted | ANL | INTER-ATOM | EIR | JEN |
|---|---|---|---|---|---|---|
| | | | **Radial** | | | |
| HEU Core | HEU in HEU Core | none | 1.02 | 1.05 | 1.08 | 1.02 |
| | | CFE-1 | 1.36 | 1.32 | 1.12 | 1.33 |
| | | SFE-1 | 1.11 | 1.14 | 1.18 | 1.14 |
| | LEU in HEU Core | CFE-1 | 1.49 | 1.48 | 1.28 | 1.47 |
| | | SFE-1 | 1.21 | 1.26 | 1.28 | 1.25 |
| LEU Core | LEU in LEU Core | none | 1.02 | 1.04 | 1.06 | 0.99 |
| | | CFE-1 | 1.31 | 1.26 | 1.10 | 1.29 |
| | | SFE-1 | 1.10 | 1.12 | 1.15 | 1.14 |
| | | | **Local** | | | |
| HEU Core | HEU in HEU Core | none | 1.44 | 1.37 | 1.36 | 1.45 |
| | | CFE-1 | 1.34 | 1.25 | 1.18 | 1.26 |
| | | SFE-1 | 1.43 | 1.38 | 1.39 | 1.39 |
| | LEU in HEU Core | CFE-1 | 1.45 | 1.35 | 1.22 | 1.34 |
| | | SFE-1 | 1.55 | 1.44 | 1.48 | 1.50 |
| LEU Core | LEU in LEU Core | none | 1.56 | 1.52 | 1.85 | 1.58 |
| | | CFE-1 | 1.33 | 1.24 | 1.12 | 1.24 |
| | | SFE-1 | 1.55 | 1.48 | 1.85 | 1.51 |
| | | | **Radial × Local** | | | |
| HEU Core | HEU in HEU Core | none | 1.46 | 1.44 | 1.47 | 1.48 |
| | | CFE-1 | 1.81 | 1.66 | 1.32 | 1.67 |
| | | SFE-1 | 1.59 | 1.57 | 1.64 | 1.59 |
| | LEU in HEU Core | CFE-1 | 2.16 | 1.99 | 1.57 | 1.97 |
| | | SFE-1 | 1.87 | 1.82 | 1.90 | 1.88 |
| LEU Core | LEU in LEU Core | none | 1.58 | 1.57 | 1.97 | 1.56 |
| | | CFE-1 | 1.75 | 1.56 | 1.23 | 1.60 |
| | | SFE-1 | 1.71 | 1.66 | 2.13 | 1.72 |

It should be noted that the local peaking factors were computed in a different manner by each of the contributors. Individual appendices should be consulted for descriptions of the methods used. In order to calculate the maximum peaking factors, care should be taken to choose a small enough mesh width and extrapolate to the peak power value or to modify computer codes to edit power densities based on flux values at the edges of mesh intervals rather than at the centers of mesh intervals. The latter approach yields peak power densities that are reasonably independent of the mesh width.

Axial power peaking due to partially-withdrawn control absorbers is also of interest. In many light-water-moderated MTRs, the axial peaking factor is 1.30 - 1.35 with the control absorbers fully-withdrawn. Three dimensional calculations of the BOC benchmark cores with the absorbers at different bank positions show that the peak axial power density is obtained when the abosorbers are 50% withdrawn and that the peak is located at a height of about 20 cm from the bottom of the active core. Axial power density profiles in SFE-1 and CFE-1 are shown in Fig. 7.2 for this case. Table 7.6 lists the peak values of the power density with the absorbers 50% withdrawn and 100% withdrawn.

Table 7.6 Peak Power Densities ($W/cm^3$) in CFE-1 and SFE-1 for HEU and LEU BOL Cores with Control Rods Withdrawn 50% and 100%.

| Control Rod Position | CFE-1 | | | SFE-1 | | |
|---|---|---|---|---|---|---|
| | HEU | LEU | LEU/HEU | HEU | LEU | LEU/HEU |
| 50% Out | 258 | 249 | 0.97 | 277 | 289 | 1.04 |
| 100% Out | 222 | 218 | 0.98 | 238 | 252 | 1.06 |
| Ratio $\frac{50\%\ \text{Out}}{100\%\ \text{Out}}$ | 1.16 | 1.14 | | 1.16 | 1.15 | |

The peak power densities in all four cases are about 15% larger with the rods 50% withdrawn rather than 100% withdrawn. Thus, an axial peaking factor of 1.50 - 1.55 is appropriate in 2D calculations to account for the axial power bulge with the control absorbers 50% withdrawn.

### 7.2.5 Control Rod Worths

The results of the calculations of control rod worths are shown in Table 7.7 for the HEU and LEU cores with fresh fuel and in Table 7.8 for the specified HEU and LEU BOC cores. An excellent agreement between the different methods is observed.

With LEU fuel, the effectiveness of the control rods (measured in $) decreases by about 5 - 8% for the fresh-fuel core and by about 10 - 15% for burned fuel in the BOC benchmark core.

The results of calculations (see Appendix G-1) with a 3D model of the HEU and LEU BOC cores with different control rod bank positions are shown in Fig. 7.3. The total reactivity worth in dollars is about 11% smaller in the LEU case, but the shape of the curves is very similar. The HEU and LEU cores would be critical with the rods withdrawn about 64% and 68%, respectively.

Fig. 7.2 HEU and LEU BOL CORES: CFE-1 and SFE-1
Axial Power Densities at Midplane Power Peak
with the Four Control Rods 50% Withdrawn

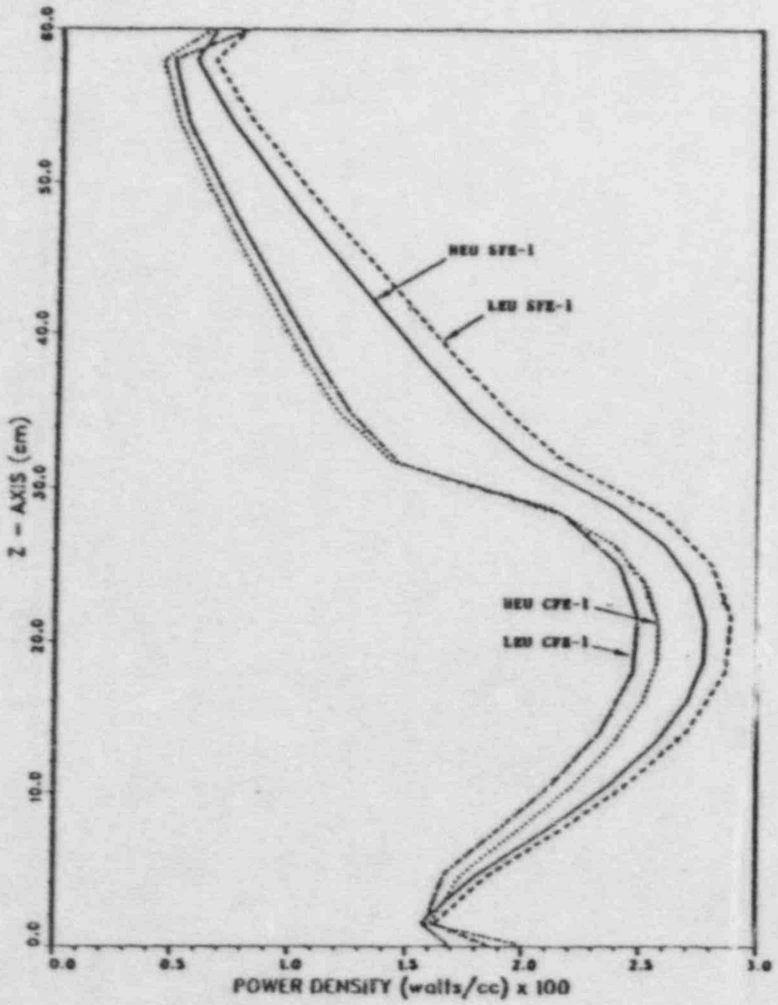



Fig. 7.3 Reactivity ($) vs. Rod Position
for HEU and LEU BOL Benchmark Cores.

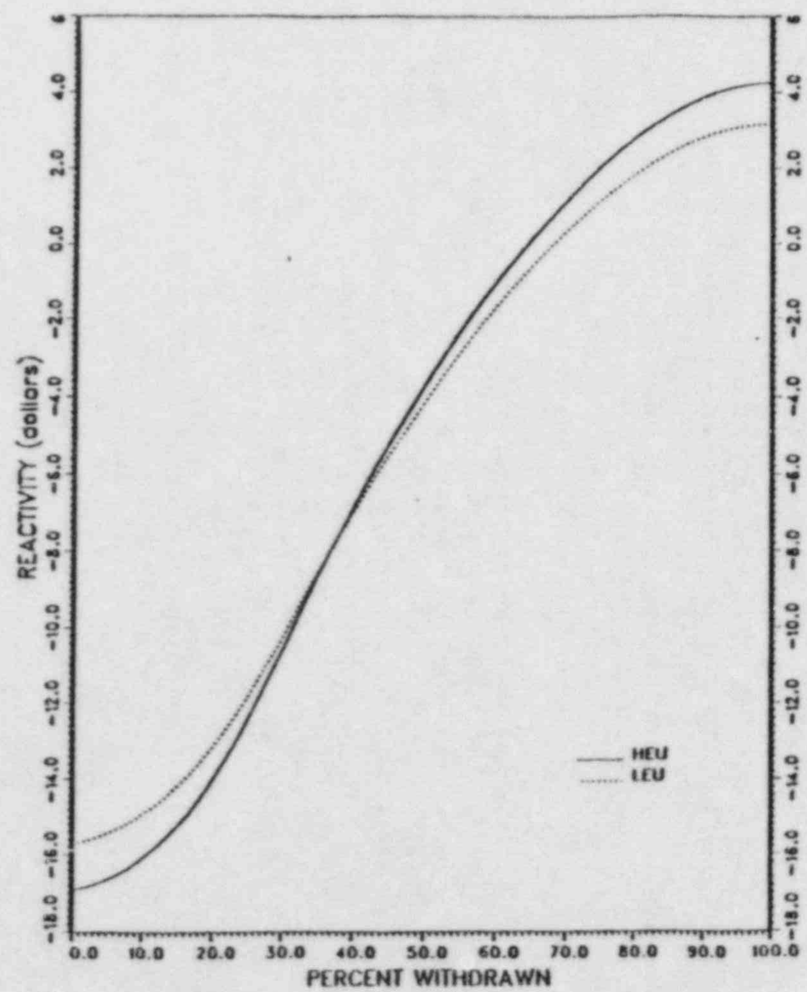


Table 7.7 Control Rod Worths for Fresh Fuel Cores (Δρ)

| | Absorber | ANL Monte Carlo %, $ | ANL Diffusion % | ANL Diffusion $ | INTERATOM % | INTERATOM $ |
|---|---|---|---|---|---|---|
| HEU Core | Ag-In-Cd | 13.43 ± 0.38<br>17.65 ± 0.50 | 13.0 | 17.1 | 13.3 | 17.5 |
| | $B_4C$ | 16.82 ± 0.38<br>22.11 ± 0.50 | 17.0 | 22.4 | 17.2 | 22.6 |
| | Hf | 12.70 ± 0.36<br>16.70 ± 0.47 | 12.6 | 16.6 | | |
| LEU Core | Ag-In-Cd | 11.24 ± 0.38<br>15.45 ± 0.52 | 11.5 | 15.9 | 11.7 | 16.0 |
| | $B_4C$ | 14.95 ± 0.40<br>20.55 ± 0.55 | 15.4 | 21.2 | 15.3 | 20.9 |
| | Hf | 11.07 ± 0.36<br>15.22 ± 0.49 | 11.2 | 15.4 | | |
| Difference HEU-LEU | Ag-In-Cd | 2.19 ± 0.54<br>2.20 ± 0.72 | 1.5 | 1.2 | 1.6 | 1.5 |
| | $B_4C$ | 1.87 ± 0.55<br>1.56 ± 0.74 | 1.6 | 1.2 | 1.9 | 1.7 |
| | Hf | 1.63 ± 0.51<br>1.48 ± 0.68 | 1.4 | 1.2 | | |

Table 7.8 Control Rod Worths for BOC Cores (Δρ)

| | Absorber | ANL (Diffusion) % | ANL (Diffusion) $ | INTERATOM % | INTERATOM $ | JAERI % | JAERI $ |
|---|---|---|---|---|---|---|---|
| HEU Core | Ag-In-Cd | 17.0 | 22.4 | 16.9 | 22.2 | 17.5 | 23.5 |
| | $B_4C$ | 21.7 | 28.6 | 21.3 | 28.0 | 23.1 | 31.0 |
| | Hf | 16.4 | 21.6 | | | | |
| LEU Core | Ag-In-Cd | 14.5 | 19.9 | 14.2 | 19.4 | 13.9 | 19.3 |
| | $B_4C$ | 18.9 | 26.0 | 18.3 | 25.0 | 19.0 | 26.4 |
| | Hf | 14.0 | 19.2 | | | | |
| Difference HEU-LEU | Ag-In-Cd | 2.5 | 2.5 | 2.7 | 2.8 | 3.6 | 4.2 |
| | $B_4C$ | 2.8 | 2.6 | 3.0 | 3.0 | 4.1 | 4.6 |
| | Hf | 2.4 | 2.4 | | | | |

Outside of the specified benchmark problem, calculations were also done (see Appendix G-2) to compare the relative effectiveness of fork-type and oval-type absorber designs. The fork-type absorber blades consisted of AgInCd-alloy with steel cladding and the oval-type absorbers consisted of natural boron carbide ($B_4C$) with a layer of cadmium. The fresh HEU fuel elements had 23 plates with a fissile content of 180 g. Models for the control fuel elements were typical of those that are used with the two absorber designs. The core was modeled as an infinite array with a repeating unit of five fuel elements and one control element. The total reactivity worth of the oval-type absorbers was computed to be 9.9% $\delta k/k$ and that of the fork-type absorber was computed to be 13.6 $\delta k/k$, an increase of about 36% in shutdown efficiency for the fork-type absorber.

7.2.6 Decay Heat Power

Calculated values of decay heat power versus time after shutdown (see Appendix G-2) are shown in Fig. 7.4 for the HEU and LEU cores with 10 MW initial power. For short shutdown times, the deviation between the two curves is small. For long times, the deviation can be as much as 40%, but the magnitude of the decay heat power is quite small.

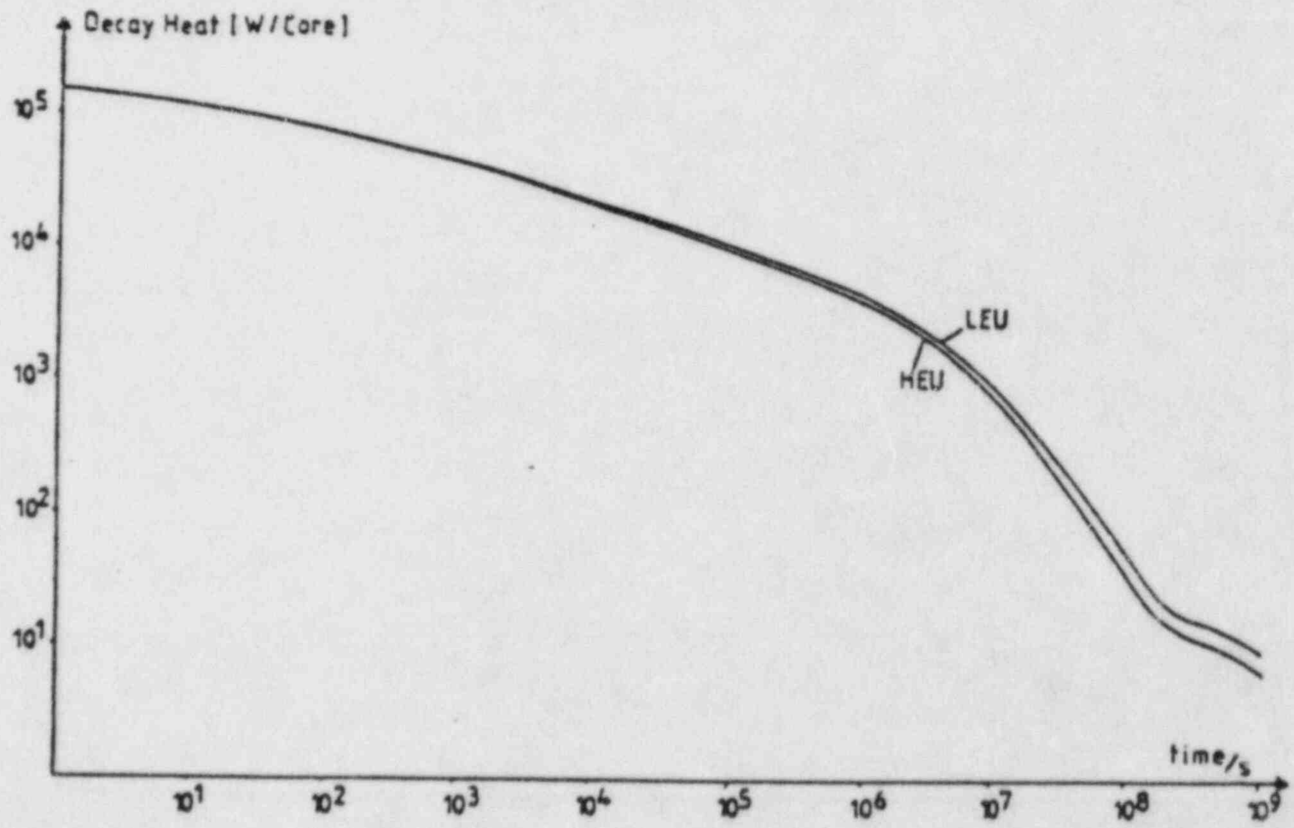



Fig. 7.4 Decay Heat versus Shutdown Time for HEU and LEU Benchmark Cores.

## 7.3 RESULTS OF TRANSIENT CALCULATIONS

Analyses of the behavior of the HEU and LEU benchmark cores were performed for the four specified transients:

- Fast Loss-of-Flow Transient
- Slow Loss-of-Flow Transient
- Slow Reactivity Insertion Transient
- Fast Reactivity Insertion Transient

Outside of the problem specifications, calculations were also performed for the $1.5/0.5s fast reactivity insertion transient to determine (1) the sensitivity of the results to uncertainties in the kinetics parameters and thermal conductivity, and (2) the effect of removing the specified scram.

In addition, the behavior of the HEU benchmark core for reactivity insertions leading to clad melting was compared with results for two SPERT I experimental cores (B-24/32 and D-12/25), and reactivity insertion limits for clad melting as a function of ramp duration were determined for both the HEU and LEU benchmark cores.

In the calculational models, the core was represented by two channels: one representing the average thermal-hydraulic behavior of the core and the other representing the hottest channel. The axial source distributions in both channels were represented using a number of regions, a chopped cosine shape, and the power peaking and engineering hot channel factors that were specified.

### 7.3.1 Loss-of-Flow Transients

Fast and slow loss-of-flow transients for the HEU and LEU cores were modeled with exponential flow decay and time constants of 1.0s and 25.0s, respectively. The transients were initiated from a power of 12 MW with a flow trip point at 85% of nominal flow and a 200 ms time delay before beginning a shutdown reactivity insertion of -$10 in 0.5s.

The results that were obtained are compared in Tables 7.9 and 7.10. Figures 7.5 and 7.6 show typical examples of the variation with time of the fuel centerline temperature, the clad surface temperature, and the coolant outlet temperature. The results show that there are almost no differences between the HEU and LEU cases. The peak surface clad temperature is far below the melting temperature of the cladding and the flow instability parameter, $\eta$, is much larger than its limiting value.

Table 7.9 Fast Loss-of-Flow Transient, 1.0-sec Time-Constant, Exponential Decay, Initial Power: 12 MW, Flow Trip Point: 85% of Nominal Flow

| | Core | ANL | INTERATOM | JAERI | JEN |
|---|---|---|---|---|---|
| Power Level at Scram, MW | HEU | 11.9 | 11.5 | 11.7 | 11.8 |
| | LEU | 11.9 | 11.4 | 11.7 | 11.7 |
| Peak Fuel Center-Line Temp., °C | HEU | 89.2 | 91.0 | 99.4 | 94.5 |
| | LEU | 90.3 | 91.9 | 98.7 | 95.4 |
| Peak Clad Surface Temp., °C | HEU | 87.5 | 89.5 | 98.4 | 94.0 |
| | LEU | 87.5 | 89.3 | 97.1 | 93.9 |
| Peak Coolant Outlet Temp., °C | HEU | 60.3 | 56.5 | 58.4 | 59.4 |
| | LEU | 60.3 | 56.4 | 58.1 | 59.3 |
| Minimum $\eta$, $cm^3$ K/Ws | HEU | 234 | 257 | | 268 |
| | LEU | 235 | 258 | | 262 |

Table 7.10 Slow Loss-of-Flow Transient, 25.0-sec Time-Constant, Exponential Decay, Initial Power: 12 MW, Flow Trip Point: 85% of Nominal Flow

| | Core | ANL | INTERATOM | JAERI | JEN |
|---|---|---|---|---|---|
| Power Level at Scram, MW | HEU | 11.6 | 11.6 | 11.6 | 11.8 |
| | LEU | 11.6 | 11.5 | 11.6 | 11.7 |
| Peak Fuel Center-Line Temp., °C | HEU | 85.8 | 87.4 | 97.4 | 91.2 |
| | LEU | 86.8 | 88.2 | 97.7 | 91.9 |
| Peak Clad Surface Temp., °C | HEU | 83.9 | 85.8 | 96.4 | 90.7 |
| | LEU | 83.7 | 85.5 | 96.1 | 90.3 |
| Peak Coolant Outlet Temp., °C | HEU | 58.9 | 55.6 | 57.7 | 58.3 |
| | LEU | 58.8 | 55.4 | 57.5 | 58.1 |
| Minimum $\eta$, $cm^3$ K/Ws | HEU | 270 | 293 | | 301 |
| | LEU | 271 | 295 | | 304 |

Fig. 7.5 Transient Responses of HEU and LEU Benchmark Cores to a Fast Loss-of-Coolant Flow with Decay Time of 1.0 sec.

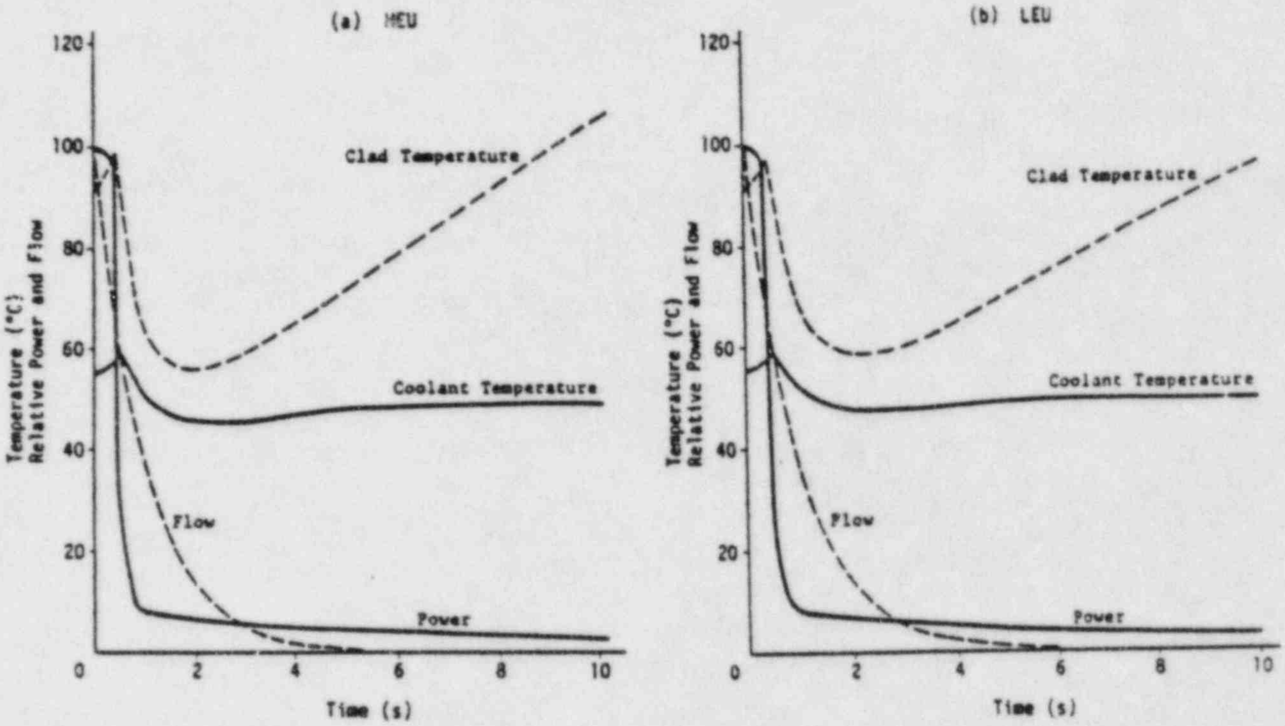
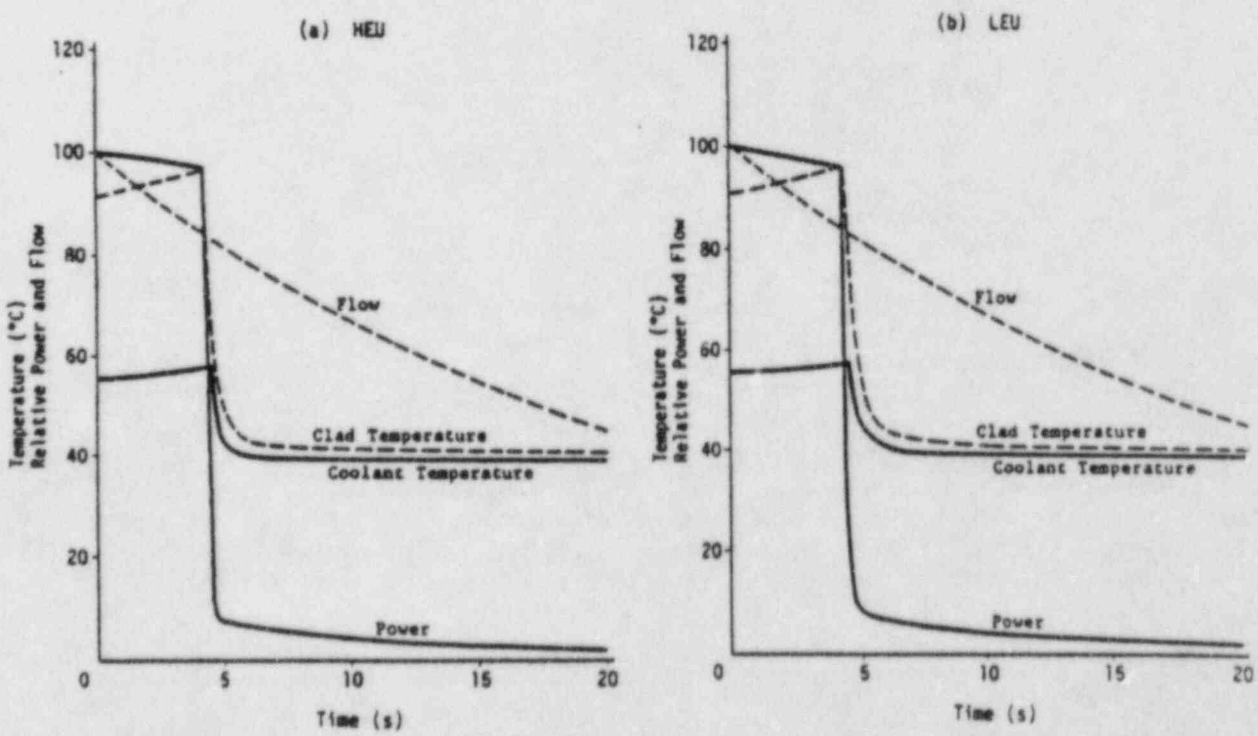



Fig. 7.6 Transient Responses of HEU and LEU Benchmark Cores to a Slow Loss-of-Coolant Flow with Decay Time of 25.0 sec.



### 7.3.2 Slow Reactivity Insertion Transient

The slow reactivity insertion transient was initiated by ramp rates of \$0.10/s in the HEU core and \$0.09/s in the LEU core starting with the reactor critical at an initial power of 1 W and full flow. The safety system trip point was 12 MW with a time delay of 25 ms before beginning a shutdown reactivity insertion of -\$10 in 0.5s.

The results are listed in Table 7.11 and an example of the power profiles and the temperature profiles at the clad surface, fuel centerline, and coolant outlet are shown in Fig. 7.7. The minimum periods, peak powers, and peak temperatures in the fuel, cladding, and at the coolant outlet are in very good agreement, but there are differences among contributors in the energy release to peak power.

The HEU transient reaches the 12 MW trip point about one second earlier and has a higher peak power than the LEU case because the reactivity feedback is smaller. The LEU case has a much broader burst shape because the strong prompt feedback from the Doppler component plays a significant role. Thus, even though the peak power in the LEU case just exceeds the trip point, the energy released to the time of peak power is larger and the peak temperature at the surface of the cladding is about 78°C instead of about 69°C in the HEU case. However, both peak cladding temperatures are well below the melting temperature of the cladding. The energy released beyond 12 MW is significantly larger in the HEU case.

### 7.3.3 Fast Reactivity Insertion Transients

The fast reactivity transients were initiated by ramp insertions of \$1.5/0.5s in the HEU core and \$1.5/0.5s and \$1.35/0.5s in the LEU core starting with the reactor critical at an initial power of 1 W and full flow. As for the slow reactivity insertion transient, the safety system trip point was 12 MW with a time delay of 25 ms before beginning a shutdown reactivity insertion of -\$10 in 0.5s.

The results for the HEU and LEU cases with \$1.5/0.5s are compared in Table 7.12 and examples of the power and temperature profiles are shown in Fig. 7.8. Overall, the data are in very good agreement.

Since the LEU core has a shorter prompt neutron generation time and thus a smaller minimum period, the peak power is reached slightly earlier. The power burst for the LEU core is slightly narrower than in the HEU core, and even though the peak power is slightly higher for the LEU case, the energy release is lower. The prompt Doppler feedback from the LEU fuel does not play a significant role in these fast transients with scram.

The peak fuel centerline temperature is about 13 - 16°C higher in the LEU core, mainly due to the smaller specified thermal conductivity of the LEU fuel meat. The peak clad surface temperature is a few degrees higher and the peak coolant outlet temperature is about the same or a few degrees lower in the LEU case. A brief period of localized nucleate boiling was predicted for the hot channel in both cores. Overall, there are no significant differences between the HEU and LEU results for this transient.

Table 7.11 Slow Reactivity Insertion Transient
Ramps of 0.10 $/s for HEU, 0.09 $/s for LEU
Initial Power: 1W; Flow Rate: 1000 $m^3/h$
Trip Point: 12 MW; Time Delay: 25 ms

| | Core | ANL | INTERATOM | JAERI | JEN |
|---|---|---|---|---|---|
| Minimum Period, s | HEU | 0.10 | 0.10 | 0.10 | 0.10 |
| | LEU | 0.11 | 0.11 | 0.11 | 0.11 |
| Peak Power, MW | HEU | 14.1 | 14.4 | 13.8 | 14.9 |
| | LEU | 12.4 | 12.2 | 12.4 | 13.0 |
| Energy Release to Peak Power, MJ | HEU | 1.74 | 1.53 | 1.75 | 1.63 |
| | LEU | 4.55 | 5.94 | 4.69 | 2.10 |
| Peak Fuel Center-Line Temp., °C | HEU | 70.6 | 70.5 | 70.5 | 69.9 |
| | LEU | 80.6 | 80.8 | 81.2 | 73.2 |
| Peak Clad Surface Temp., °C | HEU | 69.0 | 69.2 | 69.2 | 69.5 |
| | LEU | 77.7 | 78.1 | 78.5 | 71.9 |
| Peak Coolant Outlet Temp., °C | HEU | 48.1 | 45.2 | 47.7 | 47.5 |
| | LEU | 53.9 | 51.1 | 52.8 | 48.8 |
| Energy Released Beyond 12 MW, kJ | HEU | | 50 | | 76 |
| | LEU | | 2 | | 19 |
| Minimum $\eta$, $cm^3$ K/Ws | HEU | | 483 | | 537 |
| | LEU | | 374 | | 502 |
| (t=20 sec) P, kW | HEU | 5 | | 6 | 7 |
| | LEU | 15 | | 15 | 9 |
| E, MJ | HEU | 2.29 | | 2.35 | 2.20 |
| | LEU | 5.30 | | 5.48 | 2.66 |

Fig. 7.7 Transient Responses of the HEU and LEU Benchmark Cores to a Slow Reactivity Insertion

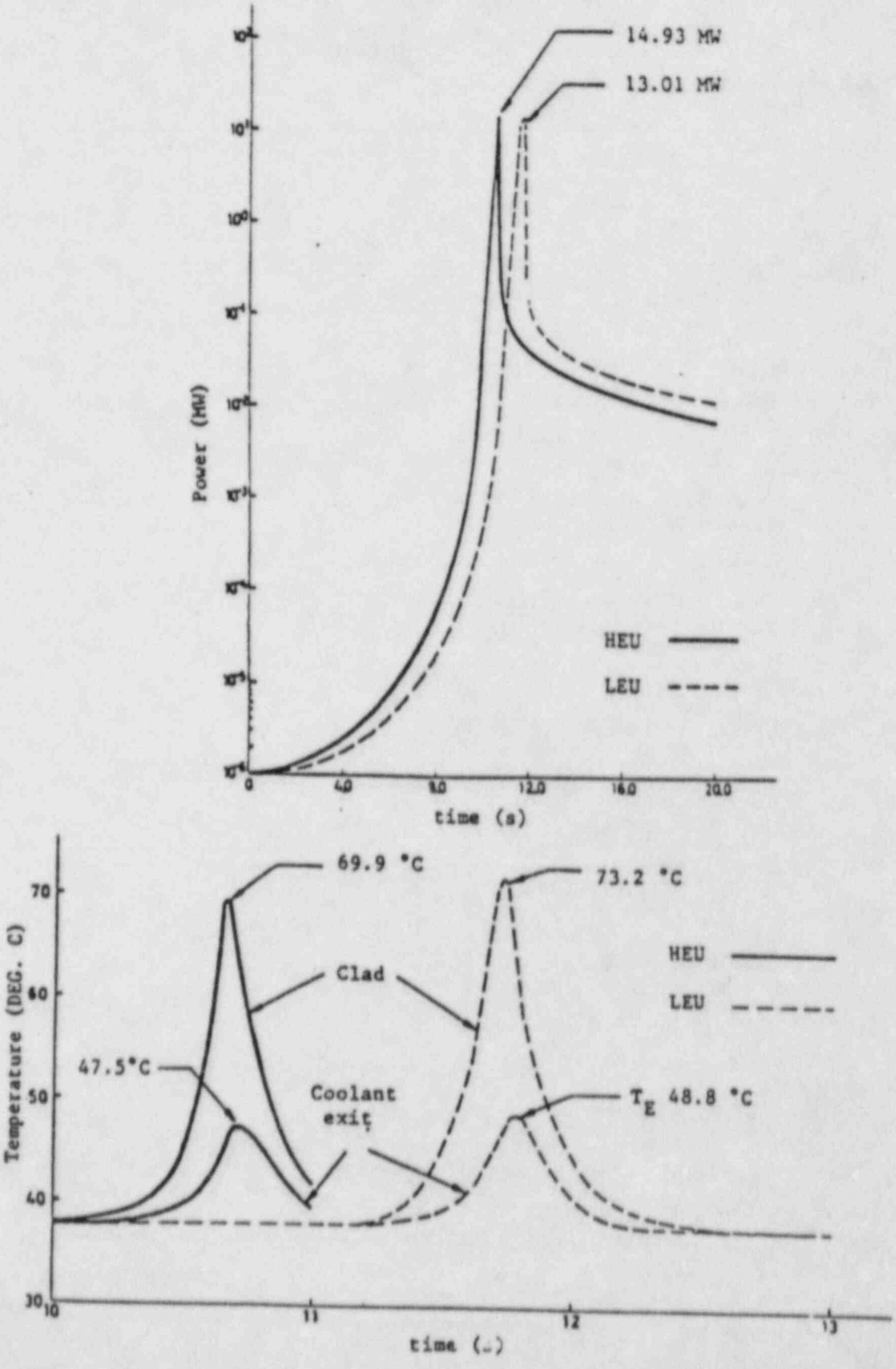


Table 7.12 Fast Reactivity Insertion Transient
Ramp of 1.5 \$/0.5s for HEU and LEU Cores
Initial Power: 1W; Flow Rate: 1000 $m^3/h$
Trip Point: 12 MW; Time Delay: 25ms

| | Core | ANL | INTERATOM | JAERI | JEN |
|---|---|---|---|---|---|
| Minimum Period, ms | HEU | 15 | 14 | 15 | 14.5 |
| | LEU | 12 | 12 | 12 | 13.5 |
| Peak Power, MW | HEU | 132 | 135 | 115 | 133 |
| | LEU | 148 | 144 | 144 | 116 |
| Energy Release to Peak Power, MJ | HEU | 3.26 | 3.14 | 2.86 | 3.47 |
| | LEU | 2.95 | 2.83 | 2.95 | 2.62 |
| Peak Fuel Centerline Temp., °C | HEU | 171 | 173 | 155 | 167 |
| | LEU | 183 | 186 | 171 | 166 |
| Peak Clad Surface Temp., °C | HEU | 156 | 160 | 147 | 162 |
| | LEU | 157 | 168 | 149 | 157 |
| Peak Coolant Outlet Temp., °C | HEU | 84 | 71 | 62 | 109 |
| | LEU | 82 | 63 | 63 | 80 |
| Minimum $\eta$, $cm^3K/Ws$ | HEU | | 34 | | 36 |
| | LEU | | 46 | | 58 |

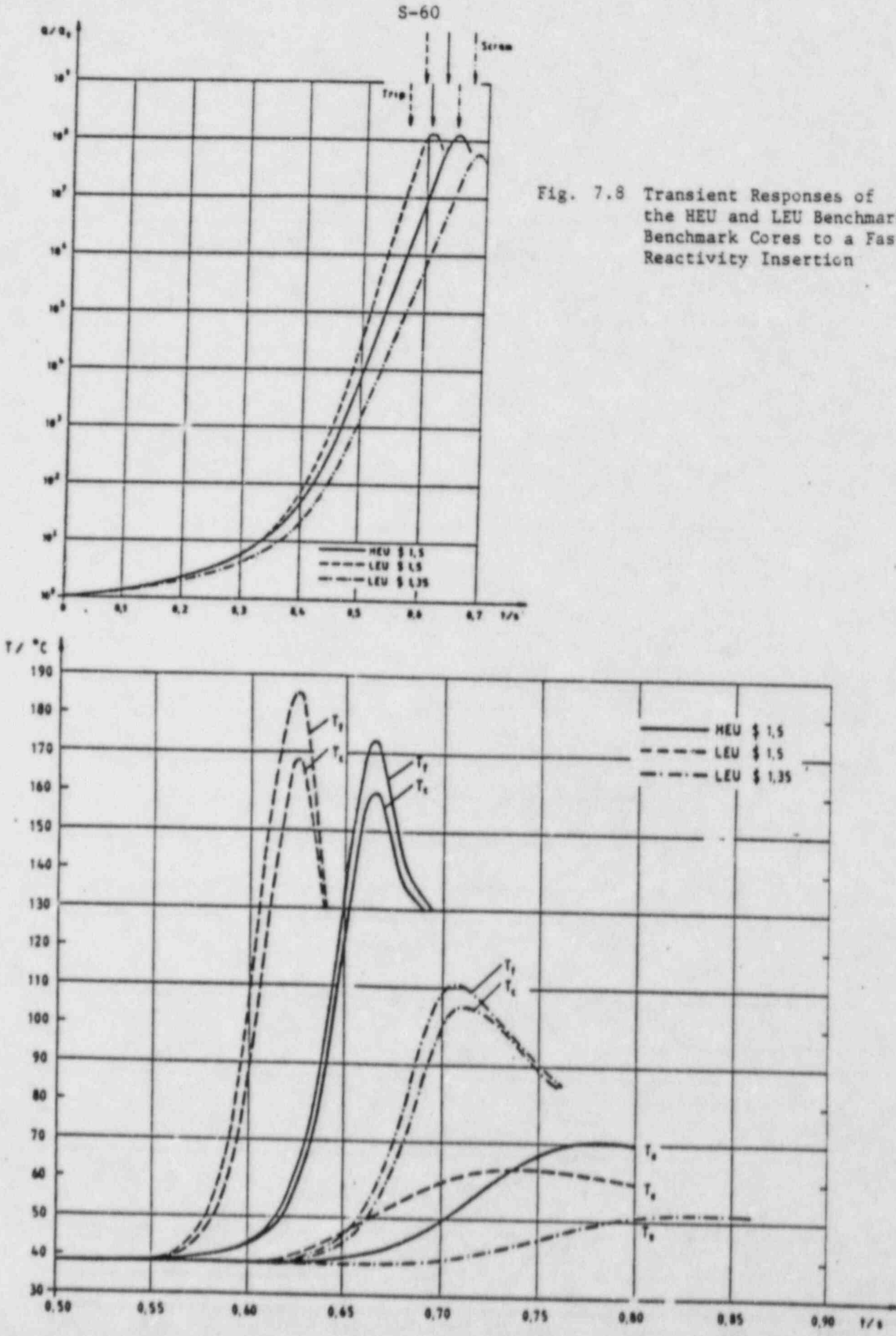



Fig. 7.8 Transient Responses of the HEU and LEU Benchmark Benchmark Cores to a Fast Reactivity Insertion

### 7.3.4 Sensitivity of Results to Variations in Thermal Conductivity and Kinetics Parameters

In this section, the influence of variations in the thermal conductivity of the LEU fuel meat and in some of the kinetics parameters of the HEU core is considered for the $1.5/0.5s fast reactivity insertion transient. Detailed results provided in Appendix G-1 are shown in Table 7.13. Only the parameter indicated was changed in each case.

The thermal conductivity of the LEU fuel meat in the LEU BOC core was varied from the 0.5 W/cmK value in the benchmark specifications to a maximum of 1.5 W/cmK. As expected, the largest change occurred in the peak fuel temperature. The smallest change occurred in the peak temperature at the surface of the cladding. In changing the thermal conductivity from 0.5 to 1.5 W/cmK, for example, the peak fuel temperature decreased by 6.3% from 183°C to 172°C and the peak clad temperature increased by only 0.3%. Thus, uncertainties in the thermal conductivity would not have a significant impact on the LEU benchmark conclusions.

The effect of variations in the kinetics parameters $\Lambda$ and $\beta$ were addressed by changing the base values by 10% in the HEU BOC core. The magnitude of the changes in the results are larger for a 10% decrease in $\Lambda$ than for a 10% increase in $\Lambda$. Since the inverse period for a super-prompt-critical step insertion is proportional to $\beta/\Lambda$, increasing $\beta$ is approximately equivalent to decreasing $\Lambda$ by the same amount.

Changing the reactivity feedback of the moderator (water temperature and density) in the HEU core by ± 10% changes the peak fuel temperature by ∓ 0.4% and the peak temperature at the surface of the cladding by ∓ 0.2%. Again, the conclusions of the benchmark studies would not be affected by these changes.

Table 7.13 Sensitivity of Results for the $1.50/0.5s Fast Reactivity Insertion Transient to Variations in Thermal Conductivity, Kinetics Parameters, and Moderator Feedback Coefficient

| Parameter | BOC Core | Change in Parameter | Peak Power | Energy Release to Peak Power | Peak Fuel Temp. | Peak Clad Temp. | Peak Coolant Outlet Temp. |
|---|---|---|---|---|---|---|---|
| | | | Relative Changes, % | | | | |
| Thermal Conductivity, W/cmK | LEU | 0.5→1.0 | +1.0 | | -4.7 | +0.2 | +0.7 |
| | | 0.5→1.5 | +1.3 | | -6.3 | +0.3 | +0.9 |
| Prompt Neutron Gen. Time | HEU | +10% | -19.0 | -11.3 | -5.1 | -2.2 | -10.3 |
| | | -10% | +27.8 | +21.2 | +6.6 | +2.7 | +13.3 |
| $\beta_{eff}$ | HEU | +10% | +24.3 | +19.1 | +5.7 | +2.4 | +11.6 |
| Moderator Feedback Coefficient | HEU | +10% | -1.4 | -1.2 | -0.4 | -0.2 | -1.3 |
| | | -10% | +1.4 | +4.2 | +0.4 | +0.2 | +1.4 |

### 7.3.5 Self-Limited Transients

Although the transients specified for the benchmark cores do not include self-limiting cases, it is of interest to some reactor operators to consider cases where the specified scram is removed. Table 7.14 taken from Appendix G-1 provides a comparison of both the HEU and LEU benchmark cores for both protected and unprotected transients of $1.50/0.5 s. Differences in the prompt neutron generation time (Λ) and the Doppler coefficient are largely responsible for the observed differences in the results.

In the cases with scram, the influence of the larger Doppler coefficient for the LEU core is overshadowed by the negative reactivity from the insertion of control rods. The smaller Λ for the LEU core yields a shorter initial period and a faster rise in power. Consequently, the LEU case with scram shows a slightly higher peak power than the HEU case. However, the peak temperatures reached at the clad surface are very similar in both cases.

In the unprotected (self-limited) transients, the influence of the large Doppler feedback in the LEU core is apparent. All of the values recorded are substantially lower for this LEU case. The larger void/density coefficient with LEU also contributes to the differences shown. The maximum clad surface temperature in all cases is substantially below the melting point of the clad.

Table 7.14 Self-Limited Transients: $1.50/0.5 s Cases With and Without Scram for HEU and LEU Cores

| | Case | Period, ms | $\hat{P}$, MW ($t_m$, s) | $E_{t_m}$, MWs | $\hat{T}_{clad}$, °C at $t_m$ | $\hat{T}_{clad}$, °C Max. (t, s) |
|---|---|---|---|---|---|---|
| HEU | With Specified Scram | 14.5 | 132 (0.656) | 3.26 | 131 | 156 (0.672) |
| | Self-limited | 14.5 | 371 (0.667) | 7.30 | 220 | 308 (0.685) |
| LEU | With Specified Scram | 11.9 | 148 (0.613) | 2.95 | 126 | 157 (0.628) |
| | Self-limited | 11.9 | 283 (0.622) | 5.56 | 181 | 263 (0.642) |

Reactivity Coefficients and Parameters

| | Λ, μs | $\beta_{eff}$ | Coolant Temperature $/°C | Void/density, $/% Void | Doppler, $/°C |
|---|---|---|---|---|---|
| HEU | 55.96 | 7.607-3 | 1.537-2 | 0.3257 | 3.6-5 |
| LEU | 43.74 | 7.275-3 | 1.082-2 | 0.4047 | 3.31-3 |

### 7.3.6 Clad Temperature Limits Compared with SPERT I Experiments

Appendix G-1 contains a comparison of measurements and calculations for two HEU SPERT I experimental cores (B-24/32 and D-12/25) in order to validate the PARET/ANL code for transient calculations in which the temperature of the cladding reaches its melting point. The code was then used to predict the reactivity insertions (as a function of ramp duration) that would lead to clad melting in the HEU and LEU benchmark cores.

Figure 7.9 shows the measured and calculated data for the SPERT I D-12/25 core (12 plates per element, 25 elements). The D-12/25 core included destructive tests which indicated extensive plate melting for inverse periods greater than ~ 166 $s^{-1}$ (~ \$2.36 insertion). Also shown in Fig. 7.9 are results for the step reactivity insertion (~ \$2.35) that would lead to clad melting in the HEU benchmark core. The agreement with experiment is remarkably good even though the D-12/25 core and the HEU benchmark core have somewhat different characteristics. This similarity of behavior was also noted in the diverse cores considered in the SPERT I series of experiments. The damage line in Fig. 7.9 (~ 140 $s^{-1}$) shows the threshold for clad damage from thermal stress.

Figure 7.10 provides a comparison of the HEU and LEU benchmark cores showing the clad melting threshold for reactivity insertions over a range of ramp durations from a step to 0.75 s. The areas above the curves indicate where clad melting would be expected. Also shown in this figure is the corresponding maximum net reactivity inserted (the difference between the external reactivity inserted and the reactivity from feedback). This maximum generally occurs at the same time in the transient as the minimum period.

The curves in Fig. 7.10 show clearly that the LEU core can tolerate a larger reactivity insertion before clad melting than the HEU core. The maximum step insertion is ~ \$2.80 for the LEU core compared to ~ \$2.35 for the HEU core. The ramp insertions of short duration are equivalent to a step insertion since the entire ramp is inserted before the power, temperatures, and feedback have increased substantially, and the limiting reactivity insertion remains constant. For ramps of longer duration, the feedback reactivity limits the net reactivity and turns over the transient before the maximum of the ramp is reached. A limiting ramp rate (constant slope) is reached, and a constant maximum net reactivity is observed for each case. The limiting ramp rate for the LEU core (~ 14.8 \$/s) is more than twice that for the HEU core (~ 6.4 \$/s). The LEU core also shows an earlier transition from the limiting step portion of the curve to the limiting ramp rate range.

In order to quantify the effect of differences in the HEU and LEU feedback coefficients on the limiting reactivity insertions, the LEU case with a 0.5s ramp duration was redone first with zero Doppler coefficient and second with zero Doppler coefficient and the HEU void coefficent. The results show that about 67% of the difference between the HEU and LEU reactivity insertion limits is due to the LEU Doppler coefficient and that about 28% of the difference is due to the larger void coefficient in the LEU core. The remaining 5% difference can be attributed to differences in other parameters such as $\Lambda$ and $\beta_{eff}$. The benefits of a prompt Doppler coefficent with LEU fuel are clearly demonstrated by these results.

Fig. 7.9 Comparison of PARET Calculations with Measurements in the SPERT I D-12/25 Core.

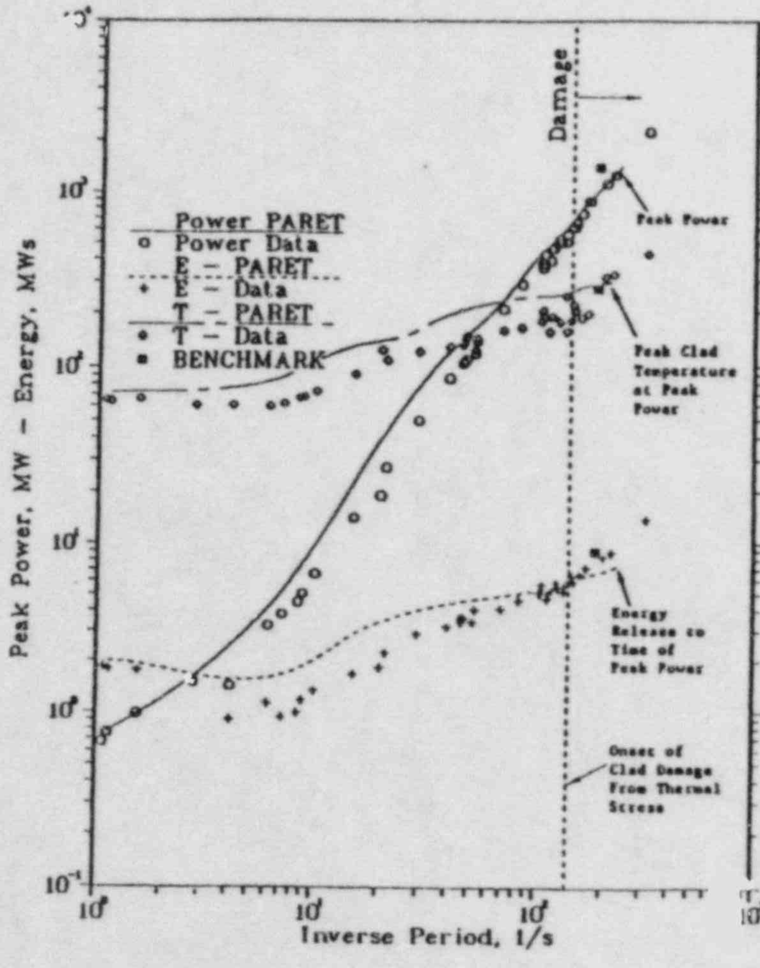
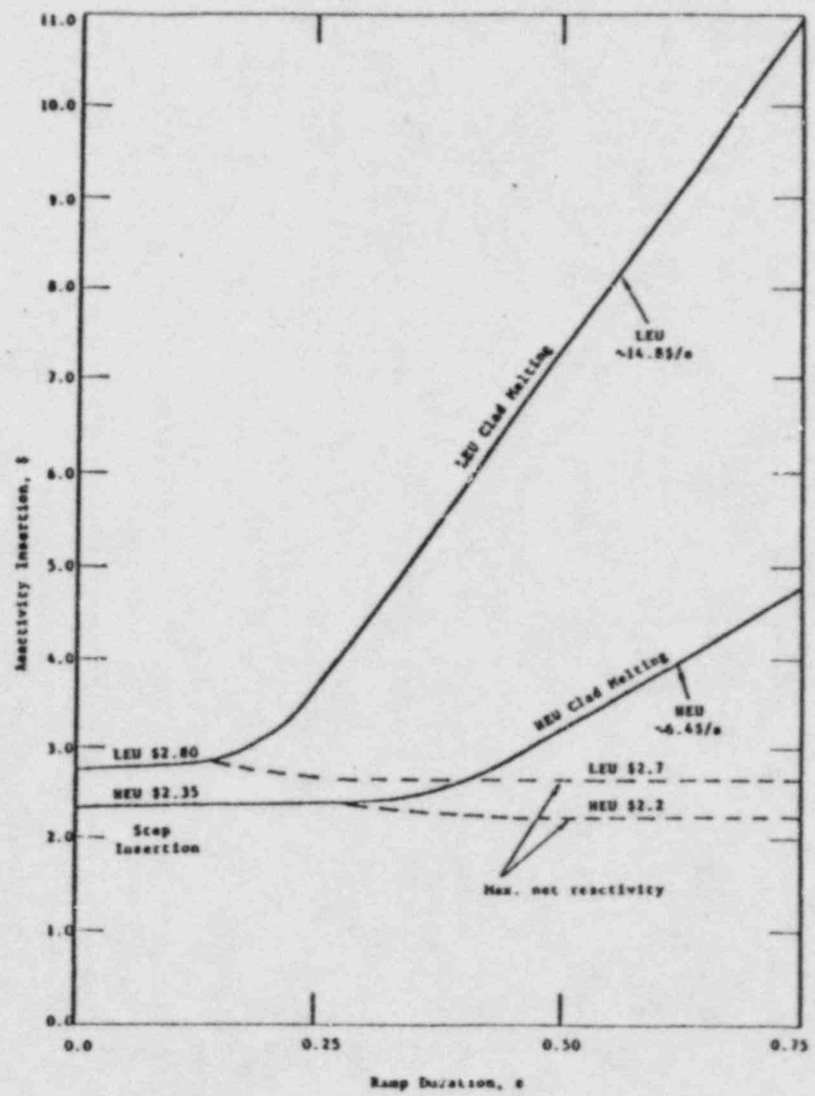



Fig. 7.10 Reactivity Insertion Limits for Clad Melting in the HEU and LEU Benchmark Cores.



## 7.3.7 Self-limiting Transients in Heavy Water Moderated Research Reactors

The methods and models for reactivity transient calculations developed at the AAEC's Lucas Heights Research Laboratories, Sydney, are briefly described in Appendix G-6. They were validated for application to HIFAR by comparison with experimental transient data from the very similar SPERT II BD22/24 heavy water moderated core. Experimental and calculated transient parameters for this core are compared in Fig. 7.11.

By use of relatively minor modifications of the model, transient parameters for HIFAR under zero, low power and operating coolant flow modes were calculated and are given in Appendix G-6. It can be expected that very similar results would apply to the 10 MW benchmark reactor discussed in IAEA-TECDOC-324, because it so closely resembles HIFAR.

Although the data presented are solely for HEU fuelling, they demonstrate, particularly in Fig. 7.11, that there are validated methods of transient estimation for heavy water moderated research reactors, comparable to those available for light water moderated research reactors.

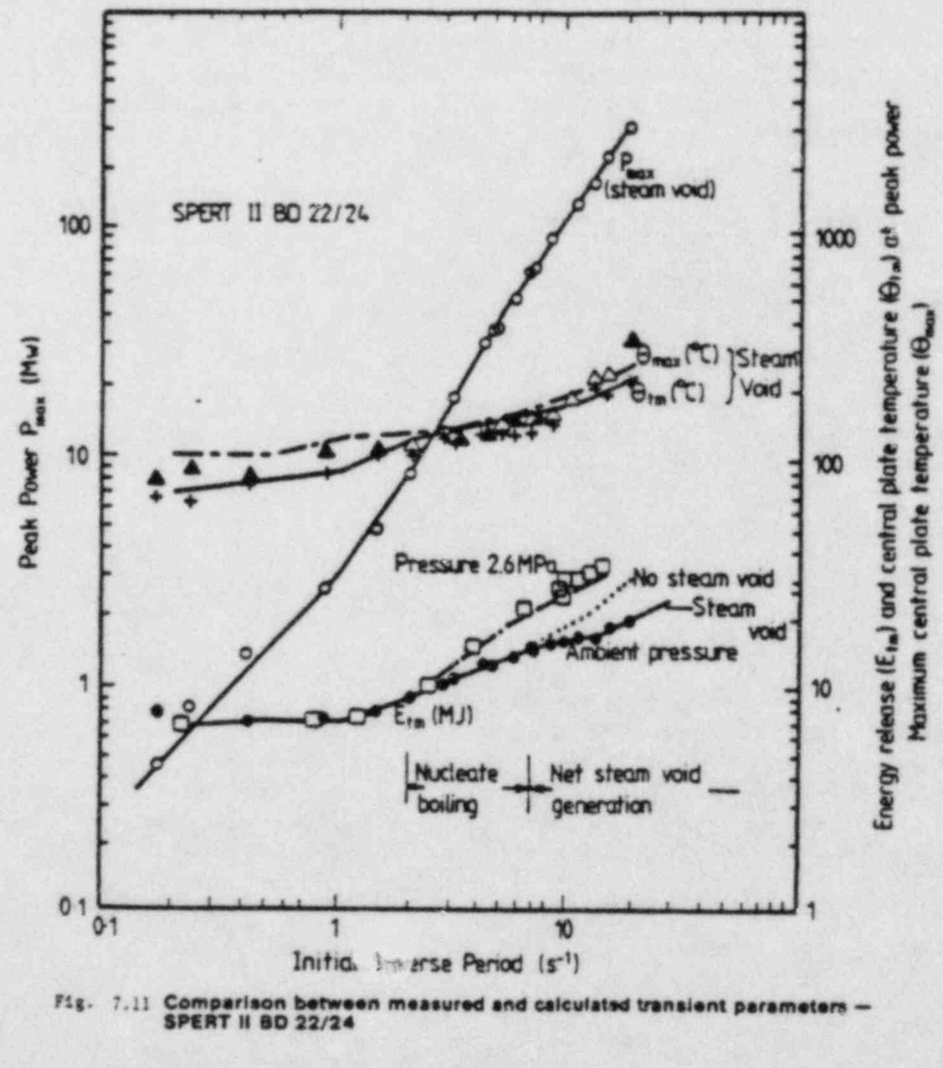



Fig. 7.11 Comparison between measured and calculated transient parameters — SPERT II BD 22/24

# CHAPTER 8

## COMPARISON OF CALCULATIONS WITH MEASUREMENTS

In accordance with core conversion, several institutions have completed experiments using LEU or MEU fuel to determine the accuracy of neutronic calculations. The experimental results were compared with the calculations, most methods of which are reported in IAEA-TECDOC-233. Detailed results are contained in Appendix H.

### Full Core Comparisons

#### ISIS/CEA-Saclay (App. H-1)

Critical experiments with Caramel fuel for three enrichments of LEU were performed by the CEA in the ISIS reactor at Saclay. The calculations show an excellent agreement with experiments for reactivity and control rod worth, but overestimated neutron flux in the core, except for the first outer periphery of the core.

#### KUCA/KURRI (App. H-2)

KURRI supplied a variety of data on full-core critical experiments with MEU and HEU plate-type fuels in the Kyoto University Critical Assembly (KUCA). The experimental results were analyzed using independent ANL and JAERI code systems. The comparison between calculations and experiments were very good for criticality in various core geometries, for flux distributions, and for void coefficients.

#### FNR/University of Michigan (App. H-3)

A series of experiments on a full core of LEU and mixed cores of LEU and HEU fuel of plate type were carried out up to about 7% burnup of the LEU elements.

The LEU $UAl_x$-Al elements were manufactured by NUKEM and by CERCA. ANL and University of Michigan analyzed experiments on criticality, rod worth, flux distributions and reaction rate distributions, and a very good agreement between calculations and measurements was obtained.

### Single Element Comparisons

#### PCA/ORNL (App. H-4)

Prior to the full core experiments in the FNR at University of Michigan, a series of critical experiments using HEU, MEU, LEU and mixed fuels in the Pool Critical Assembly (PCA) at ORNL were performed. These measurements provided valuable data for comparison with research reactor calculation modelling for all fresh, mixed-enrichment, and all HEU fueled cores. The experiments included criticality, flux distribution, rod worth, fission rate, etc.

SAPHIR/EIR (App. H-5)

The reactivity value of an MEU element with 320 g $^{235}U$ was measured in each position of a 4 × 4 core arrangement of HEU fuel elements with 280 g $^{235}U$. The results agree reasonably well with two dimensional diffusion calculations.

DR 3/RISØ (App. H-6)

Calculations for full MEU and LEU cores in the Danish reactor DR 3 are compared with measurements on three MEU $UAl_x$-Al test fuel elements and three LEU $U_3O_8$-Al test fuel elements irradiated in HEU cores.

The fast/thermal flux ratios plotted against the $^{235}U$ content in each fuel element confirm the calculated ratios within a few percent. The reactivity losses owing to the increased $^{238}U$ content in the MEU and LEU elements, as compensated by increased $^{235}U$ content, were in good agreement with the calculated values. The comparisons comprise measurements up to more than 50% burnup of the fuel elements.

RA2/CNEA (App. H-7)

The calculated reactivity change of the control rods worths from HEU to LEU fuels in the RA-3 reactor was compared with measurements of the control rod worth in HEU fueled RA-2 core. The comparisons show reasonable agreement.

# CHAPTER 9

## FUEL MATERIALS DATA

See Vol III, [illegible]

# CHAPTER 10

## IRRADIATION AND POST-IRRADIATION-EXAMINATION (PIE) OF DISPERSION FUELS WITH HIGH URANIUM DENSITY

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# CHAPTER 11

## EXAMPLES OF FUEL SPECIFICATIONS AND INSPECTION PROCEDURES

see Vol III, the

# CHAPTER 12

## STARTUP PROCEDURES AND EXPERIMENTS

Appendix L contains information related to startup procedures and experiments for the conversion of a reactor facility from a highly enriched uranium core to lower enrichment.

These startup procedures and experiments for an LEU core and for an HEU core are in principle not different from each other. The experiments and measurements necessary would depend on the following:

- Completeness of the nuclear and thermohydraulic calculations
- Completeness of the dynamic and safety related calculations
- Comparison of the HEU and LEU core design
- Whether operation will be a mixed (HEU + LEU) core or only LEU core
- Changes in fuel element design
- Changes in control rod or system design

This contribution gives some recommendations for experiments believed to be necessary or optional. Additional information can be found in a technical document: "Core Instrumentation and Pre-Operational Procedures," IAEA-TECDOC-304 (1983).

The startup programme should be submitted for an independent review to the regulatory authority.

# CHAPTER 13

## EXPERIENCE WITH MIXED CORE OPERATION

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# CHAPTER 14

## TRANSPORTATION, SPENT FUEL STORAGE AND REPROCESSING

This chapter summarizes the contributions in Appendix N on transportation of fresh and spent fuel elements, spent fuel storage, and reprocessing. This information does not influence the core conversion procedure. However, some items may influence parameters such as the $^{235}U$ loading and in turn they can influence the licensing procedure.

### Transportation

The transportation of fresh fuel elements needs to be licensed. The IAEA recommendations on physical protection of nuclear material are contained in INFCIRC/274 (see also Chapter 16). In many countries, the requirements are more stringent. For transportation, the transport companies have developed different casks which are described in Appendix N-1. As fresh fuel transport casks are inexpensive, there are nearly no limiting factors for fresh fuel elements.

The transportation of spent fuel elements needs to be licensed. Since the casks are very expensive, the reactor operator should carefully consider the limiting conditions for the casks: decay heat, maximum $^{235}U$ loading, weight, etc. Usable casks are described in Appendix N-2. In some cases, it may be economical to cut the end boxes from the fuel elements and return only the fueled region of the fuel element to the reprocessing plant since reprocessing charges depend on the total delivered weight of the aluminum and uranium.

### Spent Fuel Storage

The subcriticality of spent fuel storage configurations needs to be reconfirmed. Example calculations for several storage rack configurations with various HEU and LEU $^{235}U$ loadings are provided in Appendix N-3.

### Reprocessing

The U.S. Department of Energy (DOE) will accept until the end of 1987, aluminide and oxide fuel elements with an initial enrichment >20%, and TRIGA fuel elements for reprocessing. Studies are in progress for an amended U.S. policy that includes acceptance of LEU aluminide, oxide, and silicide fuels. This policy is expected to be issued during 1985.

# CHAPTER 15

## RESEARCH REACTOR SAFEGUARDS

Contributed by IAEA

### 1.0 Introduction

A basic purpose of the Treaty on the Non-Proliferation of Nuclear Weapons, 1 July 1968 (NPT) is to prevent the spread of nuclear weapons. The NPT recognises the International Atomic Energy Agency as a major instrument in the implementation of the Treaty in this regard. Article III of the NPT states, in part, "Each non-nuclear-weapon State Party to the Treaty undertakes to accept safeguards, as set forth in an agreement to be negotiated and concluded with the International Atomic Energy Agency in accordance with the Statutes of the International Atomic Energy Agency and the Agency's safeguards system, for the exclusive purpose of verification of the fulfillment of its obligations assumed under this treaty with a view to preventing diversion of nuclear energy from peaceful uses to nuclear weapons or other nuclear explosive devices." The Agency's safeguards agreements and its application of safeguards are not aimed at nuclear weapons per se, but rather nuclear materials. In keeping with the intent of the NPT, the Agency's programme places major emphasis on the safeguarding of nuclear materials that can most readily be made into nuclear weapons. High enriched uranium is one such material and Agency resources can be conserved if this nuclear material is not used in applications where low enriched uranium would serve just as well.

In this section, the IAEA safeguards requirements for research reactors are briefly described and estimates are given for the Agency effort to safeguard research reactors fuelled with uranium of different enrichments. Based on Agency safeguards activities, costs are estimated for the State and facility operator to meet their IAEA safeguards obligations using four different facility fueling examples. It is shown that there can be a safeguards cost differential of one or more orders of magnitude between using high enriched and low enriched uranium.

### 2.0 International Safeguards System

### 2.1 Basic Safeguards Documents

The International Atomic Energy Agency applies safeguards according to Agency-State Safeguards Agreements based on the provisions of its Statute[1] and either INFCIRC/66/REV. 2,[2] or INFCIRC/153 (corrected)[3]. Although stated somewhat differently in each agreement, the objective of IAEA safeguards is to verify compliance with the undertakings of the Agreements. In Agreements based on INFCIRC/66/Rev. 2, Article III.A.5 of the Agency Statute is referenced

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1/ Statute, International Atomic Energy Agency, June 1, 1973.

2/ INFCIRC/66/Rev. 2. The Agency's Safeguards System (1965), As Provisionally Extended in 1966 and 1968, 16 September 1968.

3/ INFCIRC/153 (Corrected). The Structure and Content of Agreements Between the Agency and States Required in Connection with the Treaty on the Non-proliferation of Nuclear Weapons, June 1972.

as authorising the Agency to "establish and administer safeguards designed to ensure that special fissionable and other materials, services, equipment, facilities, and information made available by the Agency or at its request or under its supervision or control are not used in such a way as to further any military purpose." In Agreements according to INFCIRC/153 (corrected), Article 1, States undertake "to accept safeguards, in accordance with the terms of the Agreement, on all source or special fissionable material in all peaceful nuclear activities within its territory, under its jurisdiction or carried out under its control anywhere, for the exclusive purpose of verifying that such material is not diverted to nuclear weapons or other nuclear explosive devices." The objectives of safeguards as stated in Article 28 of INFCIRC/153 (corrected) are "the timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or of other nuclear explosive devices or for purposes unknown, and deterrence of such diversion by the risk of early detection."

## 2.2 Technical Objectives of Safeguards

Both INFCIRC statements relate the purpose of Agency safeguards to the detection and deterrence of the use of nuclear material for nuclear weapons. In order for this purpose to be served, the Agency applies safeguards according to the significant nature of the material and the time estimated to covert the material to the components of a nuclear weapon. A significant quantity of nuclear material is defined as that approximate quantity, taking into account any conversion process involved, such that the possibility of manufacturing a nuclear explosive device cannot be excluded.[4] Table I gives the numerical values of significant quantities for all nuclear materials under Agency safeguards.

The Agency considers conversion time should correspond in order of magnitude to detection time, defined as the maximum time that may elapse between diversion and its detection by IAEA safeguards. The Agency's safeguards programme and its distribution of resources, which directly relate to the effort a State with which the Agency has a safeguards agreement devotes to its safeguards programme, are roughly measured in terms of the State's nuclear material and its safeguards significance.

Tables I and II illustrate the differences between low enriched uranium (LEU) and high enriched uranium (HEU) in terms of safeguards concerns, i.e. significant quantities and conversion time. The reason for the differences is that HEU (uranium where the U-235 content is equal to or greater than 20 percent) can be used directly to make metallic components or a nuclear weapon. If the HEU is in the form of a compound or alloy, it can be separated out by chemical processing: this is considered to be a simple, straightforward procedure and the estimated conversion time for HEU as a compound is 1-3 weeks, as the metal it is 7-10 days. Low enriched uranium (uranium in which the U-235 content is less than 20 percent) cannot be made into the components of a nuclear weapon. LEU must be further enriched, or used in a reactor producing thermal power to produce plutonium. Neither further enrichment nor the reactor production of plutonium are considered straightforward; hence the estimate of one year for the conversion time of low enriched uranium.

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4/ IAEA Safeguards Glossary, IAEA/SG/INF/1, Vienna, June 1980.

TABLE I

Significant Quantities

| | Material | Significant Quantity | Safeguards Apply to |
|---|---|---|---|
| Direct-use nuclear material | Pu* | 8 kg | Total element |
| | U-233 | 8 kg | Total isotope |
| | U[U-235>20%] | 25 kg | U-235 contained |
| | - Plus rules for mixtures where appropriate - | | |
| Indirect-use nuclear material | U[U-235<20%]** | 75 kg | U-235 contained |
| | Th | 20 kg | Total element |
| | - Plus rules for mixtures where appropriate - | | |

*For Pu containing less than 80% Pu-238.

**Including natural and depleted uranium.

Conversion times are shown in Table II.[4]

TABLE II

Estimated Material Conversion Times to Finished Pu or U Metal Components

| Beginning material form | Conversion time |
|---|---|
| Pu, HEU or U-233 Metal | Order of days (7-10) |
| $PuO_2$, Pu $(NO_3)_4$, or other pure Pu compounds; HEU or U-233 oxide or other pure compounds; MOX or other non-irradiated pure mixtures containing Pu, U[(U-233 + U-235)>20%]; Pu, HEU and/or U-233 in scrap or other miscellaneous impure compounds | Order of weeks (1-3)* |
| Pu, HEU or U-233 in irradiated fuel** | Order of months (1-3) |
| U containing <20% U-235 and U-233; Th | Order of one year |

*This range is not determined by any single factor but the pure Pu and U compounds will tend to be at the lower end of the range and the mixtures and scrap at the higher end.

**Criteria for establishing the irradiation to which this classification refers are under review.

The higher the enrichment of HEU, the more suitable the material is for use in nuclear weapons. The higher the enrichment of LEU, the easier it is to enrich that material to 20 percent U-235 or more. The Agency takes enrichment into account in its implemented safeguards program through use of the term "effective kilogram." Effective kilogram is defined in both INFCIRC/66/Rev. 2 and INFCIRC/153 (Corrected) as a quantity of uranium obtained by taking:

(a) For plutonium, its weight in kilograms;

(b) For uranium with an enrichment of 0.01 (1%) and above, its weight in kilograms multiplied by the square of its enrichment;

(c) For uranium with an enrichment below 0.01 (1%) and above 0.005 (0.5%), its weight in kilograms multiplied by 0.0001; and

(d) For depleted uranium with an enrichment of 0.005 (0.5%) or below, and for Thorium, its weight in kilograms multiplied by 0.00005."

## 2.3 Basic Agency Safeguards Measures for Uranium

The specific details of the safeguards implemented at any facility are based upon the Agreement between the State and the Agency. There are many similarities in the safeguards programmes for both HEU and LEU. For example, records must be kept of transfers to and from the facility and of the location of material within the facility in terms of quantity and type of material; the facility inventory must be periodically reported to the Agency; changes in design information and changes in nuclear material inventory must be reported to the Agency. A system of records and reports and a programme of material accounting complemented by containment and surveillance devices is the basis for State-Agency Agreements for both high enriched and low enriched uranium. Also for both classes of material, the Agency conducts inspections to independently verify that the quantity of nuclear material reported as on inventory at the facility by the facility operator is actually at the facility. The effort devoted by Agency inspectors to this verification activity is determined by the quantity of nuclear material at the facility as measured in effective kilograms. Table III shows the relationship between nuclear material enrichment, weight in kilograms and effective kilograms.

TABLE III

Effective Kilogram Quantities for Uranium of Varying Enrichments

| MASS (kg) of Uranium | EFFECTIVE KILOGRAMS (ekg) | | |
|---|---|---|---|
| | 10% Enrichment | 20% Enrichment | 95% Enrichment |
| 10 | 0.1 | 0.4 | 9 |
| 20 | 0.2 | 0.8 | 18 |
| 50 | 0.5 | 2 | 45 |
| 100 | 1 | 4 | 90 |
| 500 | 5 | 20 | 451 |
| 1000 | 10 | 40 | 902 |
| 5000 | 50 | 200 | 4512 |

## 3.0 Safeguards Implementation at Research Reactors

### 3.1 Categorisation of Research Reactors According to Inventory

Both INFCIRC/66/Rev. 2 and INFCIRC/153 (Corrected) relate inspection effort to the inventory of nuclear material as measured in effective kilograms at the facility. The State-Agency Agreement guidelines in these documents have been studied by Agency staff. Agency inspection effort has been estimated for four different situations. These are summarised in Table IV.

Table IV shows both maximum effort and actual effort. The maximum effort is the maximum routine inspection effort the Agency would be authorised to spend at the facility. The actual effort is the actual routine inspection effort the Agency would most probably spend at the facility and would depend on the Agency-State Agreement and the real conditions of the Agency safeguards implementation programme.

**TABLE IV**

Estimated Agency Inspection Effort for Research Reactors (Mandays/year)

| Type and Quantity of Fuel | Maximum Effort | Actual Effort |
|---|---|---|
| a. Enrichment < 20% U-235; or Inventory less than 5 ekg. | 2 - 3 | 0.5 - 1 |
| b. Enrichment > 20% U-235; Inventory < 25kg U-235. | 50 | 1 - 2 |
| c. Enrichment > 20% U-235; Inventory > 25kg U-235, but < 25kg in fresh fuel. | 50 | 9 - 15 |
| d. Enrichment > 20% U-235; Inventory > 25kg U-235 in fresh fuel. | 50 | 50 |

3.2 Agency Inspection Effort

For research reactors in either of the first two categories in Table IV, the inspection effort would consist of an annual physical inventory verification. Inspection activities during a physical inventory include the following:

- establishment of the updated book inventory based on the examination of shipping and receiving documents;
- examination of records for self-consistency and verification of the consistency between records and reports;
- verification of the fresh and irradiated fuel assemblies or elements based on item counting and identification;
- checking, removing and reapplying seals as appropriate.

For research reactors in the third category, an annual PIV would be taken as above. In addition, interim inspection would be carried out four to six times per year. The activities performed by the inspectors during these inspections would include the following:

- establishment of the updated book inventory based on the examination of shipping and receiving documents;
- verification of the fresh and spent fuel elements;
- verification of the integrity of applied seals;
- activities related to installed surveilance units.

Essentially, the Agency inspection effort for a research reactor with this type and quantity of fuel is estimated to be the same as for a light water reactor.

For research reactors having fuel as described in the last category, it would be necessary to inspect the facility on a frequency corresponding to the 1-3 week detection time. i.e. routine inspections about every 3 weeks. In addition, because of the safeguards significance of the HEU, the Agency may recommend physical inventory verifications up to four times per year. All these activities by the Agency inspectors are estimated to require 50 mandays of effort.

3.3 Safeguards at Generic Research Reactor Fuelled with Different Uranium Enrichments

Changes in reactor core parameters as a result of converting the core from high enriched to low-enriched uranium have already been reported.[5] The Agency safeguards programme as it applies to HEU and LEU fuelled reactors can be illustrated by considering the generic 10 MW research reactor described in that report. Calculations were made for the fuel loadings for the reactor using uranium enriched to 93%, 20% and 6.5% in the U-235 isotope. These fuel loadings along with estimated new fuel requirements are shown in Table V.

5/ Research Reactor Core Conversion from the Use of Highly Enriched Uranium to the Use of Low Enriched Uranium Fuels. Guidebook IAEA-TECDOC-233, Vienna 1980.

TABLE V

Fuel Inventories for Generic 10 MW Reactor[5]

| Enrichment (% U-235) | New Fuel (kg U-235) | Core Fuel (kg U-235) | Spent Fuel (g Pu/Element) |
|---|---|---|---|
| 93 | 8 | 4.5 | 0.6 |
| 20 | 11.6 | 8 | 14 |
| 6.5 | 15 | 10 | 44 |

The U-235 content of the core fuel is calculated on the basis that the fissile material has been depleted to where new fuel must be added. Burnup is fifty to sixty percent and therefore the core fuel is safeguarded according to the measures applied to spent fuel. (See Tables I and II). Plutonium content has also been calculated for spent fuel and unless the spent fuel is allowed to accumulate, the plutonium inventory should remain less than 1 ekg. According to the definition of ekg and the information in Table IV, this reactor fueled with 93% enriched uranium would receive 1-2 mandays of Agency inspection per year with the Agency capable of devoting up to 50 mandays per year if it were considered necessary. If the total facility inventory exceeded 25kg U-235 or if the new fuel inventory exceeded 25 kg U-235, the Agency inspection effort would increase to 9-15 and 50 mandays, respectively.

At a U-235 enrichment of 20%, the 11.6kg of U-235 as new fuel represents about 0.4 ekg. In this case, the Agency would conduct one inspection per year of 1-2 mandays. This would apply as long as the inventory is kept below 5 ekg. In the third situation, 15kg U-235 in 6.5% enriched fuel is 0.6 ekg. Again, one inspection per year of 1 manday duration would be the expected Agency effort.

Currently, the IAEA has Agency-State Agreements to the effect that the actual routine inspection effort (ARIE) at thirty three research reactors operating with cores of uranium enriched to 20% or greater in the uranium-235 isotope is more than 500 man days. If the cores of these reactors were converted to less than 20% U-235 enrichment, the Agency ARIE could be significantly reduced.

## 4.0 Conclusions

The significance to the research reactor operator of the IAEA inspector effort in fulfilling its safeguards obligations is that the facility staff can be expected to devote an effort comparable to that of the Agency in the preparation for and during an IAEA inspection. Certainly, while the inspector is conducting the inspection, facility records must be made available to him along with explanations and discussions of the records by the facility staff. As he is verifying the presence of the fuel, checking seals or servicing surveillance devices, facility staff must accompany him. The Agency inspection effort as estimated in Table IV can be used as a lower limit to approximate the facility effort in support of the Agency's programme. It is credible to suggest the facility effort added to the State effort in support of Agency inspections could be 2-3 times the mandays shown in Table IV; e.g. in those instances where the State wishes to have its governmental representatives present during the inspection.

In summary, the enrichment and quantity of uranium at a research reactor determines the safeguards costs to the IAEA, the State and the facility operator. The difference between the safeguards costs for a research reactor fueled with HEU and one fueled with LEU can be as great as fifty.

# CHAPTER 15

## RESEARCH REACTOR SAFEGUARDS[4]

Contributed by IAEA

### 1.0 Introduction

A basic purpose of the Treaty on the Non-Proliferation of Nuclear Weapons, 1 July 1968 (NPT) is to prevent the spread of nuclear weapons. The NPT recognises the International Atomic Energy Agency as a major instrument in the implementation of the Treaty in this regard. Article III of the NPT states, in part, "Each non-nuclear-weapon State Party to the Treaty undertakes to accept safeguards, as set forth in an agreement to be negotiated and concluded with the International Atomic Energy Agency in accordance with the Statutes of the International Atomic Energy Agency and the Agency's safeguards system, for the exclusive purpose of verification of the fulfillment of its obligations assumed under this treaty with a view to preventing diversion of nuclear energy from peaceful uses to nuclear weapons or other nuclear explosive devices." The Agency's safeguards agreements and its application of safeguards are not aimed at nuclear weapons per se, but rather nuclear materials. In keeping with the intent of the NPT, the Agency's programme places major emphasis on the safeguarding of nuclear materials that can most readily be made into nuclear weapons. High enriched uranium is one such material and Agency resources can be conserved if this nuclear material is not used in applications where low enriched uranium would serve just as well.

In this section, the IAEA safeguards requirements for research reactors are briefly described and estimates are given for the Agency effort to safeguard research reactors fuelled with uranium of different enrichments. Based on Agency safeguards activities, costs are estimated for the State and facility operator to meet their IAEA safeguards obligations using four different facility fueling examples. It is shown that there can be a safeguards cost differential of one or more orders of magnitude between using high enriched and low enriched uranium.

### 2.0 International Safeguards System

### 2.1 Basic Safeguards Documents

The International Atomic Energy Agency applies safeguards according to Agency-State Safeguards Agreements based on the provisions of its Statute[1] and either INFCIRC/66/REV. 2,[2] or INFCIRC/153 (corrected)[3]. Although stated somewhat differently in each agreement, the objective of IAEA safeguards is to verify compliance with the undertakings of the Agreements. In Agreements based on INFCIRC/66/Rev. 2, Article III.A.5 of the Agency Statute is referenced

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1/ Statute, International Atomic Energy Agency, June 1, 1973.

2/ INFCIRC/66/Rev. 2. The Agency's Safeguards System (1965), As Provisionally Extended in 1966 and 1968, 16 September 1968.

3/ INFCIRC/153 (Corrected). The Structure and Content of Agreements Between the Agency and States Required in Connection with the Treaty on the Non-proliferation of Nuclear Weapons, June 1972.

4/ Physical protection safeguards are discussed in Ch. 16

as authorising the Agency to "establish and administer safeguards designed to ensure that special fissionable and other materials, services, equipment, facilities, and information made available by the Agency or at its request or under its supervision or control are not used in such a way as to further any military purpose." In Agreements according to INFCIRC/153 (corrected), Article 1, States undertake "to accept safeguards, in accordance with the terms of the Agreement, on all source or special fissionable material in all peaceful nuclear activities within its territory, under its jurisdiction or carried out under its control anywhere, for the exclusive purpose of verifying that such material is not diverted to nuclear weapons or other nuclear explosive devices." The objectives of safeguards as stated in Article 28 of INFCIRC/153 (corrected) are "the timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or of other nuclear explosive devices or for purposes unknown, and deterrence of such diversion by the risk of early detection."

## 2.2 Technical Objectives of Safeguards

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Both INFCIRC statements relate the purpose of Agency safeguards to the detection and deterrence of the use of nuclear material for nuclear weapons. In order for this purpose to be served, the Agency applies safeguards according to the significant nature of the material and the time estimated to convert the material to the components of a nuclear weapon. A significant quantity of nuclear material is defined as that approximate quantity, taking into account any conversion process involved, such that the possibility of manufacturing a nuclear explosive device cannot be excluded.[4] Table I gives the numerical values of significant quantities for all nuclear materials under Agency safeguards.

The Agency considers conversion time should correspond in order of magnitude to detection time, defined as the maximum time that may elapse between diversion and its detection by IAEA safeguards. The Agency's safeguards programme and its distribution of resources, which directly relate to the effort a State with which the Agency has a safeguards agreement devotes to its safeguards programme, are roughly measured in terms of the State's nuclear material and its safeguards significance.

Tables I and II illustrate the differences between low enriched uranium (LEU) and high enriched uranium (HEU) in terms of safeguards concerns, i.e. significant quantities and conversion time. The reason for the differences is that HEU (uranium where the U-235 content is equal to or greater than 20 percent) can be used directly to make metallic components of a nuclear weapon. If the HEU is in the form of a compound or alloy, it can be separated out by chemical processing: this is considered to be a simple, straightforward procedure and the estimated conversion time for HEU as a compound is 1-3 weeks, as the metal it is 7-10 days. (Low enriched uranium (uranium in which the U-235 content is less than 20 percent) cannot be made into the components of a nuclear weapon.) LEU must be further enriched, or used in a reactor producing thermal power to produce plutonium. Neither further enrichment nor the reactor production of plutonium are considered straightforward; hence the estimate of one year for the conversion time of low enriched uranium.

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4/ IAEA Safeguards Glossary, IAEA/SG/INF/1, Vienna, June 1980.

TABLE I

Significant Quantities

| | Material | Significant Quantity | Safeguards Apply to |
|---|---|---|---|
| Direct-use nuclear material | Pu* | 8 kg | Total element |
| | U-233 | 8 kg | Total isotope |
| | U[U-235>20%] | 25 kg | U-235 contained |
| | - Plus rules for mixtures where appropriate - | | |
| Indirect-use nuclear material | U[U-235<20%]** | 75 kg | U-235 contained |
| | Th | 20 kg | Total element |
| | - Plus rules for mixtures where appropriate - | | |

*For Pu containing less than 80% Pu-238.

**Including natural and depleted uranium.

Conversion times are shown in Table II.[4]

TABLE II

Estimated Material Conversion Times to Finished Pu or U Metal Components

| Beginning material form | Conversion time |
|---|---|
| Pu, HEU or U-233 Metal | Order of days (7-10) |
| $PuO_2$, Pu $(NO_3)_4$, or other pure Pu compounds; HEU or U-233 oxide or other pure compounds; MOX or other non-irradiated pure mixtures containing Pu, U[(U-233 + U-235)>20%]; Pu, HEU and/or U-233 in scrap or other miscellaneous impure compounds | Order of weeks (1-3)* |
| Pu, HEU or U-233 in irradiated fuel** | Order of months (1-3) |
| U containing <20% U-235 and U-233; Th | Order of one year |

*This range is not determined by any single factor but the pure Pu and U compounds will tend to be at the lower end of the range and the mixtures and scrap at the higher end.

**Criteria for establishing the irradiation to which this classification refers are under review.

The higher the enrichment of HEU, the more suitable the material is for use in nuclear weapons. The higher the enrichment of LEU, the easier it is to enrich that material to 20 percent U-235 or more. The Agency takes enrichment into account in its implemented safeguards program through use of the term "effective kilogram." Effective kilogram is defined in both INFCIRC/66/Rev. 2 and INFCIRC/153 (Corrected) as a quantity of uranium obtained by taking:

(a) For plutonium, its weight in kilograms;

(b) For uranium with an enrichment of 0.01 (1%) and above, its weight in kilograms multiplied by the square of its enrichment;

(c) For uranium with an enrichment below 0.01 (1%) and above 0.005 (0.5%), its weight in kilograms multiplied by 0.0001; and

(d) For depleted uranium with an enrichment of 0.005 (0.5%) or below, and for Thorium, its weight in kilograms multiplied by 0.00005."

2.3 Basic Agency Safeguards Measures for Uranium

The specific details of the safeguards implemented at any facility are based upon the Agreement between the State and the Agency. There are many similarities in the safeguards programmes for both HEU and LEU. For example, records must be kept of transfers to and from the facility and of the location of material within the facility in terms of quantity and type of material; the facility inventory must be periodically reported to the Agency; changes in design information and changes in nuclear material inventory must be reported to the Agency. A system of records and reports and a programme of material accounting complemented by containment and surveillance devices is the basis for State-Agency Agreements for both high enriched and low enriched uranium. Also for both classes of material, the Agency conducts inspections to independently verify that the quantity of nuclear material reported as on inventory at the facility by the facility operator is actually at the facility.[5/] The effort devoted by Agency inspectors to this verification activity is determined by the quantity of nuclear material at the facility as measured in effective kilograms. Table III shows the relationship between nuclear material enrichment, weight in kilograms and effective kilograms.

TABLE III

Effective Kilogram Quantities for Uranium of Varying Enrichments

| MASS (kg) of Uranium | EFFECTIVE KILOGRAMS (ekg) | | |
|---|---|---|---|
| | 10% Enrichment | 20% Enrichment | 95% Enrichment |
| 10 | 0.1 | 0.4 | 9 |
| 20 | 0.2 | 0.8 | 18 |
| 50 | 0.5 | 2 | 45 |
| 100 | 1 | 4 | 90 |
| 500 | 5 | 20 | 451 |
| 1000 | 10 | 40 | 902 |
| 5000 | 50 | 200 | 4512 |

5/ J Powers, "Safeguarding Research Reactors", STR-118, March 1983

## 3.0 Safeguards Implementation at Research Reactors

### 3.1 Categorisation of Research Reactors According to Inventory

Both INFCIRC/66/Rev. 2 and INFCIRC/153 (Corrected) relate inspection effort to the inventory of nuclear material as measured in effective kilograms at the facility. The State-Agency Agreement guidelines in these documents have been studied by Agency staff. Agency inspection effort has been estimated for four different situations. These are summarised in Table IV.

Table IV shows both maximum effort and actual effort. The maximum effort is the maximum routine inspection effort the Agency would be authorised to spend at the facility. The actual effort is the actual routine inspection effort the Agency would most probably spend at the facility and would depend on the Agency-State Agreement and the real conditions of the Agency safeguards implementation programme.

TABLE IV

Estimated Agency Inspection Effort for Research Reactors (Mandays/year)

| Type and Quantity of Fuel | Maximum Effort | Actual Effort |
|---|---|---|
| a. Enrichment < 20% U-235; or Inventory less than 5 ekg. | 2 - 3 | 0.5 - 1 |
| b. Enrichment > 20% U-235; Inventory < 25kg U-235. | 50 | 1 - 2 |
| c. Enrichment > 20% U-235; Inventory > 25kg U-235, but < 25kg in fresh fuel. | 50 | 9 - 15 |
| d. Enrichment > 20% U-235; Inventory > 25kg U-235 in fresh fuel. | 50 | 50 |

### 3.2 Agency Inspection Effort

For research reactors in either of the first two categories in Table IV, the inspection effort would consist of an annual physical inventory verification. Inspection activities during a physical inventory include the following:

- establishment of the updated book inventory based on the examination of shipping and receiving documents;
- examination of records for self-consistency and verification of the consistency between records and reports;
- verification of the fresh and irradiated fuel assemblies or elements based on item counting and identification;
- checking, removing and reapplying seals as appropriate.

For research reactors in the third category, an annual PIV would be taken as above. In addition, interim inspection would be carried out four to six times per year. The activities performed by the inspectors during these inspections would include the following:

- establishment of the updated book inventory based on the examination of shipping and receiving documents;
- verification of the fresh and spent fuel elements;
- verification of the integrity of applied seals;
- activities related to installed surveilance units.

Essentially, the Agency inspection effort for a research reactor with this type and quantity of fuel is estimated to be the same as for a light water reactor.

For research reactors having fuel as described in the last category, it would be necessary to inspect the facility on a frequency corresponding to the 1-3 week detection time. i.e. routine inspections about every 3 weeks. In addition, because of the safeguards significance of the HEU, the Agency may recommend physical inventory verifications up to four times per year. All these activities by the Agency inspectors are estimated to require 50 mandays of effort.

### 3.3 Safeguards at Generic Research Reactor Fuelled with Different Uranium Enrichments

Changes in reactor core parameters as a result of converting the core from high enriched to low-enriched uranium have already been reported.[6] The Agency safeguards programme as it applies to HEU and LEU fuelled reactors can be illustrated by considering the generic 10 MW research reactor described in that report. Calculations were made for the fuel loadings for the reactor using uranium enriched to 93%, 20% and 6.5% in the U-235 isotope. These fuel loadings along with estimated new fuel requirements are shown in Table V.

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6/ Research Reactor Core Conversion from the Use of Highly Enriched Uranium to the Use of Low Enriched Uranium Fuels. Guidebook IAEA-TECDOC-233, Vienna 1980.

TABLE V

Fuel Inventories for Generic 10 MW Reactor[5]

| Enrichment (% U-235) | New Fuel (kg U-235) | Core Fuel (kg U-235) | Spent Fuel (g Pu/Element) |
|---|---|---|---|
| 93 | 8 | 4.5 | 0.6 |
| 20 | 11.6 | 8 | 14 |
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The U-235 content of the core fuel is calculated on the basis that the fissile material has been depleted to where new fuel must be added. Burnup is fifty to sixty percent and therefore the core fuel is safeguarded according to the measures applied to spent fuel. (See Tables I and II). Plutonium content has also been calculated for spent fuel and unless the spent fuel is allowed to accumulate, the plutonium inventory should remain less than ~~1~~ ekg. According to the definition of ekg and the information in Table IV, this reactor fueled with 93% enriched uranium would receive 1-2 mandays of Agency inspection per year with the Agency capable of devoting up to 50 mandays per year if it were considered necessary. If the total facility inventory exceeded 25kg U-235 or if the new fuel inventory exceeded 25 kg U-235, the Agency inspection effort would increase to 9-15 and 50 mandays, respectively.

2.3 At a U-235 enrichment of 20%, the 11.6kg of U-235 as new fuel represents about ~~[illegible]~~ ekg. In this case, the Agency would conduct one inspection per year of 1-2 mandays. This would apply as long as the inventory is kept below 5 ekg. In the third situation, 15kg U-235 in 6.5% enriched fuel is [illegible] ekg. Again, one inspection per year of 1 manday duration would be the expected Agency effort.

Currently, the IAEA has Agency-State Agreements to the effect that the actual routine inspection effort (ARIE) at thirty three research reactors operating with cores of uranium enriched to 20% or greater in the uranium-235 isotope is more than 500 man days. If the cores of these reactors were converted to less than 20% U-235 enrichment, the Agency ARIE would be significantly reduced.

## 4.0 Conclusions

The significance to the research reactor operator of the IAEA inspector effort in fulfilling its safeguards obligations is that the facility staff can be expected to devote an effort comparable to that of the Agency in the preparation for and during an IAEA inspection. Certainly, while the inspector is conducting the inspection, facility records must be made available to him along with explanations and discussions of the records by the facility staff. As he is verifying the presence of the fuel, checking seals or servicing surveillance devices, facility staff must accompany him. The Agency inspection effort as estimated in Table IV can be used as a lower limit to approximate the facility effort in support of the Agency's programme. It is credible to suggest the facility effort added to the State effort in support of Agency inspections could be 2-3 times the mandays shown in Table IV; e.g. in those instances where the State wishes to have its governmental representatives present during the inspection.

In summary, the enrichment and quantity of uranium at a research reactor determines the safeguards costs to the IAEA, the State and the facility operator. The difference between the safeguards costs for a research reactor fueled with HEU and one fueled with LEU can be as great as fifty.

# CHAPTER 16

## PHYSICAL PROTECTION

Contributed by IAEA

### 1. Background

The physical protection of nuclear material and facilities has been attracting increasing attention in the last decade. There has been growing concern that the theft of plutonium, enriched uranium or uranium-233 would enable individuals or groups of individuals to construct a nuclear explosive device or to use these materials as radiological contaminants and that an act of sabotage against a nuclear facility could also create a radiological hazard to the public. In order to cope with these potential hazards through international efforts, the Agency was called upon to prepare a set of recommendations for the protection of nuclear material and facilities against willful hostile acts, e.g. theft or unauthorized removal of nuclear material or sabotage against nuclear facilities committed on sub-national levels. The Agency's work toward this end resulted in the publication in 1975 of the booklet entitled "The Physical Protection of Nuclear Material" which was revised and re-issued in 1977 under the same title (INFCIRC/225/Rev.1). The recommended measures of this document relate to the physical protection of nuclear material in use, transit and storage and have been used by an increasing number of Member States as guides in the preparation of their national regulations. These recommendations which are by themselves not mandatory upon States are also incorporated into international agreements binding upon States and/or the Agency, such as safeguards agreements (see, for instance, the Safeguards Agreement of 10 February 1977 between the Agency, Canada and Spain - INFCIRC/247, Second Amendment to the Agreement for Cooperation between the Agency and the United States of America which entered into force on 6 May 1980 - (INFCIRC/5/MOD.2). It should also be noted here that the Convention of the Physical Protection of Nuclear Material (INFCIRC/274/Rev.1) concluded on 26 October 1979 and opened for signature on 3 March 1980 provides for the levels of physical protection applied in international transport of nuclear material categorized in a form similar to that of INFCIRC/225/Rev.1.

### 2. Nature and Scope of the Agency Recommendations

INFCIRC/225/Rev.1 is a set of recommendations which may serve as guidelines for national regulatory authorities of Member States. It recommends that the State should promulgate and review regularly its comprehensive regulations for the physical protection of nuclear material whether in State or private possession. It further provides that the State should license activities only when they comply with these regulations.

The recommended measures of INFCIRC/225/Rev.1 are intended for all nuclear facilities and shipment: however, it is recognized that research type facilities outside the nuclear fuel cycle and their corresponding shipments may not be able to meet the recommendations, in which case the State's physical protection system may make specific exceptions on a case-by-case basis.

It should be noted that the measures recommended in INFCIRC/225/Rev.1 are based on the state of the art in physical protection and on the types of nuclear facilities existing in 1977. The document also recognizes that the design of a physical protection system for a specific facility may vary from those recommended when prevailing circumstances indicate a need for a different level of physical protection.

As for recommendations pertaining to nuclear power reactors as distinct from other nuclear facilities it provides that at reactors using natural or low-enriched uranium, the principal attractiveness relates only to possible sabotage, which could prove a radiological hazard to the public, whereas at reactors using plutonium or highly enriched uranium, the fuel material is attractive to a potential thief before it is irradiated in the reactors.

3. Categorization of Material

The physical protection measures recommended in INFCIRC/225/Rev.1 are based on the assumption that the more attractive a given nuclear material or a nuclear facility using such material is as a potential target for a hostile act which could lead to a nuclear diversion or radioactive dispersal, the more stringent and extensive the protective measures applied should be. INFCIRC/225/Rev.1 categorizes nuclear material in order to ensure an appropriate relationship between the nuclear material concerned and the protective measure applied. The categorization of nuclear material set out in INFCIRC/225/Rev.1 is reproduced below.

TABLE: CATEGORIZATION OF NUCLEAR MATERIAL[e]

| Material | Form | Category | | |
|---|---|---|---|---|
| | | I | II | III |
| 1. Plutonium[a,f] | Unirradiated[b] | 2 kg or more | Less than 2 kg but more than 500 g | 500 g or less[c] |
| 2. Uranium-235[d] | Unirradiated[b] | | | |
| | - uranium enriched to 20% $^{235}U$ or more | 5 kg or more | Less than 5 kg but more than 1 kg | 1 kg or less[c] |
| | - uranium enriched to 10% $^{235}U$ but less than 20% | - | 10 kg or more | Less than 10 kg[c] |
| | - uranium enriched above natural, but less than 10% $^{235}U$ | - | - | 10 kg or more |
| 3. Uranium-233 | Unirradiated[b] | 2 kg or more | Less than 2 kg but more than 500 g | 500 g or less[c] |

[a]All plutonium except that with isotopic concentration exceeding 80% in plutonium-238.

[b]Material not irradiated in a reactor or material irradiated in a reactor but with a radiation level equal to or less than 100 rads/hour at one meter unshielded.

[c]Less than a radiologically significant quantity should be exempted.

[d]Natural uranium, depleted uranium and thorium and quantities of uranium enriched to less than 10% not falling in Category III should be protected in accordance with prudent management practice.

[e]Irradiated fuel should be protected as Category I, II or III nuclear material depending on the category of the fresh fuel. However, fuel which by virtue of its original fissile material content is included as Category I or II before irradiation should only be reduced one Category level, while the radiation level from the fuel exceeds 100 rads/h at one meter unshielded.

[f]The State's competent authority should determine if there is a credible threat to disperse plutonium malevolently. The State should then apply physical protection requirements for Category I, II or III of nuclear material, as it deems appropriate and without regard to the plutonium quantity specified under each category herein, to the plutonium isotopes in those quantities and forms determined by the State to fall within the scope of the credible dispersal threat.

The categorization of material shown is based on the potential hazard of the material which itself depends on the type of material, isotopic composition, physical and chemical form and quantity of material. It covers all plutonium, except that with isotopic concentration exceeding 80% in $Pu^{238}$, uranium$^{-235}$ enriched to 10% or more and uranium$^{-233}$ (though outside of the scope of the present study), irrespective of their quantities; but it is foot-noted that less than a radiologically significant quantity of these materials should be exempted. Also covered is uranium$^{-235}$ enriched above natural, but less than 10% in the quantity of 10 kg or more. It suggests in a footnote that natural uranium, depleted uranium and thorium, and less than 10 kg of uranium$^{-235}$ enriched to less than 10% not falling within the scope of this categorization should be protected in accordance with prudent management practice. This categorization applies to irradiated fuel as well.

4. Protection Measures

Against the scope and the structure of INFCIRC/225/Rev.1, as summarized in the preceding paragraphs, should be weighed the relevance and applicability of the recommended physical protection measures to the present subject of licensing issues related to research reactor core conversion.

Although it is not the purpose here to present an array of recommended measures in detail, it may still be useful to outline the basic contents of the recommendations as they relate to the requirements for nuclear material in use and storage:

i) A higher level of physical protection is required for Category I material. The nuclear material in this category is to be used and stored within special inner areas encircled by the protected area, which is characterized by separate physical barriers, differentiated restrictions and controls on personnel access; limiting the number of personnel having access to the minimum necessity, constant personnel surveillance, search of all persons and packages entering and leaving the inner area, badging of persons, the checking and custody of keys or key-cards, 24 hour guarding service, special design of storage area within an inner area, alarms and TV monitors for storage area, internal and external patrol, use of independent duplicated transmission system, etc.

ii) A medium level of physical protection is required for Category II material. The nuclear material in this category should be used and stored within a protected area for which a somewhat less rigid protection system needs to be provided; such a system includes access control by limiting the number of persons admitted, issuing of badges and provision of visitor-escourts, occasional search of persons and packages entering the protected area, checking and custody of keys, continuous personnel surveillance, etc.

iii) A lower level of physical protection is required for Category III material. The nuclear material in this category should be used and stored within an area to which access is controlled but no special protected area is necessary: the requirements for this category are limited to such matters as training of personnel, the operator's general responsibility for movements of nuclear material, detection of unauthorized intrusion and appropriate actions by guards or off-site emergency teams, preparation of emergency plan of action and security surveys. All of these measures are to be applied equally or more rigidly to Categories I and II above.