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Good Practices for Qualification of High Density Low Enriched Uranium Research Reactor Fuels

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GOOD PRACTICES FOR QUALIFICATION OF HIGH DENSITY LOW ENRICHED URANIUM RESEARCH REACTOR FUELS

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FOREWORD

Several national and international efforts are under way to develop, qualify, and license low enriched uranium (LEU) training, research, test and isotope production reactor fuel. This development work, based on the use of g phase uranium–molybdenum alloys, seeks to provide the fuels needed to extend the use of LEU to those reactors requiring higher densities than currently available in the uranium–silicide dispersions being used in research and test reactors designed outside the Russian Federation and in the uranium–oxide dispersions being used in Russian designed training, research, test and isotope production reactors. In addition, uranium–molybdenum alloys are expected to be more easily reprocessed than uranium–silicide dispersions.

Consistent with increasingly critical non-proliferation concerns brought about by the use of high enriched uranium (HEU) in training, research, test and isotope production reactor fuels, conversion of research reactors from HEU to LEU is acquiring strong momentum worldwide. This implies that an important number of commercial operations involving LEU training, research, test and isotope production reactor fuels are foreseeable in the near future. For upcoming LEU fuel supply arrangements, clear knowledge and appreciation of the requirements that procured fuel should meet are crucial. One of the main prerequisites is that fuels supplied for research reactor core conversion should be qualified. Therefore, a common understanding of what 'qualified fuel' means is mandatory.

Three IAEA-TECDOCs on research reactor core conversion and research and test reactor related issues have been published by the IAEA. IAEA-TECDOC-233 addresses feasibility studies and fuel development potential for light water moderated research reactors, and IAEA-TECDOC-324 addresses these topics for heavy water moderated research reactors. IAEA-TECDOC-643, in five volumes, addresses the effects of changes within the safety related parameters of mixed cores and the converted core. IAEA-TECDOC-643 presents information and test data on some reduced enrichment fuels, detailed data on fuel materials, irradiation tests, and post-irradiation examinations (PIE), as well as providing examples of fuel specifications and inspection procedures.

The IAEA recognizes that, even though a great deal of information on training, research, test, and isotope production reactor fuel qualification is available in open literature, so far no comprehensive document addressing the rationale underlying qualification of these fuels has been published.

This good practices publication has been prepared to provide points of reference for the type, quality, and completeness of information to be generated in order to ensure the acceptable performance of high density LEU fuels to be used in existing and new training, research, test, and isotope production reactors. The IAEA anticipates that the information presented here will be of value to fuel developers, reactor operators planning to use a new fuel, and to regulatory bodies faced with deciding whether a specific reactor can be licensed to use a new fuel.

The IAEA wishes to thank all contributors to this publication. Special thanks are due to J.L. Snelgrove who compiled, edited and revised the text. The IAEA officer responsible for this publication was P. Adelfang of the Division of Nuclear Fuel Cycle and Waste Technology.

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CONTENTS

1. INTRODUCTION

Worldwide non-proliferation concerns about the use of high enriched uranium (HEU) in training, research, test, and isotope production reactors[1] have resulted in a number of national and international fuel development and qualification programmes aimed at allowing these reactors to fully achieve their missions while using low enriched uranium (LEU) fuels[2] and isotope production targets. During the 1980s, uranium densities of research reactor fuels at that time, dispersions of UAl_x, U_3O_8, and UO_2 in an aluminium matrix, were significantly increased, to 2.3, 3.2, and 2.5 Mg U/m^3, respectively [1, 2]. In addition, a fuel capable of much higher uranium densities, U_3Si_2 dispersed in aluminium, was developed and qualified at a density of 4.8 Mg U/m^3 [3]. In 1996, the need for a fuel more easily reprocessed than the U_3Si_2 dispersions and for even higher uranium densities to enable the operation of a number of high performance reactors using LEU resulted in a renewed and continuing international effort to develop and qualify fuels based on gamma phase U–Mo alloys which could be used in aluminium dispersions to provide 8–9 Mg U/m^3 and as solid (monolithic) fuels to provide up to 16 Mg U/m^3 [4, 5].

In late 2003 and early 2004, analysis of several U–Mo dispersion fuel experiments which had either failed by breakaway swelling or showed signs of incipient failure, indicated that the interaction product formed between the U–Mo fuel and the aluminium matrix was apparently amorphous and, hence, unstable under high fission rates, temperatures, and burnup conditions [6]. Fortunately, the unreacted U–Mo fuel showed very stable behaviour. Since that time, major international efforts to solve the interaction product swelling problem and to develop the monolithic U–Mo fuel alternative, which does not have an aluminium matrix, have been productive. Irradiations in both France and the US have demonstrated that adding at least 2 wt% silicon to the aluminium matrix both decreases the rate of formation of the interaction product and stabilizes it with respect to the phenomena responsible for breakaway swelling, to the limit of fission density possible with LEU [7, 8]. Other remedies, including an oxide coating on the U–Mo particles, are also being pursued [7, 9]. In addition, considerable progress has been made in the development of monolithic fuel plates, both in their manufacturing and in controlling the same interaction product phenomenon at the interface of the U–Mo fuel and the cladding [8]. As fission rates and fission densities have been increased in more recent irradiation tests, using uranium enriched higher than 20% to compensate for limited neutron flux available in test reactors, U–Mo fuel itself has been shown to have a stability limit [10]. This unstable growth of fission gas bubbles at the fuel particle periphery begins at a fission density comfortably beyond that physically possible with LEU, however. Work is continuing on both the dispersion and monolithic U–Mo fuels.[3]

Fuel development and qualification programmes are conducted somewhat differently in each country and generate somewhat different information, depending on country practices and regulations. Most programme elements have a common thread, however, and international collaborations usually concentrate on those common elements.

One result of international collaboration in fuel development and qualification has been the realization that participants from different countries do not always understand terminology the same way. For example, the term ‘qualification’ is currently interpreted differently in different countries. Therefore, one goal of this document is to suggest a set of commonly accepted definitions for the terminology of fuel development and qualification activities.

The principal purpose of this document is to provide a set of good practices to guide fuel developers, staff members of reactor operating organizations responsible for preparing safety analyses to justify the use of new

[1] In this publication the term ‘research reactor’ will be assumed to refer to training, research, test, and isotope production reactors unless otherwise explicitly stated.

[2] In this publication the term ‘fuel’ will be assumed to refer only to LEU fuel unless otherwise stated.

[3] Extensive reports on international fuel development activities are presented annually at the Reduced Enrichment for Research and Test Reactors (RERTR) international meetings in autumn and at the Research Reactor Fuel Management (RRFM) meetings in spring. Most papers presented since 1996 at RERTR meetings are available at <http://www.rertr.anl.gov/index.html#meetings> and the transactions from RRFM meetings are available at <http://www.euronuclear.org/events/rrfm.htm>.

fuels in order to convert from HEU fuel to LEU fuel, and members of regulatory bodies responsible for reviewing applications for research reactor license amendments (or for new licenses). This is especially important for the development and qualification of a new fuel form, such as U–Mo monolithic fuel.

It must be clearly understood that this is not a safety document. Nevertheless, developing fuel, fuel elements, and fuel assemblies that will operate safely under specified operating conditions is the ultimate goal of any fuel developer. Therefore, it is expected that this document will contribute to the safety of any reactor using newly developed fuel.

2. SCOPE

This good practices publication addresses basic definitions, approaches, and processes relevant to the qualification of research reactor fuel, and defines essential information required for the licensing and use of fuels in research reactors. It concentrates on development and qualification of high density fuels of the type used in most research and test reactors, i.e., fuel consisting of a fuel meat contained within a metallic cladding. Any other type of fuel, e.g. a homogeneous solution fuel, is outside the scope of this publication.

Essential definitions are presented in Section 3. In Section 4, a description of the type and extent of information needed to support a fuel qualification report is presented, while many additional pointers are given in Appendix I. An overview of the qualification process is presented in Section 5. The similar and somewhat flexible qualification processes employed for research reactor fuels in Argentina, Canada, France, the Republic of Korea, and the United States of America on one hand and the very structured and codified process used in the Russian Federation on the other hand are described in Section 6. Examples and case histories for both types of approaches are presented in Appendix II. A list of the acronyms used in this document is presented in Appendix III.

3. DEFINITIONS

During the history of research reactor fuel development and application, many special terms have been developed to describe various fuel properties and processes. So that there will be no misunderstanding of the meaning of these terms as they are used in this report, they are defined below. In the definitions, terms that are capitalized are also clarified in this section. The reader is urged to study these definitions carefully before reading Sections 4–6 and to refer back to this section as needed.

areal uranium density. The quantity of uranium per unit area of cladding surface contained within the volume of fuel meat between the cladding of a plate type fuel element; note that the fuel volume is associated with two unit areas of cladding, one on each side of the fuel meat (see also uranium density).

bend test. A test carried out on a plate type fuel element consisting of bending a strip cut from the excess material at the end of the manufactured fuel plate and bending it back, and/or around a mandrel to test the bonding of the cladding material to the frame material.

blister test (during fuel plate manufacturing). A test carried out on a plate type fuel element at the end of the hot bonding process in which the fuel plate is heated to a given temperature (approximately equal to the hot bonding temperature) for a given time to verify that residual gases in the meat do not cause a local deformation (blister) of the fuel cladding; a blister free fuel element surface indicates that the fuel cladding is sufficiently bonded to the fuel meat.

blister test (on irradiated fuel plates). A test carried out on an irradiated plate type fuel element in which the fuel plate is sequentially heated to incrementally higher temperatures to determine the temperature at which gases (principally fission gases) in the fuel meat diffuse to the fuel meat–fuel cladding interface in sufficient quantity to locally deform the fuel cladding into a blister on the fuel cladding surface. The **blister threshold temperature (or blister temperature)** is the temperature at which one or more blisters first appear on the surface of the fuel element.

breakaway swelling. A type of unstable swelling that can occur in fuel meat due to coalescence of fission gas bubbles within the fuel meat to the extent that the surface tension of the bubbles is reduced enough to allow rapid expansion of the bubbles and a consequent loss of fuel meat integrity during a relatively short increment of irradiation.

burnup. Fraction (usually expressed in percent) of U-235 atoms initially present in a given volume of fuel meat that are depleted by fission and neutron capture.

continuous phase (in a dispersion fuel meat). Constituent of the fuel meat that provides an uninterrupted path for heat transfer to the cladding.

cooling (or coolant) channel. Hydraulic space between neighbouring fuel elements in the fuel assembly through which coolant flows. For a plate type assembly, it is the space between two neighbouring plates; for a pin-type assembly, it is the space in the assembly mesh between neighbouring fuel pins.

customer. An organization procuring fuel assemblies from a fuel manufacturer for use in a research reactor, either for testing or for routine operation.

dispersion fuel element. A fuel element containing dispersion fuel meat.

dispersion fuel meat. Fuel meat consisting of a mixture of fuel particles and non-fissionable matrix particles that have been blended, compacted, and either rolled or extruded into a dense, cohesive solid.

dummy fuel element (assembly). A fuel element (assembly) manufactured to required specifications, except for those applying to the fuel meat, and used to test the fuel assembly manufacturing process or test some hydraulic or mechanical property of the fuel element (assembly). A dummy fuel element can be manufactured from the fuel cladding material alone or using a surrogate fuel material (including natural or depleted uranium) in the fuel meat, depending on its intended use.

fission density. The number of fissions that have occurred per unit volume of fuel meat (fission density in the fuel meat) or of fuel containing particles (fission density in the fuel particle).

fuel assembly (FA). Any device made of fuel elements configured as a single unit and able to be irradiated.

fuel cladding. Heat conducting material that surrounds (or contains) the fuel meat of a fuel element, preventing the escape of fission products to the environment and providing structural integrity to the fuel element.

fuel developer. An organization carrying out fuel development.

fuel development. All activities, involving materials science and nuclear engineering, focused on designing and manufacturing a new type of nuclear fuel and/or fuel element and testing its behaviour under irradiation.

fuel element (FE). Any device containing fissionable material which separates this material from the reactor coolant, and which is able to be irradiated (e.g., fuel plate, fuel tube, fuel pin, or fuel rod).

fuel element swelling. The fractional change (usually expressed in percent) of the fuel element volume owing to fuel meat swelling.

fuel foil. A thin strip of metallic fuel material used as the fuel meat in a monolithic fuel element.

fuel licensing. The process through which a regulatory body approves (issues a license for) the use in a specific research reactor of a particular fuel type in a particular geometric configuration under a specific set of irradiation conditions.

fuel manufacturer. A company or organization that manufactures fuel elements (FEs) and fuel assemblies (FAs) under industrial conditions in sufficient quantity to supply the FAs needed for fuelling at least one research reactor.

fuel material. A fissionable uranium compound or alloy used to form, or used as a constituent of, the fuel meat in a fuel element. 'Fuel material' is often shortened to 'fuel' when the context is clear, e.g., when speaking of the fuel–matrix interface.

fuel meat (or fuel core). Fuel material, or a combination of fuel materials with a non-fissionable heat conducting (matrix) material, confined inside a fuel element. The term 'fuel meat' is recommended in order to avoid confusion with the use of the term 'core', meaning 'research reactor core.'[4]

fuel meat swelling. The fractional change (usually expressed in percent) of the unirradiated fuel meat volume during irradiation owing to a combination of densification of the fuel meat resulting from sintering, reduction in as–manufactured porosity resulting from creep and/or swelling of matrix material and fuel particles, swelling of fuel particles resulting from the build up of fission products in the fuel particles, and nucleation and subsequent swelling of pores at the interface of the interaction product with the matrix material or the fuel cladding.

fuel particle. A small piece (maximum dimension <300 μm) of fuel material produced by mechanical action on a larger piece of the material (e.g., grinding) or by atomization from a melt.

fuel powder. A collection of many fuel particles with a specific particle size distribution.

fuel qualification. A process carried out by: (1) a fuel developer to provide sufficient information about a new fuel type or about a new use for an existing fuel type for a regulatory body to license that fuel type for use under a set of bounding geometric configurations and irradiation conditions and (2) a fuel manufacturer to demonstrate to regulatory bodies and customers that fuel elements and fuel assemblies of a particular fuel type can be reliably and consistently manufactured to required specifications.

fuel type. A specific combination of fuel material, fuel meat type (e.g., dispersion or monolithic), and fuel element–fuel assembly design.

good practice (also called best practice). An efficient and effective means to consistently achieve a desired outcome or outcomes within a specific organization. Recognition of a good or best practice within one organization does not imply that other practices employed to achieve the same or similar outcomes are of lesser quality.

in-pile test. A test on a fuel specimen performed in a nuclear reactor.

[4] The use of the term 'meat' resulted from an analogy of a fuel plate, in which the fuel bearing material is contained between two layers of cladding, to a sandwich containing a piece of meat between two slices of bread.

interaction product. The material formed by the interaction of the fuel material and matrix material and/or fuel cladding as a result of thermal processes during manufacturing (often called a reaction product) and irradiation induced and thermal processes during irradiation. The interaction product acts as a fuel material in the fuel meat.

lead test assembly (LTA). The first fuel assembly, containing a fuel meat that has previously been qualified with respect to irradiation performance, to be irradiated in a specific reactor for the purpose of qualifying a fuel manufacturer and/or demonstrating the adequacy of a fuel element–assembly design.

linear uranium density. The quantity of uranium per unit length of fuel meat in a rod or pin type fuel element (see also uranium density).

matrix material. A non-uranium bearing material in the fuel meat of a dispersion fuel meant to allow the fuel particles to 'flow' during manufacturing, to bind the fuel particles together in the as-manufactured fuel meat, and to improve the thermal conductivity of the fuel meat.

miniature fuel element (mini-element). A scaled down version of a fuel element, in which the dimension related to potential manufacturing difficulties and/or irradiation effects is maintained at normal scale.

monolithic fuel meat. A fuel meat consisting of a solid piece of fuel material.

out of pile test. A test on an unirradiated fuel specimen that does not involve irradiation in a nuclear reactor.

pillow (pillowing). A 'pillow shaped' deformation of the fuel cladding of a fuel element caused by the loss of the cladding–fuel meat bond over a substantial portion of the fuel meat area and the pressure of fission gas in the non-bonded region.

prototype fuel assembly. A fuel assembly designed for eventual use in a specific research reactor which contains an unqualified fuel and which is intended for irradiation testing performed in that research reactor or in a designated test reactor as part of the qualification process.

qualified fuel. A licensed fuel type that a regulatory body uses within specified limits of uranium density and operating conditions in one or more research reactors and that a customer agrees can be manufactured reliably and consistently to required specifications by at least one manufacturer.

reactor core designer. An organization that designs fuel assemblies using existing fuel assembly designs and designs the reactor core using these fuel assemblies.

scoping irradiation test. In the context of this document an irradiation test, generally performed using miniature fuel elements (mini-elements), to separate fuel materials or fuel meats exhibiting poor irradiation behaviour from those that are candidates for further development.

surrogate fuel (or fuel material). A non-uranium bearing material having a similar density and similar mechanical properties to a fuel material under development.

TRIGA®. A research reactor developed and marketed by General Atomics based on the use of a hydrided uranium–zirconium alloy ($UZrH_x$). TRIGA is an acronym for Test, Research, Isotope, General Atomics.

uranium density. The mass of total uranium contained per unit volume within the fuel meat, usually expressed in g/cm^3 (Mg/m^3). Areal (for plate) or linear (for pin) uranium density can be deduced by dividing the thickness by two (for plate-type fuel, which has two cladding surfaces) or by the transverse cross-section area (for pin-type fuel) of the fuel meat (see also areal uranium density and linear uranium density).

4. INFORMATION OBTAINED BY THE FUEL DEVELOPER FOR FUEL QUALIFICATION

4.1. PRELIMINARY REMARKS

The fuel development and qualification process will be discussed in Sections 5 and 6. However, before a process can be developed, the desired product needs to be understood. The outcome of the qualification process is the information needed to convince regulatory bodies and reactor operating organizations that the new fuel can be safely used in a specific research reactor. This section discusses the information that should be obtained by a fuel developer in order to qualify a new fuel type. Some of the items of information discussed are needed only to calculate other items and will not, by themselves, be of direct interest to a reactor designer, reactor operating organization, or regulatory body. However, all of the items discussed should be documented and detailed in a fuel qualification report.

Experience indicates that a fuel developer should try to obtain as broad a range of data as possible at each step of development, within constraints of scheduling and funding, in order to maximize applications for which the fuel can ultimately be qualified.

The information that should be provided by the fuel developer for fuel qualification falls naturally into five groups: (1) basic unirradiated fuel properties; (2) fuel meat and FE as-manufactured properties; (3) fuel meat and FE irradiation properties; and (4) FA properties. The various items of required information are discussed in the remainder of this section. More details on some of the topics can be found in Appendix I.

4.2. BASIC PROPERTIES OF UNIRRADIATED FUEL

Knowledge of basic fuel properties is needed to be able to interpret fuel behaviour during manufacturing and under irradiation, as well as to design FEs and assemblies for specific applications. Consequently, the characteristics must be known during the development stage so that they can be properly specified when ordering FAs for qualification and for routine use in research reactors.

4.2.1. Fuel material chemical and phase compositions

The chemical composition of a fuel material affects its phase composition and mechanical properties. Impurities also affect the mechanical properties of a fuel and, therefore, may have detrimental affects on FE fabricability. Knowledge of a fuel material's phase composition is particularly important when designing manufacturing processes and interpreting irradiation behaviour (see Appendix I for examples).

4.2.2. Fuel material heat capacity

The heat capacity of a fuel material is needed to calculate the heat capacity of the fuel meat. The heat capacity of the fuel meat is needed to determine the thermal conductivity of a fuel using the laser flash method. The heat capacity of a fuel material can be measured directly, calculated from measured values of the fuel meat heat capacity, or estimated, for example, by using the Neumann–Kopp rule, or other more sophisticated calculation methods.

4.2.3. Fuel material thermal conductivity

The thermal conductivity of a fuel material is used to calculate the thermal conductivity of high density dispersion fuel meat. In monolithic fuels, the fuel material is the fuel meat. In some cases, the thermal conductivity can be found in literature; if not, it should be measured.

4.2.4. Fuel powder properties (for dispersion fuel)

The fuel powder method of production, usually comminution or atomization, depending on the characteristics of the fuel compound or alloy, can influence several important parameters of a fuel powder, such as:

— Particle shape and particle size distribution (depending on production method) — affects blending and rolling or extrusion, fuel meat porosity, and amount and rate of fuel–matrix interaction formed during manufacturing and irradiation;
— Density (should be measured using an unfractured solid ingot, or piece thereof) — needed to determine fuel meat porosity;
— Mechanical properties (especially toughness) — affects comminution, blending, and rolling or extrusion;
— Grain size and structure (depending on production method) — affects fuel particle properties, such as thermal or in-pile fuel–matrix interaction rates and irradiation swelling.

4.2.5. Fuel foil properties (for monolithic fuel)

The method of production of a fuel foil affects uniformity of foil thickness and the distribution of grain sizes in a foil. From a fuel irradiation performance perspective, the grain size distribution is the more important characteristic, with uniformly sized grains several times smaller than the foil thickness being preferred. The thickness uniformity, however, translates directly into the uniformity of the local areal power density during irradiation.

4.3. FUEL MEAT AND FUEL ELEMENT AS-MANUFACTURED PROPERTIES

The fuel meat properties of the as-manufactured FE and the FE properties themselves are needed for determining the fuel meat and element irradiation properties and for performing FE design and safety analysis. The same properties are needed for small FE samples to be irradiated during the initial phase of fuel development, discussed in Section 5.1 of this document. These properties also are used in the detailed design process, and since all of these properties change as the fuel meat is irradiated, the as-manufactured properties provide the starting point for estimating the evolution of properties during irradiation. They include the following:

4.3.1. Fuel meat volume and constituent volume fractions

Because the items discussed in this section are needed for both the as-fabricated and the irradiated fuel meat, and because the considerations are the same or similar, both conditions will be considered here.

The volumes of fuel meat in the cladding before and after irradiation are needed to calculate fuel meat swelling. The as-fabricated fuel material volume is used, along with the volume of fuel material remaining after irradiation, to calculate the amount the fuel material has swelled during irradiation.

Pores form in dispersion fuel meat during the manufacturing process and during irradiation of both dispersion and monolithic fuels. This change in porosity of fuel meat during irradiation is an important component of fuel meat swelling. Although the term ‘void’ is sometimes used for porosity, the pores are never empty. There will always be some residual gases from manufacturing in the pores of as-fabricated fuel meat and fission gases in the pores of irradiated fuel meat.

The volume fractions of the meat constituents of a dispersion fuel (fuel, matrix, interaction product if any, and porosity), usually stated as percentages, are used to calculate fuel meat properties of that dispersion fuel, such as density and thermal conductivity.

Measurement methods and additional considerations for these terms are discussed in Appendix I.

4.3.2. Fuel meat (element) heat capacity

The fuel meat (element) heat capacity will be needed to evaluate stored energy and effects of reactor power transients during safety analyses. The fuel meat heat capacity can be measured during reaction enthalpy measurements or calculated using the heat capacities and volume fractions of the components of the fuel meat. The heat capacity of the cladding can be obtained from literature.

4.3.3. Fuel meat and cladding thermal conductivities

Thermal conductivities of the fuel meat and cladding are used when determining fuel meat temperature, which is an important parameter related to the irradiation performance of fuel meat. The thermal conductivity of a dispersion fuel meat is a function of the volume fractions of matrix, fuel, and porosity. The thermal conductivity of a fuel meat changes, sometimes dramatically, as fuel is irradiated. Determination of the thermal conductivity of as-manufactured fuel meat and the behaviour of the thermal conductivity of dispersion fuel meat are discussed in Appendix I.

4.3.4. Fuel meat (element) thermal expansion coefficient

Knowledge of the thermal expansion coefficients of the fuel meat and cladding components of the FE, especially for plate type elements, is important in evaluating the possibility that buckling stresses might cause a separation of the meat and cladding during irradiation, particularly at reactor startup or shutdown. For monolithic fuel, thermal expansion coefficients might also be useful during the manufacturing development process. The thermal expansion coefficient of an FE is needed during FA design so that stresses owing to differential thermal expansion can be evaluated and accommodated.

4.3.5. Exothermic energy release upon heating

In most cases, the reaction between fuel and aluminium, which is the main constituent of most matrix materials in dispersion fuel meat and of most claddings used in research reactor FEs, releases heat, i.e., is exothermic. The reaction rate is relatively small when the reacting materials are solid, but should portions of the matrix or cladding become molten, the reaction rate increases rapidly. The quantity and rate of heat generation from a exothermic fuel–matrix–cladding interaction should be measured for a dispersion fuel because exothermic reaction is one source of heat to be considered during the analysis of an accident resulting in FE temperatures approaching the cladding solidus temperature.[5] Such measurements are usually performed using differential thermal analysis or differential calorimetry techniques.

4.3.6. Fuel element mechanical properties

Fuel element mechanical properties, such as tensile and compressive strengths, stiffness, etc., may be needed to design the FA, especially for high performance research reactors. Such properties should be measured, although calculations might be useful to interpolate between measured points or, in some cases, to expand the range of data. Suitably designed hydraulic and mechanical tests are prudent, especially in the case of FEs designed for high performance research reactors (see also Section 4.5.1). Irradiation tends to increase the strength of FEs, but, of course, FAs must have sufficient strength to withstand beginning-of-life conditions.

[5] In most, if not all cases, exothermic reaction will contribute a negligibly small part to accident energy; however, a complete accident analysis will take it into account.

4.4. FUEL MEAT AND FUEL ELEMENT IRRADIATION PROPERTIES

As will be discussed further in Sections 5 and 6, miniature FEs are used in many irradiation studies to gain a basic understanding of fuel and fuel meat irradiation properties. In such cases, full size FEs must eventually be irradiated to demonstrate that there are no scale-up issues. In this section, most references to FEs are intended to also apply to miniature FEs; it should be obvious when this is not the case. Not only is this information needed to convince the reactor operating organization and regulatory body that the fuel will operate in a stable and predictable manner, but it is needed by the organization(s) responsible for FE, FA, and reactor core design as well as for reactor core safety analysis, both in the case of conversion of an existing reactor or design of a new reactor.

4.4.1. Fission density distribution

To properly interpret the irradiation behaviour of FEs and their contained fuel meats, one must know the fission density associated with the particular sample under study.

- Fission density in fuel meat: Mapping of the fission density profile of the fuel meat beneath the cladding of an irradiated FE provides the relative distribution of fission events, which is directly correlated to the time integrated power profile. This information is used to verify the accuracy of calculated power profiles for an irradiation experiment. The absolute fission density at, preferably, several positions along the relative profile should be measured or calculated to normalize data and to produce an absolute distribution of fission density. The peak and average fission densities reached during irradiation tests are sometimes used to set fission density limits in the technical specifications of research reactors;
- Fission density in fuel particles (absolute): This is the most useful fission density value in studying the irradiation behaviour of a fuel material, since fissions occur in the fuel material and cause the various phenomena that lead to fuel swelling. The reference fuel volume is the as-manufactured (i.e. unirradiated) fuel volume, although this might not be appropriate when substantial fuel–matrix interaction has occurred in a dispersion fuel meat. For a monolithic fuel with relatively thin fuel meat, this quantity is identical to the fission density in the fuel meat. For a dispersion fuel, one must divide the fission density in the fuel meat by the volume fraction of fuel in the as-manufactured fuel;
- ^{235}U burnup: Members of the reactor operations community are much more used to speaking in terms of ^{235}U burnup than fission density.[6] These two quantities are related through the ^{235}U capture to fission ratio and the fraction of total fissions that occurred in the ^{235}U.

Appendix I contains additional discussion and information on measurement techniques.

4.4.2. Fuel element swelling

Fuel element swelling and FE mechanical integrity, discussed in later sections, are two of the most important properties of irradiated FEs. Fuel element swelling results in a decrease in the volume of coolant channels in the FA, leading to decreased coolant flow, increased FE surface temperatures, and increased fuel meat temperatures. The amount of FE swelling that can be tolerated must be determined during the thermal–hydraulic design of the FA.

It is the responsibility of the fuel developer to analyze the swelling behaviour of different types of fuel material being qualified in order to demonstrate that FE swelling is stable and predictable, and that it has adequate margins beyond the design conditions of fission density, fission rate, and temperature. The fuel developer must also provide FA and reactor core designers with the limiting values, if any, of these parameters.

Since cladding swelling under irradiation is negligible, FE swelling effectively results only from fuel meat swelling. For dispersion fuels, the geometric configuration of the FE determines how fuel meat swelling

[6] This may have resulted because ^{235}U is directly related to the cost of fresh uranium purchased for fuel fabrication and because, at least in the past, credit was given for residual ^{235}U in reprocessed spent fuel.

translates into FE swelling. The simplest case is for fuel plates and fuel tubes, in which the fuel meat swells preferentially in the meat–thickness direction because the fuel meat is much more constrained in the other directions. (The cladding of a fuel plate is much thinner than the side rail or end portions of the plate; the stresses induced by swelling particles force the matrix to flow in the direction of least constraint, resulting in an increase in meat thickness). Not enough data are available yet to determine how monolithic fuel meat and FEs will swell.

Two modes of swelling can be distinguished:

— Linear, or stable swelling: The swelling increases at a linear rate as a function of fission density;
— Non-linear, or breakaway, swelling: The swelling rate increases continuously and rapidly as a function of fission density. Such swelling results from the formation of large pores in the fuel meat and eventually will result in the formation of bulges on the surface of plate or tube type FEs. These bulges are commonly referred to as pillows, because of their shape, and the phenomenon is referred to as pillowing. This type of swelling must be avoided.

4.4.2.1. Fuel meat swelling

As mentioned above, FE swelling results directly from fuel meat swelling, and fuel meat swelling results from different volume change components owing to complex irradiation induced phenomena, mainly identified as: (1) fuel densification resulting from irradiation induced sintering of the meat; (2) fuel swelling or densification from irradiation induced fuel matrix or fuel cladding interaction; (3) fission product swelling of the original fuel particles and any interaction product, and; (4) porosity generated by the nucleation, growth, and coalescence of fission gas pores, usually at the interaction product–matrix and product–cladding interfaces. These phenomena are discussed in Appendix I. The amount of fuel meat swelling to be expected in a particular FE configuration is of direct use during designing of FA thermal–hydraulics.

4.4.2.2. Fuel meat and fuel particle microstructures

While fuel swelling measurements provide quantitative data from which plots of fuel meat and element swelling as a function of fission density can be produced, real understanding of the nature of the swelling, as well as production of plots that accurately characterize the swelling, depend upon information obtained from micrographic studies of the fuel meat and its constituents, both optical and electron beam. Similarly, study of any fuel–matrix interaction depends heavily on information obtained from microscopy and microchemical analysis. Although the overall macrostructure of the fuel meat must be examined, the data of greatest interest is obtained from examination of the microstructures in the highest burnup region of the fuel meat. The types of examinations normally performed are discussed in Appendix I.

4.4.3. Fuel element mechanical integrity

To be a candidate for qualification, an FE must be demonstrated to be able to withstand stresses induced during irradiation by fuel meat swelling, differential thermal expansion of the fuel meat and cladding and by hydraulic forces. The fuel meat–cladding bond strength is a very significant property of FEs, particularly plate type or polyhedral tube type elements, because of potentially serious consequences should the cladding separate from the meat during irradiation. Scale-up factors can be significant, so adequate bond strength must be demonstrated by successful irradiation of full size FEs. Since significant differential thermal expansion stresses can occur at the fuel meat–cladding interface during reactor startup and shutdown, FE irradiation tests should continue for several reactor cycles.

4.4.4. Blister threshold temperature

The blister threshold temperature, or blister temperature, traditionally has been used as an indicator of the ability of an irradiated FE to withstand high temperatures resulting from off-normal conditions. It is often used for low or medium power density research reactors as a limiting temperature in the analysis of loss of coolant or transient accidents. It is measured by performing a blister test, as defined in Section 3. Blister temperatures of

successful fuels are typically above 450°C. Experience has shown that, for non-oxide fuels, blisters tend to form first at sites where oxidized fuel particles are in contact with the cladding [11].

4.4.5. Cladding corrosion behaviour

A FE under irradiation can experience two types of corrosion: general corrosion (mainly oxidation) and pitting corrosion:

— General corrosion is the natural result of irradiation of the cladding of an FE in water. For aluminium cladding, the corrosion product is a temperature dependant mix of hydrated aluminium oxides (bayerite, boehmite, gibsite), and its rate of formation depends on the water pH, the coolant–cladding interface temperature, and the heat flux. When pH is not low enough, an excessively thick corrosion layer can form. Since the thermal conductivity of this hydrated oxide layer is low, temperature drop across the corrosion layer can be significant and can lead to three problems:
  - Because temperature drop across the corrosion layer increases fuel meat temperature by an equal amount, an excessively thick corrosion layer can have a significant effect on fuel meat irradiation behaviour
  - When the corrosion layer reaches a thickness that results in a given temperature drop across it, the layer tends to spall, exposing a new cladding surface to corrosion. If spalling happens enough times at a given place, the cladding may be completely consumed
  - If the corrosion layer is excessively thick, the change in cladding and fuel meat temperatures at reactor startup and shutdown can be large enough that the difference in the thermal expansion of the cladding and that of the fuel meat can cause the cladding to separate from the meat [12], resulting in FE failure;
— If precipitates or foreign particles in the cladding are exposed during irradiation to coolant water containing certain ions, the possibility exists that pitting corrosion will occur, which can cause a hole through the cladding in a relatively short time. Pitting corrosion resistance is very sensitive to the type of aluminium alloy used for the cladding;
— Maintaining the pH of water at a low enough value can limit the amount of general corrosion formed if the cladding temperature is kept low enough, as mentioned above. Pitting corrosion can be prevented either by application of a thin, highly adherent oxide layer on the cladding surface at the end of the manufacturing process (so-called ‘prefilming’) or by using a highly corrosion resistant aluminium alloy for the cladding;
— The fuel developer must consider several issues with respect to cladding corrosion during fuel development and qualification:
  - Choice of cladding alloy
  - Choice of cladding treatment, i.e., whether or not to prefilm
  - Choice of test reactor(s), i.e., consideration must be given to the corrosion characteristics of the cladding alloy in the reactor environment
  - Design of irradiation tests with respect to the anticipated temperature drop across the corrosion layer.

4.4.6. Fission product release

Fission products are only released from the fuel meat of a FE during irradiation if a defect exists in the cladding which provides a path for fission gas to escape. A cladding defect can result from:

— Manufacturing or material problems, such as:
  - A flaw in the cladding material
  - Poorly bonded cladding in the end clad or rail regions of plate type FEs or poorly welded end pieces in pin or rod type FEs;
— Operational problems, such as:
  - Excessive cladding corrosion due to poor water quality
  - Mechanical fretting due to vibration
  - Erosion by debris or other foreign material
  - Fuel mishandling;

— Formation of a large blister or pillow during irradiation, which may result from a manufacturing problem;
— Melting of a portion of the cladding.

Experience has shown that once a cladding breach occurs and water contacts the fuel meat during irradiation, the rate of fission product release will increase, and some fuel particles will also escape [13]. All fuel meat types appear to behave in a similar manner. Therefore, it is not recommended that fission product release tests with intentionally breached FE cladding be performed unless such a test is required by a cognizant regulatory body. Additional information is provided in Appendix I.

The blister threshold temperature has also been shown to be the threshold temperature for fission product release from an overheated irradiated fuel plate [14]. A very small quantity of fission gas escapes through microcracks formed during blistering. If the temperature of the plate continues to rise, the release rate of fission products increases until the cladding and/or matrix melts, principally because the fuel cladding and/or fuel matrix material reaction rates rapidly increase in the presence of a molten reactant and the reaction releases much of the fission gas contained in the fuel.

4.5. FUEL ASSEMBLY PROPERTIES

4.5.1. Hydraulic and mechanical behaviour

Unless an FA is virtually the same as the assembly it is meant to replace, except for the fuel meat, a hydraulic test of the FA should be performed to assess its hydraulic and mechanical characteristics under coolant flow rate conditions in excess of those during its intended use. This is especially important in the case of FAs designed for high performance research reactors. Vibration of the FEs should be evaluated to determine if fretting or fatigue could result in cladding failure. The mechanical integrity of an FA should be assessed by measuring changes in dimensions or configuration. In addition, other stresses to which the FA might be subjected, e.g. during fuel handling operations, should be evaluated. It may be necessary to consider methods to limit such stresses.

4.5.2. Fuel assembly irradiation behaviour

Once an adequate fuel type and FE have been developed and the irradiation behaviour of full size FEs has been demonstrated, full size FAs (prototype FAs) obtained from a fuel manufacturer using a well developed manufacturing process should be irradiated to demonstrate that:

— Fuel elements and FAs meeting user specifications can be reliably manufactured under industrial conditions;
— The irradiation behaviour of the FEs and FAs under the worst expected normal operating conditions is satisfactory and consistent with irradiation behaviour revealed by tests of mini-elements and full size FEs and with their interpretation.

At least two[7] FAs should be irradiated under conditions that exceed those of their intended use by a comfortable margin. This means that peak fission rates and temperatures should at least be equal to peak values expected during normal use (i.e., at the upper limits of normal operating parameters), that their time histories should be prototypic, and that at least one of the assemblies should be irradiated to an average burnup well in excess of expected discharge burnup under normal use. It is even better if fission rates and temperatures can exceed the values expected during normal use by a reasonable amount. In some cases, this may require that the prototype FAs be irradiated in a test reactor that will never use these assemblies as driver fuel. This is obviously the case for a fuel that is being qualified for use in a new research reactor.

Successful completion of irradiation tests is demonstrated by proof of:

[7] See Section 6.2.2.5 for an explanation of why no fewer than two assemblies should be irradiated.

— Mechanical integrity of the FEs (absence of fission product leakage during all tests);
— Geometric stability of the FA (absence of local deformation of the FEs, such as blisters or pillows, and absence of any large scale deformation of the FA, such as plate buckling);
— Acceptable dimensional changes, e.g. FE thickness increase, FE deformation, FE twisting, with respect to geometric design margins;
— Assurance that, from an expert point of view, the fuel meat and FE are not unstable and unpredictable in behaviour.

To demonstrate that these requirements have been met, the PIE should include:

— Visual examination and photographs of the FA to determine if there is any evidence of a problem during irradiation. If the FA design allows it, a photograph through the coolant channels with backlighting is recommended;
— Dimensional measurements of the FA to demonstrate that its dimensions, straightness, and twist are within acceptable limits;
— Removal of one or more FEs, including the element with maximum peak burnup, for nondestructive and destructive examination. The number of elements to be examined should be larger if the fuel type or FE design is new. The examinations should include:
  • Visual examination and photographs of the FE to determine if there is evidence of any problem during irradiation; make note of any area of unusual appearance;
  • Gamma scanning to determine burnup distribution over the element;
  • Dimensional measurements to determine the approximate swelling distribution;
  • Metallographic examination of sections obtained from the regions of highest burnup, highest swelling, and average burnup to determine the meat microstructure and, if relevant, oxide layer thickness;
  • Radiochemical analysis of at least two samples to normalize burnup distribution determined by gamma scanning, unless the gamma scanning system provides absolute burnup values.
— Comparison of PIE data to irradiation behaviour data obtained from tests of mini-elements and full size FEs to show that behaviour was as expected and that there is no evidence of incipient unsatisfactory performance.

5. OVERVIEW OF THE FUEL DEVELOPMENT AND QUALIFICATION PROCESS

Sections 5.1 and 5.2 of this section describe a generic approach to the development and qualification of a **new** LEU fuel type; Section 5.3 is devoted to the subject of manufacturer qualification; and in Section 5.4, further qualification of previously qualified fuels is discussed. The fuel development and qualification process described focuses on tasks to be accomplished and the logical sequence of events. It neither describes the tasks in detail nor prescribes the way in which the tasks should be accomplished, because such details depend on the policies, practices, methodologies, and standards of a particular country and the organizations involved. The approach illustrated here is a consistent and integrated approach based on the experience of organizations currently involved in LEU fuel development and on international best practices. More detailed information is given in Section 6 and Appendix II, where practices in several countries are described.

The following participants are typically directly involved in development and qualification of a new fuel: a research reactor core designer, fuel developer, and fuel manufacturer. In some cases, the core designer and fuel developer may be part of the same overall research and development organization. In addition, the operating agents of irradiation and PIE facilities will be involved, and, again, could be part of the larger research and development organization. Finally, the regulatory body that will ultimately be asked to license the research reactor to use the fuel being developed is indirectly involved, at least at the beginning (through general

requirements issued for licensing, which guide the core designer) and at the end of the process (by deciding if a particular research reactor can be licensed to use the new fuel).

A phased approach is recommended for fuel development and qualification, with the first phase focused on research work and development (testing activities leading to the selection of a preferred fuel type), the second phase focused on fuel performance qualification to demonstrate that the selected fuel meets specified performance requirements, and the third phase focused on qualification of the fuel manufacturer.

5.1. PHASE 1 — FUEL RESEARCH AND DEVELOPMENT

A multidisciplinary project management team is normally selected to plan and execute fuel development activities. However, organization and responsibilities, work planning, work control, programme management, and quality management are outside the scope of this document and are not described here. The general tasks to be accomplished are outlined in the sections below.

5.1.1. Fuel concept design

The first step is to develop a fuel concept design taking into account specified requirements for a new LEU fuel, based on the following considerations:

— Rationale for developing a new fuel:
  - Political issues — There is an increasing consensus on the need to reduce uranium enrichment in civilian reactors (both existing and new), as proposed by the U.S. RERTR programme in 1978. To fulfil this goal, new LEU fuels which can replace HEU fuels — with comparable reactor performance — are needed. This implies a search for fuels with a slightly increased (5–15%) U-235 density of the fuel meat, which, in turn, implies an increase in the uranium density of the compound or alloy to be used;
  - Technical improvements — A new fuel may be required which can either enhance the performance of an existing research reactor or generic class of research reactors, or meet performance requirements for new research reactor designs. Because opportunities to develop new fuels are rare, it is important to focus not only on increasing fuel material density, but also on seeking to develop nuclear and structural materials able to accommodate increased fission rates and fission densities (U-235 burnups) and provide improved safety margins through enhancement of fuel material thermal conductivity and the reduction or limitation of fuel swelling. The first two improvements can result in: (a) higher neutron fluxes for research and development in nuclear energy and other scientific fields because of the possibility of operating at higher powers or with smaller cores, (b) an increased number of irradiation positions since the number of FAs can be reduced as FA reactivity and/or the fission rate increases, and (c) a decrease in the number of FAs used per year because of increased U-235 loading and reactivity of the FAs;
  - Backend issues — The loss of an existing backend solution for a currently used fuel can require a new fuel to meet a regulatory requirement that a backend solution be available. Specifically, a reprocessing feasibility study should be included within the framework of the global qualification pathway for the benefit of reactor operating organizations that use reprocessing for spent fuel disposition;
— Considerations affecting choice of a design concept:
  - Performance expectations — New fuel candidates should be designed taking into account neutronic and thermal–hydraulic requirements, such as reactor core reactivity, proposed fission rate, desired discharge burnup, proposed coolant flow rate, and temperatures that the fuel will experience during irradiation, which depend on the thermal properties of the fuel meat and FE. Such information is provided by the reactor core designer, and several iterations between the core design and the fuel–FE–FA designs may be needed, especially when a new, high performance research reactor is being designed;
  - Design requirements — The main design requirements concern the in-pile behaviour and properties of the fuel material, element, and assembly:
    - The fuel material should have a crystallographic structure that enhances its ability to retain fission products (especially fission gases). For example, a body-centred cubic crystallographic structure generally fulfils this criterion. It should be noted, however, that such a structure is required only after

enough fission gas has been generated for it to begin consolidating into bubbles. For example, experiments have shown that U_3Si_2 becomes amorphous very quickly once irradiation begins [15]. However, the amorphous material recrystallizes after a given amount of irradiation into a structure that very stably retains fission gas;
    - The fuel material should display adequate behaviour in terms of possible chemical reactions with cladding and matrix materials and with the coolant. This means that the irradiated fuel meat should contain either a small volume of reaction product phases or stable reaction product phases able to retain fission products (as discussed in the previous paragraph). The new fuel material must also be compatible with the reactor coolant, i.e., the fuel material should not react rapidly with the coolant;
    - New FE and assembly candidates for existing research reactors should be compatible with existing reactor coolant and hardware (avoiding things such as pitting corrosion and excessive galvanic corrosion of the cladding, increases in the FA mass so large that the core support structure must be changed, and dimensional changes so large that cooling channels become too narrow or that the FA cannot be easily removed from the core);
    - The FA must be able to withstand both thermal and hydraulic stresses without significant dimensional change;
- Safety expectations — Developing fuel, FEs, and FAs that will operate safely under specified operating conditions is a fuel developer's ultimate goal. In the context of fuel development, operating conditions are normally considered to include both normal operating conditions and at least some incident conditions. A considerable amount of fuel behaviour data discussed in Section 4 can be used to support analysis of incident and accident conditions. Also, the use of new fuel in a research reactor involves a careful evaluation of the radiological consequences associated with possible fuel damage (cladding failure or fuel melting). In this regard, data on fission product release factors in will be needed for the safety analysis of a new fuel. Before its implementation, the fuel qualification programme should be discussed with the necessary regulatory body in order to address requirements for additional data related to fuel behaviour under accident conditions. This may require the performance of special tests on the fuel under qualification;
- Assessment of potential manufacturing techniques — The technical and industrial challenges involved in manufacturing candidate fuels should be assessed, and eligible manufacturing processes for development can be selected based on this information. All activities needed for developing the technology to manufacture new fuel material at both laboratory and industrial scales should be included. Raw material specifications, material handling safety issues, manufacturing processes involved, and quality control methods, as well as cost evaluation, should be considered. To the extent possible, this activity should also include the assessment of technology for manufacturing at the industrial scale, including stability and statistical trends;
- Assessment of backend disposition options — Usually with the assistance of an industrial spent fuel disposition partner, an initial assessment of the ability to reprocess or store FAs in the long term must be made to avoid the use of fuel materials not suitable for these disposition options.

Based on all of the considerations listed above, a design concept should be chosen. Depending on the degree of maturity of the required fuel type and its manufacturing technology, more than one fuel candidate may be selected for development and different candidate manufacturing technologies can be pursued in order to guide the ultimate choice of a fuel and its manufacturing technology. A conceptual design document should be prepared, addressing the:

- Rationale for developing a new fuel;
- Choice of design concept for the FE and FA;
- Initial selection of candidate fuel and cladding materials;
- Initial selection of candidate manufacturing technologies.

5.1.2. Fuel manufacturing development

Based on the FE conceptual design and manufacturing assessment discussed above, fuel specimens will be manufactured for testing and assessment. Out of pile testing, as discussed in the next section, will be used to

characterize these specimens, and their properties will be compared to fuel specifications developed during the concept design. Through an iterative process, one or more manufacturing technologies will be developed to a level at which the development process can be transferred to a commercial scale fabricator.

The typical output or deliverable from this activity is a manufacturing process selection document that describes the following manufacturing processes and the rationale for each choice:

— Fissile material (meat) manufacturing;
— Cladding manufacturing;
— Fuel element (plates, rods, tubes) manufacturing;
— Fuel assembly manufacturing.

5.1.3. Out of pile testing

Out of pile testing should be performed to provide feedback to the manufacturing development task and to characterize the as-manufactured fuel samples prior to irradiation. Such knowledge is critical to evaluating changes in fuel properties caused by the manufacturing process and to evaluating the behaviour of the fuel under irradiation. The properties to be determined were discussed in Section 4.

5.1.4. Irradiation testing and postirradiation examination

The purpose of this activity is to assess the performance of the new fuel material and/or cladding material during irradiation under the conditions given in the performance specification developed during the concept design. Either mini elements or full size elements can be used in these tests; however, initial scoping tests can be performed much more efficiently with mini elements. This activity usually will include the definition of criteria to conduct irradiation tests and exploit information obtained from them, including scope of tests, irradiation devices and facilities, experiment details, initial measurements, in-pile and final controls, and PIE performance and analysis. The activities involved in these preliminary tests are (see also Section 4.4):

— Determination of material properties — All properties necessary to understand the behaviour of fuel material during manufacturing, under irradiation, after irradiation, and during storage, conditioning, or reprocessing must be determined by calculation, estimation, or measurement. Basic properties, material compatibility, and resistance to physical–chemical processes, such as corrosion in the presence of a coolant or interaction of fuel components, were addressed in Section 5.1.3 above;
— Determination of material performance levels — The goal of this activity is to determine the limiting conditions for satisfactory fuel performance; the most important parameters are generally operating temperature, fission density (or U-235 burnup), and fission rate. It is prudent to perform initial irradiation tests of a new fuel material at lower than normal irradiation power and fuel temperature, in order to enhance the safety of the experiment and avoid premature failures. The burnup should be meaningful or representative (more than 50% U-235). Fuel materials that show potential in initial tests can then be irradiated under more stringent conditions (with increased irradiation power and temperature and higher burnup) according to the proposed applications for the fuel;
— Selection of irradiation devices and facilities — Chosen irradiation facilities and core positions should be adequate to irradiate the new fuel under conditions set out in irradiation test specifications (neutron flux, irradiation period, thermal–hydraulic conditions and requirements). The irradiation device should be able to hold the FE in the selected in-core position, to be removed once the irradiation test is finished or stopped;
— Preparation of a safety report — A report addressing the safety of the experiment should be prepared and presented to the appropriate authorities (regulatory body, research reactor operating organization, or research reactor safety committee). This document should contain relevant information on the fuel material, the fuel test sample or element, and the expected performance of the fuel sample or element during the irradiation test;

— Performance of in-pile measurements — If the necessary facilities exist or can be procured, dimensional measurements and perhaps a sipping test to check cladding integrity should be carried out between reactor cycles according to the specifications of the irradiation test;
— Performance of PIE — The PIE should be performed according to irradiation test specifications and should include: (a) gamma profile measurements to determine fission–density distribution; (b) hot lab non-destructive testing, such as visual examinations and dimensional measurements; (c) destructive examinations, such as microstructural examinations of irradiated fuel meat and cladding (original and interaction phases, morphology of fission gases bubbles, cladding oxidation), and (d) radiochemical analysis for determination of absolute burnup;
— Preparation of final PIE report — A final report and technical assessment of performance against requirements and expectations should be prepared. This report should provide the basis, from an irradiation performance perspective, for a decision on whether or not to continue with the qualification process of a new fuel material.

5.1.5. Decision point — proceed to final design and qualification phase or continue development phase

The fuel developer will decide, based on information developed during the research and development phase (most importantly, through manufacturing and irradiation testing activities and economic assessment) whether there is sufficient information and confidence in a fuel to proceed to the next phase, or whether further development and testing are required to improve characteristics of the product and/or the manufacturing process. If further work is required, then one must iterate through the above steps as necessary. If results are satisfactory, then one can proceed to the next phase — fuel performance qualification.

5.2. PHASE 2: FUEL PERFORMANCE QUALIFICATION

Fuel performance qualification testing is limited to testing performed to demonstrate that the product, in the configuration to be used as a driver fuel in one or more reactors, meets specified requirements. This does not include testing to develop or improve characteristics of the product — a Phase-1 activity — or testing performed during or after production to ensure that the product meets quality requirements — a Phase-3 activity. Where the fuel to be qualified has been developed for a generic class of reactors, it is important that qualification irradiations be conducted under the most stringent conditions applicable to that class of reactors.

Usually, the qualification test FA is prototypic of fuel assemblies used in at least one of the higher performance reactors of the same class. It may be irradiated in the reactor in which it is meant to be used, or better yet, in a test reactor capable of creating conditions exceeding those reachable by the target reactor . There may be cases in which a full FA cannot be irradiated for lack of a suitable irradiation facility. In such a case, it is acceptable to use a partial FA, one having fewer FEs, in order to achieve desired irradiation conditions. An essential requirement is that the FA and irradiation conditions be prototypic enough to convince the regulatory body in charge to issue a license for use of the fuel in the proposed reactor.

Prior to fuel performance qualification, detailed design and technical specifications for the test FA should be prepared by the competent cognizant organization; for completeness, these activities are included below. In the following sections, the term ‘prototype FA’ can mean either a full or partial FA.

5.2.1. Detailed design

A detailed design should be developed for a prototype test FA based on design inputs, design requirements, design description (starting from conceptual design), information obtained from Phase-1 results (such as statistics on manufacturing and irradiation test results), and previous experience.

— Design Requirements:
  • Performance requirements (fuel meat temperature, fission rate, burnup, cladding temperature);
  • Design constraints (fuel stability, chemical compatibility);

— Design Description:
  • Material selection and specification;
  • Application of appropriate codes and standards;
  • Preparation of design drawings.

5.2.2. Technical specification

Technical specification documents should be prepared based on detailed design and requirements to procure one or more prototype FAs for qualification testing.

5.2.3. Prototype manufacturing assessment and development

An assessment may be required to identify whether technical and industrial challenges remain in the manufacturing of FAs with the proposed new fuel. This assessment should provide information about technical and industrial changes required in the manufacturing of the new fuel prototypic FEs (with respect to the process for manufacturing final irradiation test samples during Phase 1), if any. It is important that any changes to the manufacturing process made during Phase 2 are not detrimental to fuel performance, such as an unfavourable affect on the acceptable irradiation behaviour demonstrated during Phase 1. The following issues should be addressed:

— Manufacturing scale up issues for FEs;
— Maximum fuel loading limits;
— Fabrication porosity (both inside fuel particles and elsewhere in the fuel meat);
— Fuel meat uniformity (meat thickness and uranium density);
— Dimensional tolerances;
— Fuel element manufacturing;
— Fuel assembly manufacturing;
— Before manufacturing the LEU FAs for irradiation tests, it may be necessary or desirable to have one complete dummy FA or more (FAs containing either aluminium FEs or FEs with natural or depleted uranium in the fuel meat) manufactured to verify the:
  • Manufacturing process;
  • Design dimensions and tolerances, possibly demonstrated by insertion of the dummy assembly into an actual core position;
  • Hydraulic characteristics.

If any unfavourable characteristics are noted, either the design or the manufacturing process or both must be modified.

5.2.4. Prototype FE and FA manufacturing

Prototype FEs and FAs are to be manufactured with a new LEU fuel on a scale suitable to demonstrate feasibility of industrial manufacturing, according to technical specifications and procurement documents. The number of prototypes to be manufactured and the number of those which should be irradiated may need to be determined through interaction with the local regulator.

5.2.5. Qualification test planning

Before commencing the qualification test irradiation, a document should be produced for both out of pile and in-pile tests which gives:

— Objectives and requirements of the tests;
— List of tests to be performed, including any required PIEs;
— Test conditions (at a minimum, prototypic of the expected application);

— Acceptance criteria;
— Reports required;
— Other essential information.

In particular, the irradiation tests should demonstrate acceptable performance of the FA with respect to the design considerations discussed in Section 5.1.1, such as:

— Operating requirements ('Performance expectations' of Section 5.1.1), including:
  • Temperature range of cladding and fuel meat;
  • Fission rate range;
  • Expected EOL burnup;
— Functional requirements ('Design requirements' of Section 5.1.1), such as:
  • Stability of irradiation swelling;
  • Mechanical demands, such as stresses and strains generated by thermal expansion and irradiation effects (creep, etc.) and by hydraulic forces;
  • Thermal–hydraulic requirements (thermal conductivity, heat capacities, etc.);
  • Chemical compatibility of involved materials (fuel–matrix, fuel–cladding, and fuel–coolant interactions);
— Safety considerations ('Safety expectations' of Section 5.1.1), such as:
  • Radiological safety considerations, such as no fission product release.

5.2.6. Prototype full size FA irradiation testing, including post-irradiation examinations

The objective of these tests is to demonstrate that an FA containing a newly developed fuel and manufactured under industrial conditions performs satisfactorily and that fuel performance is consistent with that established during Phase 1. It is useful to irradiate at least two FAs under prototypic conditions — one to the normal burnup planned for the FA (expected EOL burnup) and one to a higher burnup to demonstrate a margin of safety. The post-irradiation examination (PIE) can be much less extensive than that performed during Phase 1. The activities include (see also Section 4.5):

— Irradiation — Irradiation should be carried out as required by the test plan to demonstrate acceptable performance, as noted in the previous section;
— PIE and analysis — A PIE should be performed to provide information on the following items:
  • Geometrical deformations (such as warping or twisting of the FEs or FA);
  • Fuel element (meat) swelling;
  • Irradiated fuel particle microstructure (fission gas bubbles, phases);
  • Blister threshold temperature;
  • Integrity of cladding (fission product release);
  • Cladding corrosion;
— Final report on irradiation and PIE — The report should include:
  • Irradiation conditions, based on measurement or calculation:
  • Reactor power history;
  • Coolant temperature history;
  • Fuel assembly power history;
  • Fuel element power history;
  • Fuel element heat flux history;
  • Discussion of any operational problems or abnormalities observed;
  • PIE results and analysis:
  • Processed data from PIE measurements;
  • Analysis of the processed data;
  • Comparison to expected results.

5.2.7. Fuel qualification report

A fuel performance qualification report should be prepared demonstrating that the newly developed fuel can be safely used within the realm of its intended applications. The report should discuss all evidence of appropriate and safe behaviour obtained during fuel development (Phase 1) and fuel qualification (Phase 2) activities. The items discussed should include:

— Fuel description;
— Material specification;
— Manufacturing process and manufacturer;
— Irradiation test conditions;
— Swelling versus burnup;
— Limiting irradiation conditions;
— Corrosion behaviour under irradiation;
— Fuel disposition options.

5.2.8. Fuel licensing

Proof that a fuel is qualified with respect to its irradiation performance characteristics lies in approval by a regulatory body for use of the fuel in at least one of its licensed research reactors. The reactor operating organization applying for a license to use a new fuel should attach a fuel performance qualification report to its application to cover all issues related to the fuel.

5.3. FUEL MANUFACTURER QUALIFICATION

Different manufacturers will likely have different policies, practices, and methodologies depending on the standards of each particular country and the national regulatory bodies involved. This good practices document does not describe details but provides a typical, generic roadmap for qualifying a new manufacturing process for research reactor fuel, based on international experience.

Generally, it is the customer (purchaser), guided by the requirements of regulatory bodies, who specifies manufacturing qualification requirements for a specific fuel supply contract and who determines whether a particular manufacturer is adequately qualified to manufacture fuel for the research reactor in question.

In practice, it is anticipated that all or several steps in the manufacturing process will have been previously qualified by a manufacturer involved in the fuel development programme described in Section 5.1 and which manufactured the prototype fuel used for the fuel performance qualification programme described in Section 5.2. Therefore, the basic manufacturing process flowchart is considered qualified at this stage.

Manufacturing process qualification itself is limited to activities performed to demonstrate that process and product meet specified requirements. This does not include activities to develop or improve characteristics of a product, or work performed during or after routine commercial production to ensure that a product meets quality requirements.

When the chosen manufacturer did not participate in the fuel development and irradiation performance qualification process and employs novel or proprietary methods involving materials not part of the original fuel qualification, a separate qualification exercise is appropriate. This would typically be a negotiated item between the purchaser and manufacturer, and the regulatory body governing the research reactor may be involved. Irradiation of lead test assemblies might well be required as part of the manufacturer qualification process.

Fuel and/or component designs outside the range of previously tested and qualified fuels are not in the scope of this section.

5.3.1. General manufacturer requirements

At the highest level, national regulatory bodies specify the requirements for manufacturing facilities handling nuclear materials. These are not described in this guideline because the requirements vary depending

on jurisdiction and, especially, because requirements do not pertain to specific manufacturing processes. Generally, each fuel manufacturer is expected to have business management plans and subplans that describe:

— Roles and responsibilities;
— Organization structure;
— Quality management;
— Risk management;
— Safety and licensing;
— Radiological protection;
— Physical security;
— Nuclear material control and accountability;
— Safeguards;
— Logistics–maintenance support plan;
— Training;
— Inspection and test plan, etc.

Usually manufacturers clarify these in a specific document describing the operation and management of their facility. In developing the technology (when the manufacturing process is mastered), documents are prepared describing production methods, manufacturing procedures, process control methods, inspection and test procedures, and working level documents, such as route sheets, travellers, and forms. Normally, manufacturing audits are performed (to various standards, such as ISO 9001, NQA-1, etc.), and the results are used for continuous quality improvement.

Generally, the manufacturer is expected to have:

— Demonstrated ability to manufacture the specific fuel type required;
— Demonstrated a quality management system, and associated plans;
— Demonstrated a radiological protection system;
— Demonstrated an occupational safety and health system;
— National regulatory approvals, as required.

5.3.2. Manufacturer qualification

As noted above, the purchaser determines the requirements for fuel manufacturer qualification. These requirements are driven by the desire to ensure that supplied fuel components will operate reliably, will be of high quality, and that the manufacturing and quality systems are robust enough to detect an out-of-specification product. The manufacturer must be able to demonstrate manufacturing of FEs and comparable assemblies as well as assemblies that have been qualified through irradiation testing (see Section 5.2).

Qualification may be demonstrated by producing a specified number of fuel components and then showing that these meet or exceed all specification requirements. Qualification of workers, inspectors, and related processes are also required as part of the overall qualification process, but this is generally covered by quality management systems.

The manufacturer is typically required to develop a qualification plan for all new processes, and upon completion of the qualification exercise, issue assessment or verification documents showing that the process performed as prescribed and the product meets specifications, including:

— Manufacturing process qualification; only materials that comply with specifications (except uranium enrichment) should be used;
— Qualification processes, materials, equipment, and operator qualification–training programmes should be identical to those used during production;
— The manufacturer should demonstrate to the customer's satisfaction that the manufacturing process is capable of producing fuel that satisfies all of requirements of the specification.

For example, the manufacturer must be able to:

— Demonstrate ability to manufacture statistically significant quantities of subcomponents (fuel meats, mini or full size FEs, etc.) to demonstrate that key manufacturing processes are in control and to demonstrate the range and tolerance limits for fuel specifications. Depleted or natural uranium may be used as appropriate;
— Demonstrate ability to manufacture full size prototype FEs and assemblies of the type required. Depleted or natural uranium may be used as appropriate;
— Manufacture prototype (see Section 5.2) or lead test assemblies for irradiation and comparison with a qualified reference fuel;
— Demonstrate traceability and auditability.

Audits of the quality system demonstrating adherence to internal systems and procedures can also be used to support manufacturer qualification.

5.3.3. Manufacturer requalification

Initial qualification is necessary but not always sufficient. In case of a long break in the manufacturing of a previously qualified fuel, the manufacturer must prove to the customer that:

— Fabrication quality is under control;
— The workers are still trained;
— The inspection equipment is still able to perform properly;
— The special process qualifications (welding, heat treatment, etc) are still valid;
— Previous or new subcontractors and/or material–component suppliers are either still qualified or have been requalified.

This can be carried out in different ways: through an audit by the customer, witness tests on specific processes, and manufacturing of test assemblies with depleted or natural uranium. Everything that can make a customer confident in the manufacturer's ability to still reliably achieve required quality criteria should be undertaken.

5.3.4. Manufacturer qualified by another customer

If a new customer desires to purchase FAs from a manufacturer that is currently qualified to manufacture FAs for another customer, (see Sections 5.1.1 and 5.1.2), the purchaser has several options including:

— Accepting the other customer's qualification and order the needed FAs;
— Order and irradiate lead test assemblies before ordering the remainder of the needed FAs;
— Require the manufacturer to establish qualification on certain items if the other customer's specifications were not as stringent in some respects (e.g., on fuel meat uniformity).

5.4. NEW APPLICATIONS OF PREVIOUSLY QUALIFIED FUELS

Many times a potential user of a previously qualified fuel (e.g. U_3Si_2–Al dispersion) wants to use that fuel in a FE or FA with a different design than that for which the fuel was originally qualified or under operating conditions outside of the envelope of conditions under which the fuel was originally qualified.

5.4.1. Irradiation conditions within the envelope of previous qualification

In this case, the inherent irradiation behaviour of the fuel meat can be considered well known and emphasis must be placed on showing that any changes in design and manufacturing of FEs and assemblies will still result in satisfactory irradiation performance:

— The simplest case is when the FE or FA design is very similar to a design for which the fuel has been qualified and the manufacturer has participated in the original fuel development and irradiation performance qualification process. Additional irradiation testing, e.g., with lead test assemblies (LTAs), is not considered necessary, unless specifically required by the cognizant regulatory body. Manufacturer qualification may be required, however, and as stated previously, when the manufacturer did not participate in the original fuel development and irradiation performance qualification process, irradiation of lead-test assemblies might well be required as part of the manufacturer qualification process;
— If the designs of the FE or assembly or anticipated operational hydraulic conditions are significantly different from those for which the fuel has been qualified, the potential user should analyze expected FE and/or FA behaviour under irradiation, based on existing qualification data. In addition, the manufacturer should demonstrate, through manufacturing and destructively examining a number of FEs, that the fuel meat and the fuel meat–cladding bond maintain their integrity during the manufacturing process. Depending on the details of design changes, it may be prudent to perform hydraulic tests to confirm the mechanical properties of the FE and FA. Only if analytical, manufacturing, or hydraulic studies identify items of concern is it necessary to consider performing additional FE qualification irradiations. The fuel developer–customer should confirm these results through irradiation of at least two LTAs under normal operating conditions in the research reactor for which the FAs are intended for use or in a test reactor capable of simulating expected irradiation conditions. Irradiation of LTAs might also be required as part of the manufacturer qualification. Normally, no PIEs of the LTAs other than visual examination in the reactor pool are required unless problems seemingly related to fuel behaviour are discovered during the course of, or after, irradiations are completed.

5.4.2. Irradiation conditions outside the envelope of previous qualification

The procedure for qualification of a given fuel for new use in a specific research reactor will be based on generic qualification of the fuel in question, supplemented by analysis of the intended use with respect to the original qualification conditions and specific tests under operating conditions that are representative of those in the research reactor in question (coolant temperature and pressure, fission rate, fuel–meat temperature, target burnup, etc.). The amount of effort required depends on the extent of departure from the envelope of operating conditions for which the fuel was previously qualified:

— If the qualification envelope expansion required is relatively small, data covering some of the expanded range of parameters may be available from the previous qualification. In addition, the existing data could be extrapolated for the new conditions and used to provide an estimate of expected fuel performance. In such a case, irradiation of LTAs may be sufficient to extend the qualification to the new conditions;
— If a large expansion in the qualification envelope is needed, a relatively extensive qualification programme will be needed. If operating conditions are being expanded to include higher fission rates, operating temperatures, and fission densities, the programme will be similar to that for a new fuel, as described in Sections 5.1 and 5.2. An example of this type of expanded qualification is given in Appendix II, Section II.2.2.

6. FUEL DEVELOPMENT AND QUALIFICATION PROCESSES CURRENTLY BEING USED

6.1. INTRODUCTION

The generic fuel development and qualification process described in the previous section was developed by choosing best practices from a number of current fuel development and qualification programmes. Currently used processes have evolved through a number of fuel development programmes, some going back more than a half century. This section describes the implementation of two separate fuel development and qualification processes in order to show how the generic approach described in Section 5 can be applied. The processes described are:

— The fuel development and qualification process generally followed in Argentina, Canada, France, the Republic of Korea, and the United States of America — Because this process has been carried out in a number of countries, and because of varying regulatory requirements, the process has not been standardized and codified. The general process followed, especially since the advent of the US RERTR programme in 1978, is described. Of course, this process was modelled on prior fuel development programmes;
— The fuel qualification process used in the Russian Federation — Development of the Russian process began during the time of the Soviet Union, and it has been further developed and codified by the Russian Federation.

6.2. FUEL DEVELOPMENT AND QUALIFICATION OUTSIDE THE RUSSIAN FEDERATION

The fuel development process has evolved since 1978; in particular, a much increased emphasis has been placed on gaining a detailed and more fundamental understanding of the irradiation behaviour of fuels being developed. This means that irradiation and PIE programmes have become more intensive and that increased emphasis has been placed on fundamental modelling of irradiation behaviour, especially swelling. Modelling is undertaken in anticipation that the results of modelling efforts will provide insights into the mechanisms of swelling and, thereby, help guide experiment programmes. Of course, successful modelling depends on accurate and detailed irradiation–behaviour data with which to compare the analytical results.

The phased approach suggested in Section 5 is used. However, in actuality, there is considerable overlap between the phases:

— Much of the data needed for fuel performance qualification (Phase 2) is acquired during the research and development phase (Phase 1);
— In most, if not all cases, a commercial manufacturer is involved in Phase 1, and manufactures the FEs and FAs needed for the irradiation tests of Phase 2. Therefore, the manufacturer will already have demonstrated its ability to successfully manufacture FEs and assemblies meeting specifications required for qualification, which constitutes an important part of manufacturer qualification (Phase 3).

Because fuel development and qualification is a long and expensive undertaking, developers in several countries have collaborated on certain aspects of the process to best use available expertise, equipment, and facilities and to increase both the variety and amount of data obtained.

The ultimate goal of a fuel development and qualification programme is for the fuel to be used in research and test reactors, which means that a regulatory body which licenses a research reactor must approve use of the fuel as part of the licensing process. The more regulatory body staff understand about the fuels under development and plans and progress of the development, the more efficiently they will be able to review the fuel behaviour qualification report. In the United States of America, fuel developers and Nuclear Regulatory Commission (NRC) staff overseeing research reactors have found it mutually beneficial for developers to keep USNRC staff informed of development progress and the qualification programme on a regular basis.

6.2.1. Research and development process

6.2.1.1. Conceptual design and programme planning

(a) Conceptual design

The conceptual design process has followed two general strategies:

— Initial development of a fuel concept, based on the need for high density or some other fuel property, for use in research reactors within a broad class, e.g., in plate type research reactors;
  - Based on the need for high uranium density, a literature search is made to collect as much information as possible on previous work with high uranium density compounds and alloys. The choice of fuel candidates and other FE materials will be limited if another fuel meat property is required, e.g., compatibility with existing commercial reprocessing flow sheets;
  - Potential FE manufacturing techniques are assessed. Initial efforts are generally focussed on minimizing changes in manufacturing techniques in order to take advantage of existing manufacturing infrastructure. In 1978 and again in 1996, this meant concentrating on aluminium matrix dispersion fuels clad in aluminium alloys;
— Initial development of a reactor specific conceptual FE and FA design to identify specific fuel density and performance requirements. This strategy is practical only if prior work to develop a fuel concept has shown a possible path forward. In general, manufacturing development and irradiation testing move rapidly toward, or even start at, the required density.

One example of such development is a new research reactor under design requiring a fuel with a higher than existing uranium density and which has the ability to operate under high power density conditions.

Another specific example is the development of high density TRIGA® ($UZrH_{1.6}$) fuel by General Atomics during the 1970s and 1980s.

(b) Programme planning

Concurrently with the conceptual design process, a fuel development and qualification plan should be prepared describing tasks to be performed during fuel development and qualification phases, along with estimated costs and schedules. Since the results of research are not predictable, and in many cases neither is funding, and since those who provide funding do not like open ended plans, the initial plans are usually 'success oriented', i.e., no resources are provided in the plan to fix the problems that inevitably occur. Therefore, the fuel development and qualification plan must be revised on at least a yearly basis to account for new developments during the year. In the early stages of fuel development, the plan focuses on research and development tasks.

6.2.1.2. Manufacturing development

Manufacturing development begins as soon as a conceptual fuel is identified. Initial manufacturing development is carried out in laboratories operated by, or by a commercial fuel manufacturer under contract to, the organization carrying out the fuel development. Choices depend on the availability of facilities.

Many fuel developers have found it useful to begin manufacturing development using surrogate fuel materials (materials having a similar density and similar mechanical properties to uranium) instead of actual uranium bearing materials in order to avoid the extra accountability and safety measures usually applied when working with uranium. In addition, some development may be needed to produce the fuel material itself, so use of a surrogate fuel will allow development of the FE manufacturing process to proceed while the fuel material manufacturing process is being developed.

As further discussed in the next section, performing much of the manufacturing development using scaled-down versions of the FE being developed (usually called mini-elements) is quite useful. It is important to keep the dimension related to foreseen manufacturing difficulties at a normal scale; for fuel plates or tubes, the thickness dimension is most important, while for fuel pins, the radial dimension is most important. After

problems in thickness or radial dimension are solved, full size FEs must be produced to ensure there are no scaling problems (or to solve those problems) and to produce full size irradiation specimens.

As early as possible, a commercial scale manufacturer of research reactor FEs and FAs becomes involved in the manufacturing development process. The manufacturer provides guidance on the potential commercial feasibility of various manufacturing methods being considered, and the fuel developer begins to transfer the process (or processes) under development to the manufacturer. When the fuel developer does not have the capability (perhaps because of the scale of the developer's manufacturing equipment) to manufacture full size FEs to test the effect of scale up on irradiation behaviour later in the development process, a manufacturer must provide full size elements for these tests.

6.2.1.3. Out of pile testing

Out of pile tests are used to determine basic fuel properties and as-manufactured fuel meat and FE properties, as discussed in Sections 4.2 and 4.3, respectively. These properties are needed both for comparison with post-irradiation properties to determine fuel irradiation behaviour and to provide basic data necessary for fuel qualification.

Out of pile tests are also used in the early stages of development to help select candidates for irradiation. While the reaction rates and reaction products formed may be very different under irradiation, irradiation rarely mitigates a problem identified during out of pile measurements. Therefore, it is prudent to seek a solution to any problems discovered during manufacturing of a particular fuel system or to abandon work on that fuel system. Similarly, poor resistance to corrosion in out of pile tests likely indicates even poorer resistance should a cladding breach occur during irradiation. Screening candidate fuels prior to irradiation is both cost and time effective.

6.2.1.4. Irradiation testing and PIE

(a) Mini-element tests

Using mini-elements for initial determination of basic fuel irradiation properties is advantageous because they are easier to manufacture on a laboratory scale, they require smaller amounts of material, and more samples can be irradiated in a given irradiation space, thereby making it feasible to evaluate a number of fuel, cladding, and/or manufacturing method candidates. In addition, it is usually easier to obtain permission from the operating organization of a test reactor to perform initial tests using mini-elements because the consequences of sample failure are more limited than when full size FEs are used.

If conceptual design has identified several potential fuel candidates, those not eliminated by out of pile tests are subjected to scoping irradiation tests to identify which candidates to select for further development. The mini-elements for the initial scoping tests of a new fuel material can be produced using moderate uranium densities in order to ease manufacturing problems, since the goal is to identify basic fuel meat properties, such as fuel–matrix interaction and fuel and fuel meat swelling phenomena. The initial scoping tests are usually performed under moderate irradiation conditions and to moderate burnups.

Measurements between irradiation cycles can provide valuable information on the rate of FE swelling or the onset of breakaway swelling. For example, when irradiating plate type fuel samples, it may be possible to design the irradiation device so that a visual inspection of the cooling channels can be made by illuminating the irradiation capsule from underneath and looking through the gaps in order to detect excessive plate swelling. Other devices, such as channel gap measuring gauges, either passive (go–no go) or active, or plate thickness gauges for removable samples can be used.

Non-destructive and destructive PIEs are performed on the irradiated samples. For those samples not exhibiting signs of excessive swelling during non-destructive PIE, metallographic examinations of fuel meat sections are performed to look for signs of incipient failure, such as the formation of large fission gas bubbles or the coalescence of smaller fission gas bubbles.

Based on the results of scoping tests, candidate fuels are selected for further development. If scoping tests indicate that a different conceptual FE design might lead to large improvements in a required fuel property, a decision can be made to pursue that path as well (e.g., develop both dispersion and monolithic FEs). A new

manufacturing method and additional scoping tests may be needed to pick candidate manufacturing methods to be pursued.

Following scoping tests, additional irradiation tests using mini-elements should be performed to study the irradiation performance of selected fuels under more stringent irradiation conditions, such as higher fission rates, fuel temperatures, and burnups. In fact, in order to provide a margin of safety, which is extremely useful when obtaining a license for a new fuel, some fuel samples should be tested beyond expected operating conditions and burnups set out in fuel performance specifications.

The PIEs are focused on searching for indications of incipient failure and on the measurement of fuel meat properties from which quantitative fuel performance, especially fuel and fuel meat swelling data, can be determined. These data are also used to develop fuel behaviour correlations and models.

(b) Full size FE tests

Full size FE irradiation tests are performed to detect any scale-up effects the mini-elements may be subject to. One such effect seen in at least one test is the separation of the cladding from the fuel meat of an FE owing to the difference in thermal expansion coefficients of the cladding and the meat [12]. The PIEs focus on areas of interest, such as the cladding–fuel meat interface in the example.

6.2.2. Fuel performance qualification

6.2.2.1. Fuel performance qualification plan

Fuel performance qualification (Phase 2) is demonstrated by irradiating full size FAs prototypic of research reactor FAs which may use the newly developed fuel. When a new fuel is applicable to a number of research reactors of a certain type, irradiation is carried out in a research reactor that can provide irradiation conditions as stringent as any envisioned for the fuel. Many times prototype FAs are irradiated in more than one test reactor.

The requirements for fuel performance qualification are usually incorporated into the most recent version of the fuel development and qualification plan discussed in Section 6.2.1.1 (b). Depending on the requirements of test reactor management, an additional and more detailed plan may be required.

6.2.2.2. Detailed design of test FA

An FA is designed to be compatible with available irradiation positions in the chosen test reactor. The staff of a reactor operating organization is intimately involved in the design of both the FE and the FA. If more than one test reactor is being used, a detailed design is prepared for each. In most cases, however, FE specifications are similar.

6.2.2.3. Technical specification

The technical specification developed for prototype assemblies is typically similar to technical specifications used to procure fuel currently being used in research reactors of the type that may use the newly developed fuel. Care is taken to assure that the specification for prototype FAs is consistent with specifications developed during Phase 1, especially regarding fuel chemical composition, including impurity limits, and fuel phase composition, in order to assure that the fuel will behave like that developed in Phase 1. It is also important that manufacturing criteria, such as fuel meat dimensional and density uniformity, is not more stringent than the manufacturing development of Phase 1 has shown to be achievable.

6.2.2.4. Prototype FE and FA manufacturing

As discussed in Section 6.2.1.2, the usual practice is for a commercial manufacturer to be involved in manufacturing development work and manufacturing of full size FEs for irradiation tests. In such cases, the extension to manufacturing full size FAs for qualification tests is not difficult. Prototype FAs are manufactured

and inspected under commercial conditions. A representative of the research reactor in which the prototypes will be irradiated always participates in inspections at the manufacturer's plant; this requirement is usually imposed by test reactor management.

6.2.2.5. Prototype FA irradiation and PIE

In most cases, at least two FAs are irradiated in a given test reactor in order to:

— Provide redundancy in the event that one assembly is damaged;
— Allow one assembly to be irradiated to 'normal' discharge burnup and the second to be irradiated to a higher burnup to demonstrate that a good margin to failure exists at normal burnup.

Post-irradiation examinations need not be as extensive as those conducted during the research and development phase, but they must demonstrate that prototype assembly behaviour is acceptable and performance is as expected based on the results of previous tests. At least one FE, typically the one having the highest local burnup, is removed from each FA for metallographic examination. The purpose of metallographic examination is to demonstrate expected fuel behaviour at both the macroscopic and microscopic scales and to demonstrate that there is no evidence of an incipient problem.

— Fuel behaviour qualification report and fuel licensing

A fuel behaviour qualification report is prepared containing and discussing evidence obtained during the research and development and qualification phases that the fuel will behave in a safe and predictable manner when manufactured and used appropriately. In general, the qualification report contains the information discussed in Section 4.

A research reactor operating organization applying to its regulatory body for a license to use the fuel must demonstrate through analysis that the irradiation conditions to which the fuel will be subjected are within the envelope of conditions described in the fuel behaviour qualification report. The operating organization can append the fuel behaviour qualification report to the license application as evidence that the proposed fuel will operate safely under proposed operating conditions.

6.2.3. Manufacturer qualification

As discussed above in Section 6.2.2.4, in most cases a commercial fuel manufacturer provides the FAs used for fuel behaviour qualification tests. In doing so, that manufacturer demonstrates that its manufacturing process is qualified. In fact, since the fuel developer is typically the customer and since a representative of the test reactor operating organization is intimately involved in inspections at the manufacturer's plant, the manufacturer will be qualified by a customer (as described in Section 5.3) when fuel behaviour qualification tests have been successfully completed.

Manufacturer qualification by subsequent customers has occurred on a routine basis for all fuels that have been developed. However, further discussion of this topic is beyond the scope of this good practices document.

6.2.4. Previously qualified fuels

In general, the process discussed in Section 5.4 has already been followed. Since the approval of NUREG-1313 in 1988, many research reactors have been converted from HEU fuel to LEU fuel using U_3Si_2–Al dispersion fuel. Many of the lower performance research reactors were converted without the irradiation of LTAs. However, LTA irradiations were required in several reactors (see e.g. Ref. [16]).

In addition, new research reactors are being designed or have already been built. The French CEA's Reacteur Jules Horowitz (RJH), currently under design, plans to initially use U_3Si_2–Al dispersion fuel under conditions far in excess of those for which the fuel has been qualified. A description of the programme to qualify the fuel under required conditions is presented as an example in Appendix II, Section II 2.2.

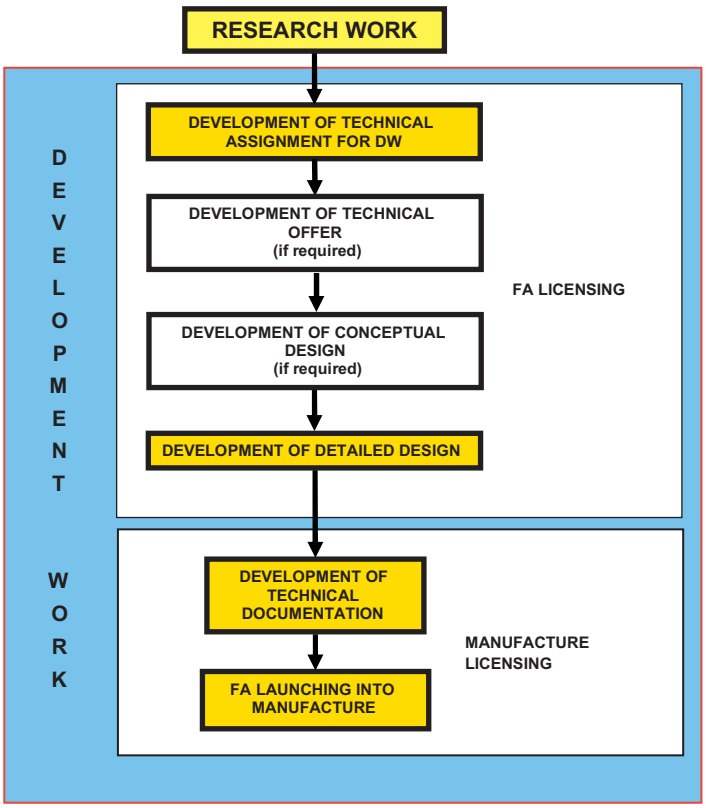

FIG. 1. System of development and the manufacturing launch of FAs for research reactors in the Russian Federation (procedure of FA and manufacturing licensing).

6.3. FUEL DEVELOPMENT AND QUALIFICATION IN THE RUSSIAN FEDERATION

6.3.1. Process description

Fuel assemblies for research reactors have been developed and manufacturing launched in the Russian Federation in accordance with Russian State standard ГОСТ Р 15.201-2000 СРПП "Products of industrial and technical designation — procedure of product development and launching into manufacture". The main phases of the process are presented in Figure 1.

6.3.2. Research work

The initial stage of the process is research work (RW), which consists of a package of theoretical and/or experimental research activities carried out with the aim of obtaining validated initial data and determining the principles and methods of making (or improving) the product. General requirements for RW organization and performance, the order of RW performance and acceptance, RW stages, the procedures for document preparation, negotiation, and approval during RW organization and performance, and the procedure of RW results implementation are specified in Russian State standard ГОСТ 15.101-98 СРПП "Procedure of scientific research and development".

Research work is carried out when developing fundamentally new designs and technological methods with the aim of:

— Determining the optimal line of research in relation to a specific problem;
— Comparing evaluation of the options offered by possible decisions, taking into account forecast oriented research results;

— Conducting theoretical and experimental research;
— Generalizing and estimating research results.

Research work is carried out in accordance with the Technical Assignment for RW (TA-RW), which is an initial technical document for RW performance specifying requirements for content, volume, and time of the work. The technical assignment is approved by the customer (if RW is performed according to an agreement) or by the manager of an enterprise performing RW (if RW is carried out on its own initiative and there is no customer; in this case, the development centre and the manufacturer take a commercial risk).

The TA–RW covers:

— Sources of work financing;
— Sequence and time periods for fulfilling stages of work;
— List of participants;
— Nomenclature and time periods for scientific–technical document preparation;
— Schedule for acceptance of RW stages and the RW as a whole.

Acceptance of an RW as a whole is carried out by an Acceptance Committee. For RW acceptance, the enterprise performing the RW must present:

— An approved TA–RW;
— Approved acceptance certificates for completed RW stages;
— An approved scientific–technical report on the RW and any other technical documentation on the RW stipulated by the TA–RW and the contract;
— Mock-ups, test programmes and procedures for mock-ups, if stipulated by the TA–RW and the contract;
— Recommendations and proposals regarding implementation and application of RW results.

RW acceptance covers consideration and examination of the results of work performed for conformity with the TA–RW, quality analysis of the technical decisions made, and, if required, testing of mock-ups with the aim of confirming research results. During RW acceptance, the scientific–technical level of research and validity of proposed decisions and recommendations regarding implementation and application of RW results aimed at manufacturing a competitive product are evaluated.

6.3.3. Development work

The next stage of the process is development work (DW), which consists of a range of activities covering the preparation of design and technological documentation and the production and testing of mock-ups, experimental prototypes, and developmental or production prototypes in support of design decisions specified in the design and analysis study. DW is carried out based on the results of previously performed RW or on the basis of developmental and operational experiences of products similar in application and principle of operation, as well as on the analysis of leading achievements in the development of advanced facilities.

At the DW stage, both FA licensing and manufacturing licensing are performed. A second technical assignment (for development work — (TM–DW)) is required for this stage.

6.3.3.1. FA development — FA licensing

A TA–DW approved by the customer and an agreement (contract) made with the customer provide the justification for DW. The main good practices for development of a TA–DW, for development of design and technological documentation, for acceptance of development results, for tooling up for and mastering of the manufacturing process, for tests of developmental prototypes and products manufactured during the mastering process, as well as for confirmation of their conformity with obligatory requirements, are specified in Russian State standard ГОСТ Р 15.201-2000 СРПП.

A TA–DW should specify technical–economic requirements for the product that define its consumer properties and efficiency of application, the list of documents to be jointly considered, and the procedure of

approval for development results. In a TA–DW, the fulfilment of all obligatory requirements in relation to the given product is foreseen, and the form of acknowledgment of product conformity to obligatory requirements specified by the laws in force is presented. The specific content of the TA–DW is determined by the customer and the development centre. No requirements contradicting the laws of the Russian Federation and obligatory requirements can be included in the TA–DW. If a product is being developed by a development centre on its own initiative, a TA–DW approved by the manager of the centre, based on the results of an assessment of the product market and on patent research, provides justification for the DW.

At any stage of the product's development, amendments and additions that do not violate the conditions of fulfilment for obligatory requirements can be inserted in the TA–DW if the customer and the development centre support these amendments and additions.

In Russian State Standard ГОСТ 2.103–68 ЕСКД "Stages of designing", the following stages of design and technological documentation development are specified:

(1) Technical offer (developed, if required);
(2) Conceptual design (developed, if required);
(3) Detailed design.

The obligation to fulfil the steps and stages of design documentation development is specified in the TA–DW for development. These stages are described in the following paragraphs.

Technical offer (TO) — a set of design documents that must contain a technical and technical–economic substantiation of feasibility for developing product documentation based on (a) an analysis of the customer's TA–DW and diverse options resulting from possible decisions, (b) a comparative evaluation of decisions, taking into account structural features and operational peculiarities of the product under development, (c) any existing similar products, and (d) patent research. The letter 'П' is assigned to technical documents developed at the TO stage. A TO is not obligatory, and is developed only if required. An approved TO is the basis for development of a conceptual and detailed design.

Conceptual design (CD) — a set of design documents containing fundamental design decisions that give a general idea of product construction and principle of operation, as well as data defining application, main parameters, and overall dimensions of the product under development. The letter 'Э' is assigned to technical documents developed at the CD stage. Like the TO, a CD is not obligatory, and is developed only if required.

At the CD stage, if required, manufacturing methods are developed, and mock-ups of the products or dummy models of some parts of them are manufactured and tested with the aim of checking design and/or construction decisions regarding the products under development. The CD is the basis for detailed design development.

Detailed design (DD) — a set of documents containing final technical decisions that give a comprehensive idea of the construction of the product under development and the initial data for the development of working documentation. Requirements for fulfilment of the detailed design of products are specified by Russian State Standard ГОСТ 2.120-73 ЕСКД "Detailed design". The letter 'Т'is assigned to technical documents developed at the detailed design stage. An agreed upon and approved detailed design is the basis for development of working design documentation (WDD). The procedure for detailed design development is presented in Figure 2.

At the detailed design stage of development, the following works are performed:

— Neutron–physics, thermal–hydraulic design, and reactor core stress calculations;
— Technical–economic substantiation of product development;
— Development of design documentation (DD) for an FA experimental prototype (mock-up);
— Development of FA manufacturing methods;
— Manufacturing of FA experimental prototype (mock-up). The necessity of manufacturing and the required quantity of mock-ups and experimental prototypes for design substantiation are defined by the development centre of the product. The list of mock-ups and experimental prototypes is specified in the TA–DW for the product;
— Bench and in-pile tests of FA experimental prototype (mock-up) according to individual test programmes. Test programmes are worked out by the product development centre or the organization performing tests;

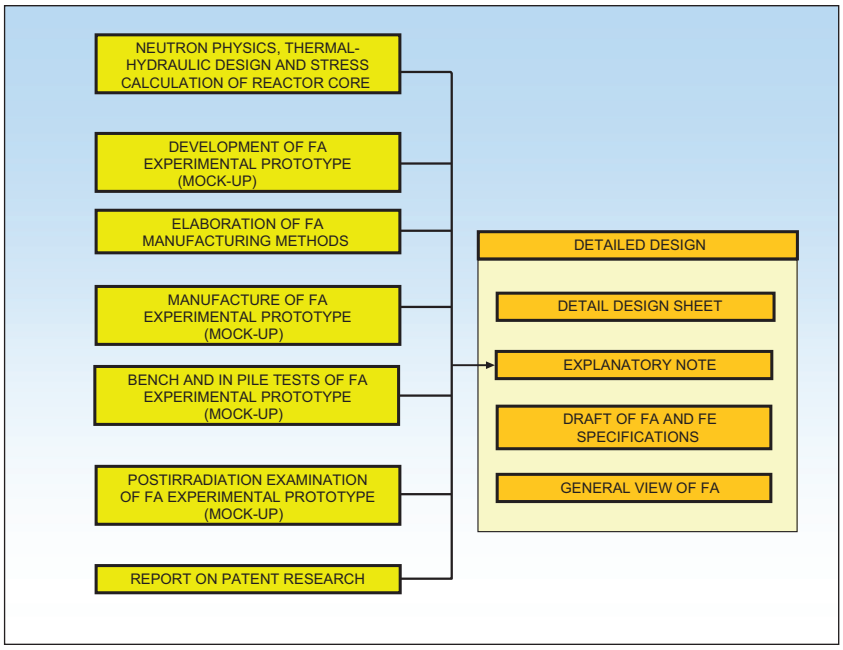

FIG. 2. Procedure for development of the detailed product design.

— Post-irradiation examination of FA experimental prototype (mock-up);
— Patent research.

When the above mentioned works are completed, the development centre draws up a detailed design for the product, which consists of:

— Detailed design sheet containing all included design documents;
— Draft of FA and FE specifications;
— General view of FA;
— Explanatory note, which consists of:
  • Neutron physics, thermal–hydraulic design and reactor core stress calculation;
  • Report on the results of bench and in-pile tests of an FA experimental prototype (mock-up);
  • Report on the results of PIE of an FA experimental prototype (mock-up);
  • Report on patent research.

At the customer's request, the detailed design can be considered at the meeting of the Scientific–Technical Council of Rosatom or at the meeting of the Scientific–Technical Council of the customer. The necessity for its consideration is specified by the customer in the TA–DW for product development. The FA detailed design is, without fail, examined and agreed to by the Russian Surveillance Authority (Rostekhnadzor). The detailed design must also be approved by the customer.

When the FA detailed design is approved, the FA development and licensing stage is considered complete.

6.3.3.2. FA launch into manufacturing — FA manufacture licensing

The launching of an FA into manufacturing (and FA manufacture licensing) is carried out in accordance with Russian State Standard ГОСТ Р 15.201-2000 СРПП.

An FA launch into manufacturing consists of the following stages (see Fig. 3):

— Preparation of working design documentation (WDD) for a developmental FA or a developmental batch of FAs;
— Preparation of technological documentation (TD) for a developmental FA;

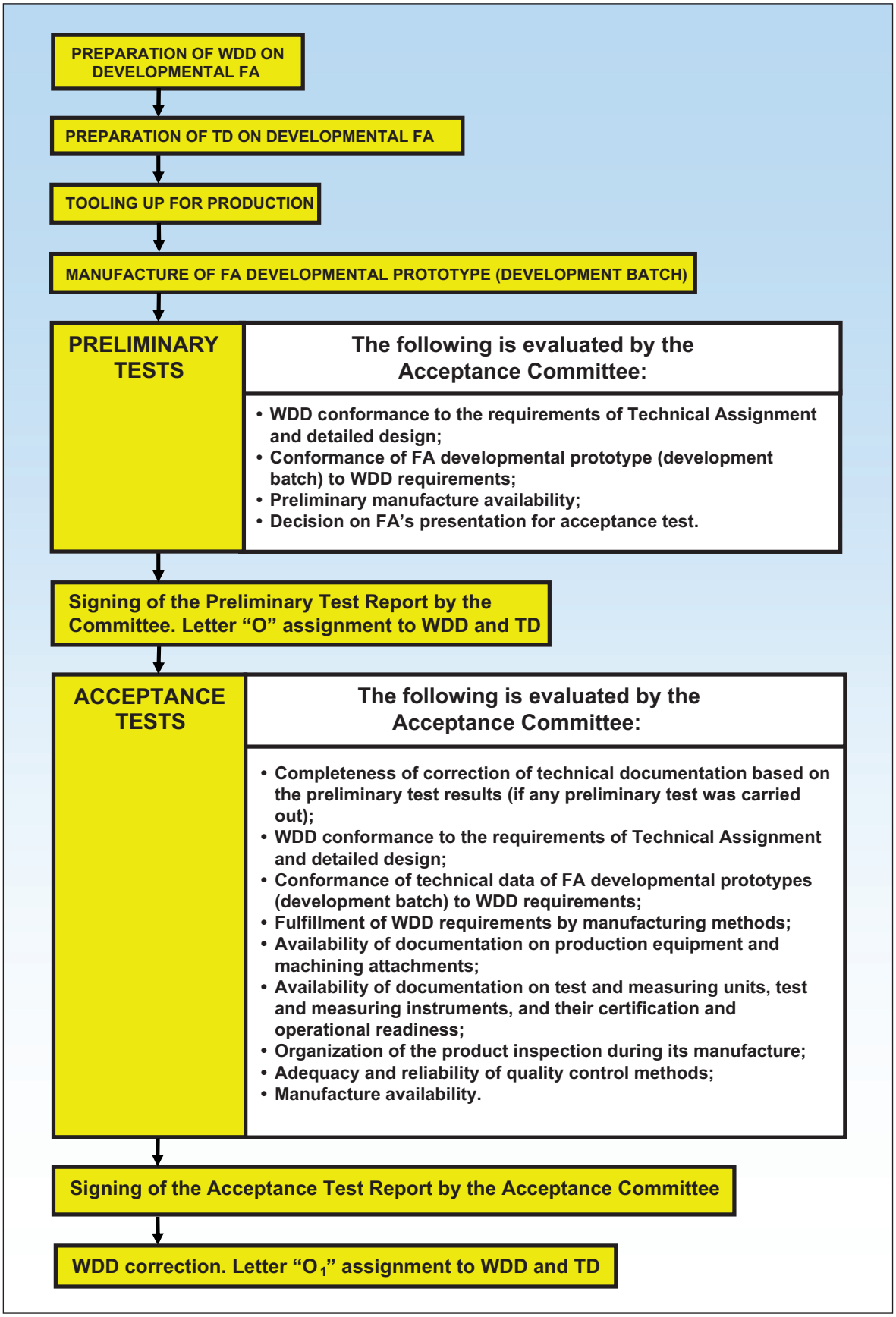

FIG. 3. Procedure of FA launch into manufacturing and manufacture licensing.

— Tooling up for production;
— Manufacture of an FA developmental prototype (development batch);
— Preliminary tests of an FA developmental prototype (development batch);
— Acceptance tests of an FA developmental prototype (development batch).

A detailed design agreed upon and approved according to the established order of the Russian Surveillance Authority (Rostekhnadzor) provides the basis for development of working design documentation (WDD).

In accordance with the rules of the unified system for design documentation, the WDD for the developmental prototype (development batch) is prepared by the manufacturing plant possessing a designing license or, if specified in TA–DW, by the detailed design development centre.

For prediction and quality control of results obtained at specific stages of DW, the FA developmental prototypes (development batch) are subjected to routine testing, as follows:

— **Preliminary test**: carried out with the aim of (a) preliminary assessment of the FA developmental prototype's (development batch) conformity with TA–DW requirements, (b) detection of possible design or manufacturing defects, and (c) determination of the possibility of its presentation for acceptance tests;
— **Acceptance tests**: carried out to determine (a) the conformity of technical data obtained for FA developmental prototypes (development batch) to approved Technical Assignment and detailed design documentation, (b) their manufacturing quality, (c) the adequacy and reliability of the quality control methods being used, (d) fulfilment of WDD requirements through manufacturing methods, and (e) determination of the possibility of their acceptance and manufacturing release.

The preliminary test is organized by the development centre; the performer of DW is involved in the preliminary test. Acceptance tests are organized by the development centre. Preliminary and acceptance tests are carried out according to the relevant test programmes and procedures worked out and approved by the party bearing responsibility for the performance of the tests.

The test programme covers the object of the test, goal of the test, scope of the test, conditions and procedures of the test, material support of the test, instrumentation support of the test, and reports on the test.

In test programmes, the lists of specific checks to be carried out in order to confirm the fulfilment of TA–DW requirements (with references to corresponding test procedures) must be given. Moreover, the programmes and procedures of acceptance tests for product developmental prototypes must include a quality check of the WDD and operation documentation (including draft detailed design) in order to make a decision on the adequacy and usefulness of the documentation for commercial production.

Test procedures cover:

— Product characteristics (properties, indicators) under assessment;
— Test conditions and procedure;
— Methods of processing, analysis, and assessment of test results;
— Test, control, and measuring devices used;
— Reports.

For the preliminary test, FA developmental prototypes (development batch) are presented. During the preliminary test the following steps are performed:

— Assessment of WDD conformity with the requirements of technical assignment and detailed design;
— Assessment of FA conformity with WDD and TD requirements;
— Preliminary assessment of manufacturing availability;
— Decision on FA presentation for acceptance tests.

Specified and actual data obtained during tests are recorded in protocols. Tests are considered complete if their results are entered in a report that confirms the fulfilment of test programmes and which contains concrete and accurate statements reflecting the conformity of the tested product developmental prototype with TA–DW requirements. The letter 'O' is assigned to the WDD and TD based on preliminary test results.

To run acceptance tests, an acceptance committee consisting of customer representatives, the development centre, the manufacturer, and Rostekhnadzor is appointed. The acceptance committee verifies the completeness, reliability, and objectivity of test results, as well as the completeness of information, observance of test time periods, and documentation of test results.

During acceptance tests, FA developmental prototypes (development batch) are manufactured and the following is checked:

— Completeness of correction of technical documentation based on preliminary test results (if any preliminary tests were carried out);
— WDD conformance to TA requirements and detailed design;
— Conformance of FA developmental prototype technical data (development batch) to WDD requirements;
— Fulfilment of WDD requirements through manufacturing methods;
— Availability of documentation on production equipment and machining attachments;
— Availability of documentation on test and measuring units and test and measuring instruments and their certification and operational readiness;
— Organization of product inspection during manufacture;
— Adequacy and reliability of quality control methods;
— Manufacturing availability.

The tests are considered finished if, after completion of the work, the results are entered into a report confirming the fulfilment of test programmes and which contains concrete and accurate statements reflecting conformity of the tested developmental product prototype to TA–DW requirements. The report shall contain:

— Statements regarding conformity of developed (manufactured) product prototypes to requirements specified in the technical assignment and detailed design;
— Recommendations regarding the possibility of further use of FA developmental prototypes;
— Decision on FA launch into manufacturing (FA developmental prototype advance to full scale production);
— Recommendations regarding letter 'O_1' assignment to WDD and TD.

When the acceptance test report is approved and signed by the Rostekhnadzor representative, the FA launch into manufacturing (and FA manufacture licensing) is considered complete.

Appendix I

SUPPLEMENTARY INFORMATION RELATED TO SECTION 4
(Information obtained by the fuel developer for qualification)

This appendix provides additional detail related to some of the topics discussed in Section 4.

I.1. FUEL MATERIAL CHEMICAL AND PHASE COMPOSITIONS

Examples of the importance of fuel material phase composition:

- Specification of U_3Si_2 chemical composition — During mini-plate irradiations, U_3Si_2–Al dispersion fuel was found to be much more stable than U_3Si–Al dispersion fuel. Therefore, when purchasing U_3Si_2 powder for full size FA tests, the lower limit of silicon content was specified to be at least 7.3 wt% (the stoichiometric composition when using LEU) in order to reduce the amount of the U_3Si phase in the powder;[8]
- Specification of U–Mo phase composition — It is known from literature that the g phase of uranium is much more stable under irradiation than the a phase; therefore, even though pure uranium is denser than uranium-molybdenum alloy, the latter has been chosen for use in high density fuels because it can be produced in a metastable g-phase state. Not only is a uranium alloy in a metastable γ phase state more stable under irradiation, it is also more stable in terms of its interaction with aluminium, the normal cladding and matrix material for dispersion fuels, at elevated temperatures. Thus manufacturing conditions can be less constrained.

I.1.1. Fuel meat volume

Fuel meat volume is normally determined by measuring total FE volume using the Archimedes method and subtracting the volume of the cladding (the cladding mass divided by its density). Since the cladding surfaces of irradiated FEs are always covered by oxide, which differs in density from the cladding, the oxide must be removed prior to application of the Archimedes method, and the mass of cladding removed must be determined by its weight difference in air from that of the as-manufactured plate.

The fuel meat volume of a monolithic fuel is easily calculated as the mass of fuel material in the FE divided by its density when the fuel material contains negligible internal porosity, as is usually the case.

I.1.2. Fuel volume and volume fraction

The fuel volume, in the simplest case, is just the mass of fuel material in the fuel meat divided by fuel material density. The issue becomes complicated when measurable interaction between the fuel and the matrix and/or cladding of a dispersion fuel or between the fuel and the cladding of a monolithic fuel occurs during manufacture and/or during irradiation. In such cases, the volumes of unreacted fuel and reaction product (which, of course, is also a fuel), must be estimated. Because the composition and density of any reaction product formed will likely be uncertain, volume estimates based on measuring area fractions of unreacted fuel and reaction product on micrographs produced from representative samples of the fuel meat are usually required. The fuel (interaction product) volume fraction, usually stated as a percentage, is the fraction of the fuel meat volume occupied by unreacted fuel material.

[8] It was later determined that minor amounts of the U_3Si phase in U_3Si_2 powder did not significantly affect the stability of the U_3Si_2–Al dispersion.

I.1.3. Matrix volume and volume fraction (dispersion fuel meat)

The matrix volume of dispersion meat is, in the simplest case, just the mass of matrix material in the meat divided by its density. The volume fraction, usually stated as a percentage, is the fraction of the fuel meat volume occupied by unreacted matrix material. The same considerations as discussed above for fuel are necessary if there is measurable fuel–matrix interaction.

I.1.4. Porosity volume and volume fraction

Pores form in dispersion fuel meat during the manufacturing process because the matrix material is too viscous to flow into all the spaces between fuel particles and into cracks formed in the fuel particles during the manufacturing process. The porosity in as-manufactured monolithic fuel meat is negligible. The volume fraction of porosity in the fuel meat is calculated by subtracting the fuel and matrix volume fractions from unity.

The amount of porosity in an as-manufactured dispersion fuel meat depends on the volume fraction of fuel particles in the meat, on the mechanical properties and size distribution of the fuel particles, and on details of the manufacturing process. When atomized fuel powder is used, the internal voids found in larger size particles contribute a negligibly small amount to total porosity, which is taken into account automatically when using the recommended method to calculate fuel volume.

Because the amount of fuel meat swelling of dispersion fuels during irradiation depends on as-manufactured porosity, there exists an optimum fuel particle size distribution that balances manufacturing and irradiation performance concerns. However, since non-optimized porosity generally is not very different from optimized porosity with respect to total fuel meat volume, the effect of optimization on fuel meat swelling is small and, if undertaken at all, optimization is usually done by the fuel manufacturer based on manufacturing concerns alone.

I.2. FUEL MEAT AND CLADDING THERMAL CONDUCTIVITIES

The thermal conductivity of as-manufactured fuel meat is usually measured either by determining the temperature drop across a sample for known heat flux or by determining the thermal diffusivity of a sample, e.g., using the laser flash method. Most often clad samples are used, and the thermal conductivity of the fuel meat is deduced from the known thickness and thermal conductivity of the cladding material. The thermal conductivity of the fuel meat can also be calculated through the use of various mixing formulas [17]. It is good practice, however, to use measured conductivities to verify and/or normalize calculated values. The thermal conductivity of the cladding is available from literature.

The thermal conductivity of the fuel meat changes as a fuel is irradiated. Early in the irradiation of a dispersion fuel it may increase slightly as the porosity volume decreases, owing to sintering of meat components. After a relatively short time, however, fuel meat thermal conductivity begins to decrease as the fuel component begins to swell and/or to react with the matrix, reducing the amount of high-conductivity matrix material. Porosity formed at the fuel–matrix interface or the fuel–interaction product interface also results in a decrease in thermal conductivity of the fuel meat. Because the rates of fuel swelling and fuel–matrix reaction increase as a function of temperature, so does the rate at which fuel meat thermal conductivity decreases, resulting in a positive feedback situation which has the potential to lead to fuel failure.

Since it is very difficult and expensive to measure the thermal conductivity of irradiated fuel, few data exist. Therefore, it is necessary to develop a method to estimate the irradiation effect on fuel–meat thermal conductivity, especially for dispersion fuels, because thermal conductivity can decrease by a factor of up to ten at high burnup. The method in Ref. [17] is recommended in such a situation.

Irradiation effects on the thermal conductivity of monolithic fuel meat should be much smaller; literature should be sought for relevant information.

I.3. FISSION DENSITY DISTRIBUTION

The relative fission density distribution is generally determined by gamma scanning and analysis of the gamma–ray spectra for the amounts of certain fission products within the volume of the FE 'seen' by the detector. Some gamma–scanning systems are calibrated to give absolute results. Care must be taken to collimate the γ–ray beam sufficiently to keep pulse pile-up and dead time in the detector within acceptable limits. To be able to convert relative to absolute fission density distributions and to verify the accuracy of those absolute distributions measured directly, samples removed from the FE are subjected to radiochemical analysis. It is good practice to analyze at least two, and preferably three, samples taken from areas of different fission density.

The basis for using as-manufactured fuel volume as the reference fuel volume for calculation of absolute fission density in a fuel is that the fission rate is essentially constant throughout the volume of a fuel particle. This assumption may be faulty for a thick monolithic fuel meat owing to neutron self-shielding in dense uranium fuel. In such a case, there will be a distribution of fission density through the thickness of the fuel; the peak/average value of this distribution will decrease as fission density (burnup) increases.

I.4. FUEL ELEMENT SWELLING

Fuel element swelling can be measured most accurately using the Archimedes method. For plate and tube type elements, an approximate value of FE swelling can be measured by comparing pre- and post-irradiation plate thickness measurements and for pin and rod type elements by comparing diameter measurements. This estimate is always larger than the actual swelling because measurements are always taken between high points on opposing cladding surfaces.

I.4.1. Fuel meat swelling

— Fuel meat densification from irradiation induced sintering of the meat: As the meat sinters during the early stages of irradiation, porosity is consolidated and fuel meat volume decreases; note that fission gas collected during irradiation will prevent as-manufactured porosity from being fully closed by sintering;
— Fuel meat swelling or densification from fuel–matrix or fuel–cladding interaction: The amount of this swelling or densification depends on the stoichiometry of the interaction product, its density, and densities of the fuel and matrix;
— Fission product swelling of original fuel particles and any interaction product: This is the most complicated contributor to fuel meat swelling. Because both unreacted fuel and interaction product are fuels, both produce fission products and undergo fission induced processes during irradiation and are thus discussed together. Therefore, unless a distinction is made, the term 'fuel' refers both to unreacted fuel and to interaction product. Much of the effort to analyze PIE results is directed toward measuring and understanding fuel swelling, which results from accumulated fission products. This includes an understanding of fuel behaviour, including that:
  • Solid fission products produce a constant amount of swelling per unit fission density, while the behaviour of gaseous fission products is more complicated;
  • Initially gaseous fission products are dissolved in the fuel material. They then begin to precipitate into bubbles which are initially so small they cannot be observed with a scanning electron microscope (SEM). This also results in a linear swelling rate as a function of fission density. The further behaviour of the bubbles depends on the nature of the fuel material under irradiation and on fission density (or burnup), fuel temperature, and the fission rate. In a fuel exhibiting stable swelling, bubbles can have diameters in the range of a few tenths to several micrometres at high burnup;
  • In fuels that remain crystalline or recrystallize during irradiation, bubbles tend to nucleate on grain boundaries and increase in size as fission gas diffuses to them; however, fission gas diffusivity in fuel is usually low and intercrystalline forces tend to restrain the growth of these bubbles, so swelling remains linear, but occurs at an increased rate. The change from lower to higher linear swelling rates occurs in a relatively short time, and the intersection of the two linear parts of the swelling versus fission density plot is referred to as the 'knee' of the overall plot;

- In amorphous materials, fission gas bubbles nucleate at various sites within the material, but because of the absence of crystalline restraining forces, bubble size is restrained only by surface tension. In addition, the diffusivity of fission gas in amorphous fuels is much larger than in crystalline fuels. These two phenomena combined result in a continuously increasing rate of volume in unconstrained fuel meat;
  - The result in plate or tube type fuels is the breakaway swelling phenomenon discussed above, which leads to unsatisfactory irradiation behaviour;
  - In a rod or pin type FE, the hoop restraint of the cladding and the axial restraint of the surrounding fuel are usually sufficient to prevent large fission gas bubbles from forming, except near the two ends of the element. At the ends, the cladding still provides radial constraint so that meat growth can only occur in length if, e.g., the end plugs of the element are not in full contact with the meat. For example, Al–U_3Si*Al dispersion fuel was qualified for use in rods (see Appendix II, Section II.3.5) [18], but abandoned for use in plates [19];

— Fuel meat swelling due to porosity generated by nucleation, growth, and coalescence of pores at the interaction product interface with the matrix and/or cladding: fission gas enters the matrix and any porous areas at the fuel or interaction product–matrix interface as fission fragments and through diffusion from fuel particles or the interaction layer. There are a few variations:
  - For many fuels, fission gas diffusivity is low enough that either no pores nucleate or nucleated pores remain separated and swell at a stable rate;
  - In some cases, however, the interaction product itself may have a large enough fission gas diffusivity that pores nucleate and collect fission gas at the interaction product–cladding or matrix interface at a rate that leads to coalescence and nonlinear increases in porosity volume, or breakaway swelling.

I.4.2. Fuel meat and fuel particle microstructures

The following types of examinations are performed:

— Neutron diffraction and small angle neutron scattering: These techniques can also be used to determine whether the fuel and/or interaction product is crystalline or amorphous. Sample preparation is much easier than that required for the transmission electron microscope (TEM).
— Optical microscopy: Transverse and longitudinal samples are taken in regions of interest, such as high and low fission density regions, one end of the fuel meat (including the meat end cladding interface), and regions of unusually high swelling or where there is evidence of an actual or incipient fission product leak. A determination is made of the extent of fuel–cladding interaction, the extent of porosity, any evidence of meat–cladding separation, as well as of microstructure details of remaining particles of original fuel material, the interaction layer, porosity, and fuel–cladding interface in areas that might indicate incipient separation. The extent of cladding corrosion also can be determined. Methods used to provide information include:
  - Low magnification pictures to obtain an overall look at the fuel meat and to see the extent of fuel–cladding interaction, the extent of porosity, any evidence of meat–cladding separation, and any evidence of actual or incipient breakaway swelling;
  - Higher magnification pictures, up to the practical limit of the metallograph, to examine details of the microstructure of remaining particles of original fuel material, the interaction layer, porosity, and fuel–cladding interface that might indicate incipient separation. Actual or incipient unacceptable behaviour is usually evident. Interaction layer thickness can be determined through dimensional measurements, and volume fractions of fuel, interaction product, residual matrix material, and porosity can be obtained by performing quantitative metallography on typical micrographs. The grain structure of fuel particles may be discernable after the sample has been air etched, i.e., stored in a hot cell for several days in an atmosphere containing some oxygen;
  - The extent of cladding corrosion can be determined during optical microscopy. However, careful preparation of a specimen is necessary;
— Electron–beam microscopy and microchemical analysis: Much higher magnification is available using a scanning electron microscope (SEM), while an electron microprobe (EMP) provides microchemical analyses, and a transmission electron microscope (TEM) is used to obtain crystallographic information.

Measurements of high interest include the size distribution of fission gas bubbles, the chemical composition of various phases of fuel material and interaction product, and the nature of fuel material and interaction product (whether they are crystalline or amorphous);

— Unless a shielded SEM is available, samples must be small because radiation levels are very high owing to high fission densities reached in fuel meat. Even then, some shielding will be needed around the SEM. With this technique, highly useful information can be obtained by studying fracture surfaces within the fuel meat, because polishing tends to obscure details, such as porosity, and to introduce artefacts. One of the most useful observations possible using an SEM is of the distribution of fission gas bubbles in the fuel particle or in the interaction product. A uniform distribution of small, separate bubbles indicates stable swelling of fuel particles. Incipient unstable swelling is indicated by bubbles beginning to coalesce, while large, randomly spaced bubbles are an indicator of fuel already experiencing unstable swelling. Zones containing distinct phases or compositions in fuel particles or the interaction layer can be identified. Backscattered electrons can be used to identify areas of different composition. Other possibilities include:
  - Although SEMs are optimized for imagery, when equipped with a wavelength dispersive X ray (WDX) analysis system, it is possible to perform chemical microanalysis of samples. An energy dispersive X ray (EDX) analysis system is not useful because the X–ray detectors are overwhelmed by gamma rays released from irradiated fuel samples;
  - An electron microprobe (EMP) is the preferred instrument for microchemical analysis, and is optimized for that task. Such information is especially useful when studying composition of the interaction layer and interdiffusion of fuel and matrix materials;
  - A transmission electron microscope (TEM) can be used to determine if the fuel and interaction product are crystalline or amorphous and to determine the crystal structure if the material is crystalline. (It must be noted that it is very difficult to distinguish a truly amorphous material from a nanocrystalline material.) As discussed earlier, fission gases diffuse more rapidly in amorphous materials (also, perhaps, in nanocrystalline materials) than in crystalline materials. In addition, a TEM may reveal basic information on the formation of fission gas bubbles. Samples for TEM study must be very thin, and their preparation is difficult, not only because of their required thinness, but because of their radioactivity;
— Neutron diffraction and small angle neutron scattering: These techniques can also be used to determine whether the fuel and/or interaction product is crystalline or amorphous. Sample preparation is much easier than for a TEM.

I.5. CLADDING CORROSION BEHAVIOUR

Additional information on general corrosion: Irradiation of essentially sibling FAs in two similar research reactors under similar conditions resulted in a difference factor of more than two in the thicknesses of corrosion layers formed. General corrosion is quite similar for the different aluminium alloys usually used as cladding under normal (<160–170°C) operating temperature conditions at the oxide–cladding interface; however, at temperatures a little higher some differences can be seen [20]. Thick corrosion layers appear to consist of distinct layers of different composition. Since the thermal conductivity of a tight oxide layer is small (usually taken to be 2.25 W/m·K [20, 21], although it is likely lower when multilayered [22]), temperature drop across the corrosion layer can be significant.

I.6. FISSION PRODUCT RELEASE

Here is some additional information related to the cladding defects listed in Section 4.4.6 of the main part of this document:

— Rarely does a flaw in cladding escape detection during the material receipt or manufacturing stages of the FE. The occurrence of such flaws should result in a review of inspection practices. However, cladding cracks can develop during irradiation as a result of stresses, especially from swelling fuel meat. The occurrence of such cracks indicates the need to re-examine an FE design with respect to the choice of

cladding material and mechanical design. Alternatively, modification of fuel meat to reduce swelling could be considered;

— Defects from poor bonding or poor welds will usually manifest themselves during irradiation testing in the early phases of manufacturing development. Such bonding problems must be resolved in the stage of continued manufacturing development;
— Blisters or pillows that result during development phase irradiation tests indicate inherent fuel irradiation performance problems or manufacturing problems that must be addressed during further development. Fission products are released through cracks formed due to stresses developed as cladding deforms;
— Fuel element (assembly) problems, such as vibration caused fretting, are normally discovered during hydraulic tests, which must be performed using an FE (usually a dummy) manufactured using the process being qualified and observing an appropriate envelope of hydraulic conditions. Such problems are corrected by redesigning the FA or by modifying the manufacturing procedure for attaching FEs to the FA structure;
— Most fission product releases during normal operation occur because of cladding corrosion (see Section 4.4.5):
  • General corrosion: As oxide forms during irradiation, cladding thickness decreases. If a fuel particle from the meat has penetrated too far into the cladding during manufacturing, the cladding covering the particle may disappear, allowing fission gas to escape and providing a path for water to reach the fuel meat. An anticipated amount of general corrosion must be considered during FE design so that sufficient minimum cladding thickness can be specified;
  • Pitting corrosion: This can result in a hole through the cladding that allows fission gas to escape and provides a path for water to reach the fuel meat. Pitting corrosion must be prevented.

If a fission product release is large enough, it will be detected by the reactor's fission product monitoring system. Prolonged release during irradiation will result in darkening of the FE surface in the shape of a plume as fission products are swept away by coolant.

Even though fuel materials and fuel meats currently in use or being developed are effectively benign in the as-manufactured state when in contact with boiling water, their behaviour is much different under irradiation. It is hypothesized that water is hydrolyzed in a high radiation field and that the fuel reacts with hydrogen and oxygen. As fuel reacts, contained fission gases are released and gases, both from fission and hydrolysis, can sweep liberated fuel particles through cracks, which may become enlarged as a result of gas pressure. In any case, the volume of fuel meat involved appears to be only a small fraction of the fuel meat in an FE. All fuel meat types appear to behave in a similar manner.

Appendix II

FUEL DEVELOPMENT AND QUALIFICATION PROCESS EXAMPLES

The following examples are presented to illustrate the processes presented in Section 6.

II.1. U_3SI_2–AL DISPERSION FUEL FOR PLATE TYPE RESEARCH REACTORS (US PERSPECTIVE)

II.1.1. Introduction

At its inception in 1978, the US RERTR programme began developing higher density dispersion LEU fuels based on uranium silicide alloys. Argonne National Laboratory-East (ANL-E), in Illinois, was the fuel developer. The development goal was to reach the highest practical uranium density in aluminium based dispersion fuel meat, the type of fuel meat being manufactured for most plate type research reactors, and to explore the possibility of higher densities with other types of fuel meat. Although this section is focused on qualification of U_3Si_2–Al dispersion fuel, the description includes information about work with other silicides in order to illustrate how a plan can change during the research and development phase.

II.1.2. Research and development phase

II.1.2.1. Conceptual design

A conceptual design study [23] identified two uranium silicide compounds of interest for use in Al-based dispersion fuels[9]: U_3SiAl[10] and U_3Si. U_3SiAl was the primary candidate owing to its better corrosion resistance. U_3Si_2 was not considered initially because its uranium density was more than 20% less than that of other U_3Si type candidates; however, it was soon added as a backup.

II.1.2.2. Fuel development plan

An initial fuel development and qualification plan was put forward [24] calling for manufacturing development to be carried out at ANL-E and irradiation of miniplates and full size FAs to be performed at the Oak Ridge Research Reactor (ORR) in Tennessee's Oak Ridge National Laboratory (ORNL). The miniplates and irradiation rig were to be designed by ORNL. Post-irradiation examination of the miniplates was to be performed at ANL. Following successful miniplate irradiations, the plan called for manufacture by commercial fabricators and irradiation at the ORR of full size FAs identical to standard ORR FAs except for the fuel meat constituents. The plan did not include irradiation of full size fuel plates. Finally, the plan called for a whole core demonstration in a research reactor to be later determined. (As an aside, the plan also called for the investigation of U_3Si and U–10 wt% Mo alloy in solid form as fuel meat, the latter with Zircaloy cladding.) The original plan envisioned the work being completed in five years. The following steps were taken:

— The fuel development and qualification plan was updated at least yearly, based on information obtained during research and development. In all cases, the schedule was extended for longer than yearly periods. Although no full size plate irradiations were included in the US part of the plan, the US began a collaboration with AREVA CERCA and the French group CEA. Full size plates of U_3Si and U_3Si_2 dispersion fuel were irradiated in CEA's SILOE reactor;

[9] Section II.1 is only concerned with the development and qualification of dispersion fuels; therefore, unless explicitly stated otherwise, references to fuel meat only indicate dispersion fuel meat.

[10] A fuel developed and patented by AECL in Canada for potential use in CANDU power reactors containing 3.5 wt% Si and 1.5 wt% Al. AECL found it to be more corrosion resistant in hot water than pure U_3Si.

— The discovery of unstable behaviour in U_3SiAl fuel in 1982 led to a revamping of the fuel development and qualification plan in late 1982 to early 1983. U_3SiAl fuel was dropped completely and replaced with U_3Si fuel as the highest density candidate. Also, more emphasis was placed on U_3Si_2. A new series of miniplate irradiations was planned to establish the performance limits of U_3Si and U_3Si_2 dispersion fuels, with uranium loading goals of 5.5 and 7.0 Mg U/m^3 (50 and 48 vol.% fuel), respectively, and to determine if the addition of a small amount of Cu to U_3Si might stabilize its swelling behaviour. Because initial tests of U_3Si_2 dispersions indicated it might perform well to much higher fission densities than those reached in tests up to that point in the programme, the plan called for irradiation of four 4.0 Mg U/m^3 U_3Si_2 miniplates at 40% enrichment and two 1.7 Mg U/m^3 U_3Si_2 miniplates at 93% enrichment, as well as U_3Si miniplates at these enrichments, to assess irradiation performance margins [25]. A uranium silicide composition with silicon content half way between that of U_3Si and U_3Si_2 was to be tested to see if the good behaviour of U_3Si_2 would dominate the performance of this alloy. Finally, a few miniplates containing USi were included, since this compound was likely to be present in highly burned U_3Si_2 dispersions.

II.1.2.3. Manufacturing development at ANL

A brief history of manufacturing development at ANL for silicide dispersion fuels follows:

— Manufacturing development included creating a method to reduce the tough U_3Si and U_3SiAl materials to powder and development of a roll-bonding technique to produce miniplates with fuel volume fractions in the fuel meat of up to 60 vol.% fuel, which would give uranium densities of approximately 8.8, 8.1, and 6.6 Mg U/m^3 in U_3Si, U_3SiAl, and U_3Si_2, respectively, which would also meet uranium homogeneity and minimum-cladding specifications for the miniplates. By mid-1980, a manufacturing method based on the standard picture–frame and roll–bonding techniques had been developed to the point that 31 miniplates with fuel volume fractions ranging from 31–44% had been manufactured for irradiation tests [26];
— By 1982, work had begun to scale up manufacturing techniques for full sized plate fabrication. First, tungsten powder was used as a surrogate for uranium silicide powder; later, depleted uranium was used;
— In 1983 manufacturing development focussed on producing miniplates for a second series of irradiations;
— In the period between 1978 and 1987, more than 250 uranium silicide fuel plates were manufactured for use in additional irradiation tests, in a whole core demonstration in the ORR, as LTAs for reactors planning to convert, and in reactor conversions.

II.1.2.4. Collaborations with commercial manufacturers, other fuel developers, and test reactor operating organizations on manufacturing development and irradiation testing

By mid–1979 collaborations, through no cost commercial type contracts, had been established with commercial manufacturers[11] of plate type FEs and FAs — AREVA CERCA in France and NUKEM in Germany and with fuel development group CNEA in Argentina. These collaborations focused initially on fabrication of full size FAs or miniplates for irradiation in the ORR with existing fuel types, but the fabricators also quickly began to develop uranium silicide fuels. In 1982, a similar collaboration was established with The Babcock & Wilcox Company (B&W), which had become the sole US manufacturer of plate type FEs and FAs in 1983:

— Manufactured by AREVA CERCA:
  • Full size plates containing U_3Si at 5.5 and 6.0 Mg U/m^3 and a full size 23-plate FA containing U_3Si at 6.0 Mg U/m^3 for irradiation testing at the SILOE reactor, in collaboration with the French CEA; four full size plates containing U_3Si_2 at 2.0–5.4 Mg U/m^3 and a full size, 23–plate FA containing U_3Si_2 at 5.2 Mg U/m^3 for irradiation testing the SILOE reactor;

[11] Several commercial manufacturers have changed their names during the past 25 years. In order to avoid confusion, the current name of the manufacturer is used.

- Three full size, 19–plate FAs containing nominally 4.8 Mg U/m^3 U_3Si_2 dispersion fuel, two with the standard (0.51 mm) meat thickness for testing in the ORR and one with 0.76 mm meat thickness for testing in the R2 reactor at Studsvik, Sweden under a collaborative agreement; one full size assembly with 0.76 mm thick, 4.8 Mg U/m^3 U_3Si_2 dispersion fuel for testing in the HFR–Petten reactor at Petten, the Netherlands, under another cooperative agreement with ECN (now NRG);
- One full size 20–plate assembly with 0.76 mm thick, 5.5 Mg U/m^3 $U_3Si_{1.5}$ dispersion fuel for testing in HFR–Petten; one full size assembly with standard meat thickness containing 4.7 Mg U/m^3 $U_3Si_{1.6}$ dispersion fuel for testing in the CEA's OSIRIS reactor at Saclay, France;

— Manufactured by NUKEM:
- Six miniplates containing U_3Si of up to 6.9 Mg U/m^3 for irradiation in the ORR;
- Two full size ORR type FAs, each containing 19 fuel plates with 4.8 Mg U/m^3 U_3Si_2 fuel meat for irradiation at the ORR; one full size, 19–plate FA containing nominally 4.8 Mg U/m^3 U_3Si_2 dispersion fuel with 0.76 mm meat thickness for testing in the R2 reactor at Studsvik;

— Manufactured by B&W:
- Two full size ORR type FAs, each containing 19 fuel plates with 4.8 Mg U/m^3 U_3Si_2 fuel meat for irradiation at the ORR;
- One full size, 19–plate FA containing nominally 4.8 Mg U/m^3 U_3Si_2 dispersion fuel with 0.76 mm meat thickness for testing in the R2 reactor at Studsvik;

— Manufactured by CNEA:
- Three U_3Si miniplates with densities between 5.2 and 6.1 Mg U/m^3 for irradiation in Module 6; one U_3Si and six U_3Si_2 miniplates for irradiation in Module 30;
- One full size ORR type FA containing 19 fuel plates with 4.8 Mg U/m^3 U_3Si_2 fuel meat for irradiation at the ORR; however, the ORR was permanently shut down before the FA was shipped to ORNL.

By 1982 technology transfer for full size plate manufacturing was underway at ANL. In addition to working with the three manufacturers participating in the fuel development and qualification work, uranium silicide manufacturing technology was ultimately transferred to fuel manufacturers/developers in Argentina, Brazil, Denmark, Indonesia, and the Republic of Korea.

II.1.2.5. Out of pile tests

Brief descriptions of the out of pile tests made follow:

— As part of manufacturing development work, several studies were performed to provide information for further development work and fuel qualification [26], including:
- Fuel alloy density measurements;
- Fuel alloy microstructure studies;
- Compatibility studies — Intended to simulate fuel–matrix–cladding interaction effects during irradiation by high temperature, long term anneals. Although different mechanisms lead to excessive plate swelling during irradiation than those leading to excessive plate swelling in compatibility tests, poor results during compatibility tests were invariably an indicator of poor irradiation performance;
- Corrosion tests — Miniplates with a 3.25 mm diameter hole drilled through the cladding and the fuel meat were exposed to boiling water for 168 h. These tests always produced benign results;
- Study of U–Si–Al phase relationships [27];

— Small fuel plate samples were also manufactured for:
- Thermal conductivity measurements of unirradiated plate type samples — Provided U_3Si–Al and U_3Si_2–Al dispersion thermal conductivities as a function of volume loading at 40°C and 60°C [28];
- Measurements of energy release from the fuel–aluminium exothermic reaction, using the differential thermal analysis (DTA) technique [29];
- Investigation of the reprocessibility of uranium silicide dispersion fuels — Experiments conducted at Savannah River Laboratories (SRL) demonstrated, using both unirradiated (depleted U_3SiAl, U_3Si, and U_3Si_2) and irradiated (~37% burnup U_3Si from Module 3, see below) dispersion FE samples, that silicide

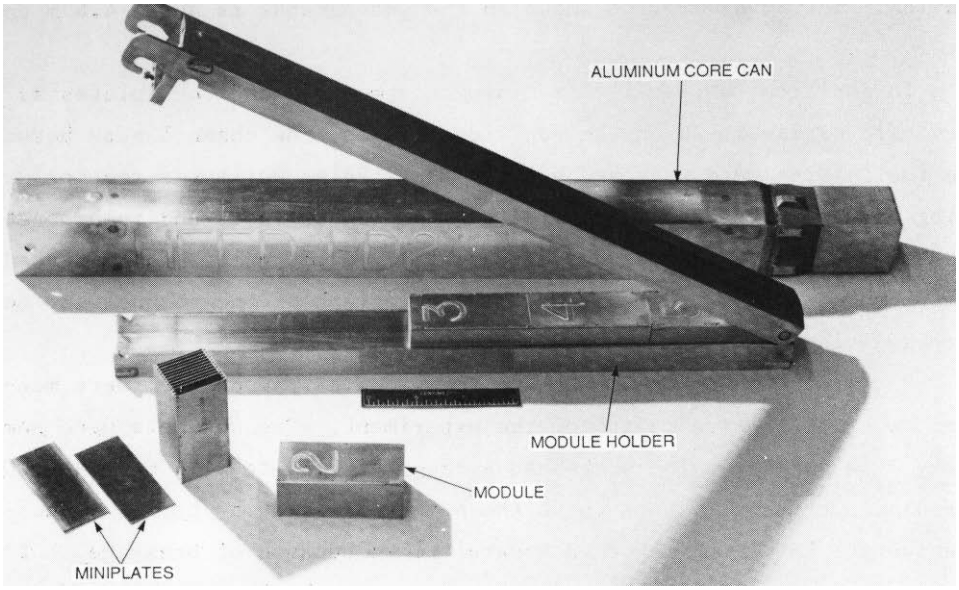

FIG. II.1. Components of miniplate irradiation rig used at the ORR.

dispersion fuels could be reprocessed using the same flow sheet utilized for reprocessing HEU dispersion fuels [30].

II.1.2.6. Miniplate irradiation tests and PIE

The initial miniplate irradiation test, which began in July 1980, included seven U_3Si and 12 U_3SiAl miniplates. A description of the miniplates, the modules that each held 12 miniplates, and the irradiation rig that held five modules is given in Ref. [31]. Figure II.1 shows the irradiation modules and irradiation rig.

- Module 3, containing five U_3Si and seven U_3SiAl miniplates, was removed for PIE in early October 1980 after achieving a calculated 34% average U-235 burnup. Post-irradiation examination at ANL showed good behaviour of the fuel at that burnup, although some small bubbles in the fuel particles were seen during SEM examination [32];
- Module 3 was replaced with Module 7, which contained four U_3Si_2 miniplates with a fuel meat uranium density of ~3.75 Mg U/m^3, as well as additional U_3Si and U_3SiAl miniplates. Irradiation of the test rig resumed in November 1980. The irradiation of Module 7 ended in November 1981 with an average burnup of 90%. Irradiation of another of the original modules (Module 4), which contained two U_3Si and two U_3SiAl miniplates, ended in May 1982 with an estimated average burnup of 96%;
- Post-irradiation examination of the miniplates from Module 7 was performed at ANL [32]. A drastic change was seen in the bubble morphology of the U_3SiAl fuel meat: the number of bubbles was greatly increased. Some had grown quite large by linking up with neighbouring bubbles, and the rate of bubble growth appeared to accelerate as burnup increased. It was concluded that the U_3SiAl fuel had entered a stage of breakaway swelling by 83% burnup, leading to pillowing of the plate at 93% burnup. The U_3Si fuel exhibited some fission gas-bubble formation and growth, but remained stable to 93% burnup. The U_3Si_2 fuel, which was only examined at 85% burnup, showed no evidence of fission gas-bubble formation [19];
- The first series of miniplate irradiation tests ended in June 1983. The 58 uranium silicide miniplates irradiated included 36 containing U_3SiAl, 18 containing U_3Si, and four containing U_3Si_2. Two of the U_3Si_2 miniplates originally irradiated in Module 7 were returned from ANL to ORNL after non-destructive PIE for further irradiation in Module 13; an average burnup of 98% was reached. PIE indicated that the irradiation behaviour of the two U_3Si_2 miniplates was extremely stable even at very high LEU burnup;
- A second series of miniplate irradiations began in March 1984; it ended in March 1987, when the ORR was permanently shut down for programmatic reasons.

FIG. II.2. ORR LEU fuel assembly and fuel assembly cross-section.

II.1.3. Fuel behaviour qualification

II.1.3.1. Qualification plan

With PIE evidence of the plates from Module 3 tested in 1981 that U_3Si_2 behaved very stably under irradiation, plans were made to begin manufacturing development for, and manufacturing of, full size U_3Si_2 FAs by commercial fabricators AREVA CERCA, NUKEM, and B&W for use in qualification irradiations in the ORR and subsequent PIE at ORNL.

II.1.3.2. Full size FA irradiations in the ORR and PIE

As already discussed in Section II.1.2.4, AREVA CERCA, NUKEM, and B&W each manufactured standard ORR FAs for irradiation in the ORR. Each assembly contained 19 fuel plates with 0.51 mm thick fuel meat. The cladding of the outer two plates was thicker than the cladding on the inner plates. Fabrication specifications were based on the standard ORR fuel plate and FA specifications, modified as appropriate for U_3Si_2 fuel. An ORR LEU FA is shown in Fig. II.2.

The FAs were irradiated in the ORR core in several locations each to simulate the environment that would be experienced during normal core fuel management. Between irradiation cycles, channel gap thicknesses were measured using a special probe (which had also been used during miniplate irradiations). One FA from each manufacturer was irradiated to about average discharge burnup (50–55%), and the second assembly was irradiated to about 80% average burnup, which took about twice as long.

Results from the various PIEs [33]:

- Visual and dimensional assembly inspection — No unusual features were seen, and dimensions of the FAs were essentially within the original manufacturing specification envelope;
- Channel spacing measurements — Channels were generally uniform with no abnormalities noted;
- Visual fuel plate inspection — The plates appeared to be in excellent condition with a uniform corrosion film over the meat and no apparent blisters, unusual swelling, or other defects;
- Plate thickness measurements — Average thickness increases in the high burnup region of the fuel plates ranged from about 40 μm in the ‘normal’ burnup plates and about 50–110 μm in the high burnup plates. The variation in thickness change at high burnup correlate well with the amount of as-manufactured porosity in the plates;
- Burnup analysis — Radiochemical burnup analysis was performed using samples removed from the part of the plate having the highest peak burnup; these were taken next to samples removed for metallographic examination. The radiochemical results were used to calibrate gamma-scan results;
- Blister threshold temperature measurement — Selected plates from each assembly were subjected to the standard post-irradiation blister test. Results showed that the blister temperature was greater than 525°C for each plate tested;
- Metallographic examination — One sample from the highest burnup region of one plate from each assembly was examined by optical metallography. In general, the microstructures were as expected from previous miniplate results and revealed no abnormal conditions. The plates with more of the U_3Si phase showed areas with larger gas bubbles, as expected. However, there was no evidence that bubbles would coalesce over large areas.

The final assessment was that assemblies from each manufacturer performed in a completely satisfactory manner.

II.1.4. Fuel qualification report and NRC review

Two reports were produced to be submitted to the USNRC for review. The first was a detailed description of the irradiation and PIEs of the six U_3Si_2 assemblies irradiated in the ORR [34]. The second was a compilation and discussion of the properties of U_3Si_2 dispersions [35]. These reports were submitted to the NRC in October 1987, with a request that the fuel be granted generic approval for use within the testing envelope in NRC licensed research reactors.

In July 1988, NRC issued NUREG-1313 [3], granting the requested approval.

II.2. U_3SI_2–AL DISPERSION FUEL FOR PLATE TYPE RESEARCH REACTORS (FRENCH PERSPECTIVE)

II.2.1. Programme conducted for OSIRIS conversion

II.2.1.1. Account of the changeover of OSIRIS to uranium silicide fuel

A decision to launch the 'changeover' of OSIRIS was made in 1989, France having opted to replace Caramel fuel (UO_2 enriched to 7.5%) with uranium silicide dispersion fuel, for technical and financial reasons, including the desire to standardize the industrial manufacturing of research reactor fuels. The decision was largely made on the basis of the USNRC evaluation report into the safety of silicide fuel (NUREG-1313, Ref. [3]), and thus the results of the programme ANL carried out at the ORR, but also on the basis of specific results of a French experimental programme conducted at the SILOE reactor on full size fuel plates representative of those in industrial use.

The decision was followed by a full set of neutron, thermal–hydraulic, and operating safety studies, and by the launch of an additional experimental programme or qualification programme, as mentioned above.

Authorization to begin changing FAs in the OSIRIS reactor was obtained from the Safety Authority at the end of 1994, after its analysis of the safety report that presented all results from the studies and tests.

The strategy adopted for the changeover to silicide fuel was to introduce an increasing number of silicide assemblies as reloads took place, replacing unloaded Caramel assemblies. 'Changeover' began in January 1995, when the first standard silicide assembly was loaded into OSIRIS, and was completed in April 1997, the date on which the first cycle took place with a core entirely made up of silicide assemblies.

II.2.1.2. Qualification programme conducted for the OSIRIS conversion

The data available for U_3Si_2 in 1989, when the decision was made to change from Caramel fuel to U_3Si_2, mainly consisted of the results of the qualification programme conducted by ANL in the context of the RERTR programme and, in France, of preliminary results obtained on full size plates.

The programme conducted by ANL involved a considerable number of irradiations, mainly on fuel with a density of 4.8 Mg U/m^3, some of which was pushed to very high burnups. In particular, six complete FEs made by three different manufacturers were irradiated in ORR to average burnups in the region of 80% for three of them. The excellent behaviour displayed under such irradiation resulted in the NRC validating U_3Si_2 in its NUREG-1313 report, but only for flux and temperature conditions that did not exceed those of the ORR irradiation (1.4 MW/m^2 for the upper limit of the heat flux and 130°C for the upper limit of the fuel meat temperature).

In France, the preliminary programme conducted in SILOE, on full size prototype plates, involved densities of between 2 and 5.4 Mg U/m^3. Irradiation was also carried out up to considerably high average plate burnups, between 75 and 78%. The results obtained made it possible to confirm the satisfactory behaviour of silicide fuel under conditions approaching those for normal OSIRIS operation. In particular, it should be noted that increases in plate thickness remained low; 55 μm at the end of irradiation for the most loaded plate (5.4 Mg U/m^3).

These results were deemed sufficiently convincing to launch the OSIRIS conversion programme. The chosen density was the NRC validation limit, or 4.8 Mg U/m^3. But in order to complete the qualification, it was considered vital to expand upon initial results by means of a programme of irradiation on components, plates, or whole assemblies representative of future industrial manufacture of silicide fuel for OSIRIS. The programme took place in two parts:

— The first was aimed at the qualification of U_3Si_2 fuel under normal operating conditions in the OSIRIS reactor;
— The second was concerned with establishing release rates in the event of cladding failure, with a view to answering a question raised by the Safety Authority regarding the ability of DRG chains to detect a possible cladding failure.

II.2.1.3. Qualification under normal OSIRIS operating conditions

This programme consisted partly of validating, for OSIRIS fuel, the irradiation-induced swelling law proposed for the U_3Si_2 4.8 Mg U/m^3 fuel in the NUREG-1313 (6% per 10^{27} fissions/m^3) and partly of qualifying the behaviour of that fuel in the form of complete assemblies in the OSIRIS reactor itself. Induced swelling laws were validated on U_3Si_2 4.8 Mg U/m^3 industrially representative plates irradiated in SILOE using the IRIS device, which provided the option of extracting plates at each end of the cycle and of following changes in thickness cycle by cycle, using an underwater test bench.

Irradiation, performed in situ in the OSIRIS reactor itself, was carried out on six lead test assemblies made by two industrial manufacturers (AREVA CERCA and B&W), which were subjected to an irradiation history representative of standard conditions. Irradiation was continued as far as average assembly burnups of 60% to 62%, providing an envelope of calculated burnups. The outcome of the irradiation (preservation of cladding integrity, limited increase in plate thickness under irradiation) made it possible to confirm the satisfactory behaviour of the U_3Si_2 fuel under conditions representative of normal OSIRIS operation.

II.2.1.4. Behaviour of defective plates (EPSILON programme)

In the context of the EPSILON programme, irradiation was carried out at the SILOE reactor on UAl_x and U_3Si_2 miniplates containing calibrated cladding defects (circular holes and grooves), with a view to simulating the behaviour of non-leaktight plates under stationary operating conditions. These tests were carried out using the facilities of a specific loop in the SILOE reactor, equipped in particular to measure and continuously purify fission products emitted in the coolant. Seven tests were carried out, two of which consisted of calibration on a leaktight miniplate. The results of the programme allowed for:

— Establishment of release rates of the U_3Si_2 fuel (R/B ratios are given in Ref. [36] for different surface power density values up to the hot spot calculated for the OSIRIS reactor (2.4 MW/m^2);
— Measurement of a possible change in the level of activity emitted in the course of a power stage;
— Comparison of levels of activity emission between U_3Si_2 fuel and UAl_x fuel under similar irradiation conditions;
— Checking of, by transposing the irradiation conditions to OSIRIS, the performance suitability of the loss of leaktightness-detection channels (DRG system based on measurement of delayed neutron emitters) to detect a cladding failure in a U_3Si_2 plate;
— Observation of the kinetics of fuel dissemination in a post-failure situation to see whether numbers remain low (less than approximately 1 mm^3 U_3Si_2 per hour), and to ensure easy detection by the DRG system response.

II.2.1.5. Additional development programme

The two programmes discussed above were supplemented by irradiations aimed at testing the sensitivity of reference specifications, namely:

— A loading of 4.8 Mg U/m^3;
— A meat thickness of 0.51 mm;
— The nature and extent of the admissible defects.

Irradiation was preceded by action taken in partnership with industrial manufacturer AREVA CERCA to develop the manufacturing process high density plates up to 5.8 Mg U/m^3, or plates with thick fuel meat up to 0.59 mm. Irradiation made it possible to confirm the validity of choices made for OSIRIS fuel, which feedback acquired since then has largely confirmed.

II.2.2. Qualification programme for Reacteur Jules Horowitz (RJH) U_3Si_2 fuel

II.2.2.1. Special characteristics of the RJH U_3Si_2 FA

The RJH U_3Si_2 FA differs considerably from the OSIRIS assembly. The OSIRIS assembly is flat and formed by a set of 22 parallel plates of standard thickness (1.27 mm) kept 2.46 mm apart from each other by swaging in grooves machined into the side plates. In the case of the RJH, the assembly is circular, formed by eight plate rings kept 1.95 mm apart by swaging into grooves machined into both faces of the stiffeners, which are placed 120° from each other. The thickness of the plates is 1.37 mm (0.61 mm for the fuel meat and twice 0.38 mm for the cladding).

Operating conditions for the RJH assembly are also considerably different than those for the OSIRIS assembly. For example, the power per assembly is 1.6 times greater and the cooling rate twice as high (15 m/s versus 7.5 m/s). The maximum heat flux of the RJH fuel is 4.0 MW/m^2.

In order to qualify all of these changes, a full qualification programme was launched. It consisted of three main parts:

— Validation of design;
— Qualification of the manufacturing process;
— Qualification of behaviour, especially under irradiation.

II.2.2.2. Validation of the design

Studies conducted on the assembly in the definition phase of the RJH project resulted in the final choice of fuel and preliminary design of the assembly. The development phase, currently in progress, should make it possible to validate the design, and in particular to show that the design:

— Meets objectives for technical and financial performance of the RJH;
— Meets design criteria, in particular in the case of structural elements, such as the stiffeners;
— Meets criteria associated with safety requirements.

These conclusions will have to be made based not only on neutron, thermal–hydraulic, and operational safety studies planned for the reference core, but also from studies of the mechanical strength of the FA itself. These studies will also be accompanied by bench tests or characterization in order to:

— Validate design codes used, and associated accuracy;
— Finalise assembly design (choice of design for end fittings, assembly head and end);
— Characterize as-manufactured assembly hydraulic behaviour (hydraulic pressure loss, vibration characteristics, stability under flow, etc.).

II.2.2.3. Qualification of the manufacturing process

Perfecting and qualifying the manufacturing process, work which has already been undertaken with AREVA CERCA, comprises:

— A set of preliminary actions, designed to validate constructs specific to the manufacture of RJH fuel;
  - Preliminary actions completed to date have made it possible to demonstrate the feasibility of chosen concepts;
  - At the end of this phase, a preliminary version of manufacturing specifications was drawn up and sent to the manufacturer;
— Perfecting manufacturing, including the introduction of necessary equipment, adjusting of manufacturing parameters, and demonstrating that the process has been mastered from an industrial point of view. This phase will lead to the issuing of a version of previous specifications which have been validated on an industrial scale;
— The manufacture of a pre-production range of assemblies, or LTAs, intended for the qualification of complete assemblies (neutron qualification, hydraulic characterization, qualification under irradiation);
  - A set of 12 complete assemblies, representative of industrial scale manufacture, will be produced with a view to performing qualification programmes on assemblies. Of the 12 lead test assemblies:
    - The first seven will be used in the CEA's EOLE neutronics mockup to form a block representative of an area of the RJH core;
    - The other five are to be used for qualification under irradiation in a specific loop of the BR2 reactor;
— Manufacture on an industrial scale;
  - After the initial production run, and provided the LTAs behave correctly under irradiation, manufacturing on an industrial scale will begin with a target of 75 to 80 assemblies per annum;
  - The objective is to deliver the first cores by mid-2013, when 'active' testing is planned under the current RJH schedule.

II.2.2.4. Qualification of fuel irradiation behaviour

(a) Preliminary test

In view of the characteristics of operating conditions, a preliminary test was carried out to validate the behaviour of representative U_3Si_2 plates under conditions similar to those in the RJH. Irradiation consisted of irradiating a mixed assembly in the BR2 reactor consisting of its five usual rings (UAl_x fuel enriched to 93% with AG3N cladding) and a final ring representative of RJH fuel (U_3Si_2 plates 0.61 mm thick with a density of 4.8 Mg U/m^3). The plates were tested during three cycles of 21 EFPD, with a maximum surface power density of 4.35 MW/m^2 but under more severe cooling conditions than those for RJH (pH = 6.5 and cooling rate = 12 m/s).

Examinations carried out after irradiation on samples taken in the most stressed areas made it possible to conclude that the silicide fuel behaved correctly under these experimental conditions.

The experiment made it possible to complete the first validation.

(b) Qualification under normal operating conditions

In order to complete the preceding demonstration under conditions fully representative of RJH conditions in terms of power and cooling rate, and insofar as there was no reactor or site directly available in which to conduct such experiments on a complete RJH assembly, it was decided to design and build a special loop in the central channel of the BR2 reactor. This device is currently under development. It should enable complete RJH assemblies to be tested (external diameter dimension 98 mm) under representative RJH conditions both in relation to power and cooling rate. The anticipated availability date for the device is September 2008. After that, a series of irradiation experiments is planned for lead test assemblies.

The tests will aim to gradually apply nominal RJH operating conditions (maximum surface power density of 4.0 MW/m^2, cooling rate of 15 m/s) and to achieve the target burnup fraction of 135 000 MWd/t U. The plan is to achieve these objectives in gradual stages, starting with an initial test of three BR2 cycles (3 × 22 EFPD) and ending with two tests of five cycles (5 × 22 EFPD), representative of the in-reactor lifetime of RJH assemblies. Criteria set for the experiments are those for normal operation, namely:

— Preservation of the mechanical integrity of the fuel plates;
  • Leaktightness can be confirmed without recourse to destructive testing;
  • Metallographic inspections will be carried out on the highest burnup areas in order to confirm the correct behaviour of uranium bearing particles;
— Preservation of geometric stability of the plates and more generally of the whole assembly by proving the absence of local instabilities on the most irradiated plates, such as blisters or pillows, and the absence for the whole assembly of global instability such as plate buckling or assembly twisting;
— The preservation of hydraulic channels and limited deformation of the plates, checked by means of three dimensional measurements, which will then be compared to similar pre-irradiation measurements.

II.2.2.5. Qualification under accident conditions

Thermal calculations will be used to show that requirements under accident conditions are met (no meltdown under category 3 situations and no violent thermal reaction under category 4 situations). As an example, the temperatures of hot spots on the cladding or in the fuel meat must be confirmed to have remained below melting temperatures of the cladding and aluminium matrix for each category 3 operating situation. These calculations will be performed under worst case conditions, of course.

II.3. QUALIFICATION OF U_3SI–AL DISPERSION FUEL FOR THE NRU AND MAPLE-TYPE REACTORS (CANADIAN PERSPECTIVE)

II.3.1. Introduction

AECL has designed, manufactured, tested, and qualified fuels in research reactors for over 50 years. The NRU and NRX reactors at Chalk River were initially fuelled using natural U metal fuel, but in the mid 1960s they were converted to HEU-Al alloy fuel. The FAs (called rods) were designed for on-power refuelling and consist of finned FEs contained within a flow tube with special end fittings to interface with the fuelling machine. The NRU FEs were extruded with straight axial fins and the NRX fuel with spiral fins. The finned element design proved to be very robust, and has been used successfully for decades.

In 1980, AECL began to develop LEU fuels [37] as part of the international effort to reduce enrichment in research and test reactors (RERTR). The initial focus was on dispersion fuels based on uranium silicides USiAl [38] and USi*Al [39] which had been developed and patented by AECL for use in CANDU power reactors; later U_3Si was added to the development programme. The original plan was to develop replacement LEU fuels for both the NRU and NRX reactors, but development of the higher density NRX fuel stopped when the reactor was placed in a standby mode of operation in 1986 (it was shutdown permanently in 1992).

When AECL began to design the MAPLE family of research and isotope production reactors (10–30 MW_{th}), the decision was made to base the LEU fuel design on the well-tested and highly successful NRU FE design, scaled appropriately for the smaller reactors. MAPLE fuel was qualified to meet requirements under Canadian Standards Association (CSA) standard N286.2, “Design Quality Assurance for Nuclear Power Plants.” MAPLE fuel qualification was based on a comprehensive programme that included irradiation testing of Al–U_3Si fuel and out of pile testing of prototype 18-element and 36-element MAPLE fuel bundles.

A modern production facility was built at Chalk River Laboratories and qualified to manufacture LEU fuel. This facility is licensed to manufacture Al–U_3Si dispersion fuel for the NRU, MAPLE 1 and 2, and HANARO reactors.

The development of Al–U_3Si fuel for research reactors is reviewed below to illustrate the qualification activities.

II.3.2. Overview of LEU fuel development for NRU

The NRU reactor is a large (125 MW_{th}) multipurpose reactor that is cooled and moderated with D_2O. The reactor vessel is 4 m diameter by 3 m high, and it contains 90 fuel sites, 30 isotope sites and 8 loops. This is a key facility for development of the CANDU reactor design and other Canadian nuclear technology. AECL and

NRC–Canada heavily utilize the reactor for fuels and materials testing, and for advanced materials research. NRU also produces about 60% of the world's radioisotope supply for medical and industrial applications (~34 000 treatments/day worldwide).

NRU was initially operated at higher power (220 MW) with natural uranium metal fuel but in the mid-1960s it converted to U–Al alloy FEs containing HEU. The finned U–Al FA design was more finely subdivided that the original fuel design and much development was focused on thermal–hydraulic assessments, manufacturing process development and the demonstration of adequate performance under irradiation. A facility was then installed at Chalk River to manufacture U–Al alloy fuel for the NRU and NRX reactors.

The robust NRU rod design and finned FE geometry were retained, but the challenge remained to develop a replacement fuel meat material containing approximately five times higher uranium density than HEU fuel to compensate for the lower U-235 enrichment of LEU. Metal matrix composites containing particles of USiAl, USi*Al and U_3Si dispersed in an Al matrix were selected for testing, based on considerable experience gained with uranium silicide fuels from earlier AECL programmes, to develop high density fuel for CANDU reactors [40 and 41]. USiAl contains 3.5 wt% Si and 1.5 wt% Al, USi*Al contains 3.2 wt% Si and 3.0 wt% Al, and U_3Si contains 3.96 wt% Si. Retention of NRU cladding material and element–cluster geometry obviated the need for corrosion testing and thermal–hydraulic reassessments.

The LEU fuel qualification irradiations included high power irradiations, high burnup irradiations, duty cycling irradiations, and irradiations of fuel with intentionally machined flaws to simulate defects. Test irradiations were also conducted to assess the effects of several manufacturing variables on fuel performance. By 1986, demonstration irradiations and post-irradiation examinations of full length prototype NRU rods containing USi*Al and U_3Si dispersion fuel had been successfully completed at Chalk River [42]. Based on its satisfactory performance, Al–U_3Si with a loading of 3.15 Mg U/m^3 was selected as the reference fuel for the NRU.

II.3.3. Mini-element irradiation of LEU silicide dispersion fuels

In eight different experiments[12] in the NRU reactor at CRL, over 100 mini-elements containing uranium silicide dispersion fuel have been successfully irradiated to burnups in the range of 21-93 atom % U-235 (86 mini-elements achieved an over 56 atom % burnup). A backup U–Al alloy containing 45% enriched U was also irradiated at the NRX. The experimental irradiations for the LEU fuel development programme are summarized in Table II.1. The mini-elements and fuel compositions tested, and reactor conditions, are described below.

II.3.3.1. Test element

Irradiation tests of silicide dispersion fuel have been performed using mini-element fuel meat with a diameter (5.5 mm) and cladding wall thickness (0.76 mm) the same as in full size NRU elements. Mini-elements are, however, only 184 mm long, compared to 2.9 m for NRU elements. The mini-elements are identical in cross-section to NRU FEs; they have six cooling fins at 60° intervals around the cladding, with a fin width of 0.76 mm and fin height of 0.96 mm.

II.3.3.2. Irradiation conditions

All of the LEU silicide dispersion fuels were irradiated at the NRU. The mini-elements were irradiated in a fuel carriage made from an aluminium cylinder with four holes bored axially through it at 90° intervals. The mini-elements were located centrally in the flow channels by four pronged anodized spiders located on the end plugs. A test assembly, containing up to 16 mini-elements, could be loaded in any normal fuel position at the NRU.

[12] These NRU experiments are designated Exp-FZZ-###, where ### is the serial number assigned (### is meant to show the generic form of the experiment designation).

TABLE II.1. AECL LEU FUEL IRRADIATION PROGRAMME SUMMARY

Experiment	Fuel Material[1]	Linear Power[2] (kW/m)	Burnup[3] (wt%)	Objective	Results
FZZ-905	Al-61.5wt% USiAl (8) Al-21 wt% U (4) Al-37 wt% U (4)	64–75	57	Compare LEU dispersions with 45 & 94% HEU alloys @ 3.15 Mg U/m^3	LEU fuel performance comparable to MEU and HEU (4.5 vol.% swelling)
FZZ-909A	Al-72.4wt% USiAl (6) Al-72.4 wt% USi*Al (6)	30–65	82	Test dispersion with high loads, 4.5 Mg U/m^3	Fuel swelling marginally above 1 vol.% per 10 at.% burnup
FZZ-909B	Al-61.5wt% USiAl (6) Al-72.4 wt% USi*Al (6)	20–80	93	Confirm high burnup performance @ 3.15 Mg U/m^3	Swelling linear with burnup < 1 vol.% per 10 at.% burnup
FZZ-910	Al-72.4wt% USiAl (8) Al-73.4 wt% USi*Al(8)	50–60 (86)	60	Test fuel with high loading of fine particles	Fines cause high swelling @ 4.5 Mg U/m^3
FZZ-911	Al-61.5wt% USiAl (4)	60–87 (106)	21–57	In-reactor corrosion	In reactor corrosion rates low
	Al-61.5wt% USiAl (8) Al-62.4wt% USi*Al (4)		93	Assess effect of meat imperfections	Surface defects cause no detrimental effects
FZZ-913[4]	Al-61.4 wt% U_3Si (48) Al-62.4wt% USi*Al (36)	30–50 (80)	78–87	First prototype full size LEU rod test in the NRU	USi*Al and U_3Si both satisfactory. U_3Si selected.
FZZ-915	Al-61.5wt% USiAl (2) Al-72.4wt% USiAl (2) Al-72.4 wt% USi*Al(2)	28–92	30–80	Assess in-reactor corrosion of pre-irradiated fuel	Pre-irradiation reduces corrosion rates of dispersion fuel
FZZ-918	Al-61.4 wt% U_3Si (16)	36–90 (112)	93	Establish swelling dependence on particle size distribution	Swelling increases with loading of fines, 6–7 vol.% @ 93 wt% burnup

[1] Number of mini-elements in brackets.
[2] Linear power output: Range (peak).
[3] Calculated burnup.
[4] Full length (3 m) 12-element NRU rods.

The mini-elements were irradiated at linear power outputs that extend beyond those experienced by a typical NRU fuel rod, which is normally limited to 40–80 kW/m. The nominal neutron flux density was ~1.1×10^{18} n/m^2·s, the heavy water coolant flow ~7 L/s, and the coolant velocity up to ~11 m/s. The coolant inlet temperature ranged from between 30 and 37°C and the coolant outlet temperature ranged from between 40 and 45°C during mini-element irradiations.

II.3.3.3. Mini-element test irradiation and PIE results

(a) Comparison of HEU–Al and LEU silicide dispersion fuel performance

In experiment Exp-FZZ-905, mini-elements containing A1-21wt% U alloy fuel (93% enriched U), Al-37 wt% U alloy fuel (45% enriched U) and Al-61.5 wt% USiAl silicide dispersion fuel (20% enriched U) were irradiated at NRU to compare behaviour of the fuel materials [43 and 44]. Fissile density was 0.63 gU-235/m^3 (the reference loading for the NRU). The mini-elements were irradiated at linear power outputs of between 64 and 75 kW/m to a final burnup of 57 atom % of U-235 destroyed. The calculated fuel meat and cladding surface maximum temperatures were 433 and 385 K respectively [44]. Interim and final visual examinations showed that elements developed a dull oxide layer on the cladding, but otherwise appeared the same as in as-manufactured condition.

Post-irradiation metallographic examinations revealed that dispersion fuel and U–Al alloy fuel diametrical increases (swelling) were similar. All element diametrical changes were less than 1.6% and length changes were within 0.2% after 57 atom % burnup. The measured increases agree with calculated volume changes based on post-irradiation immersion density measurements.

Swelling in the Al–21 wt% U meats was attributed to the build up of fission products, since no significant reaction was observed between the UAl_4 precipitates and the Al matrix in the alloy fuels, and no voids or fission gas bubbles were detected with the optical microscope. However, in the LEU fuel the silicide particles reacted with the aluminium matrix material, forming a thin interfacial layer, $UA1_3$ with dissolved Si, around the fuel particles, and fission gas bubbles ranging in size up to 5 μm were observed in the kernels of the fuel particles after 57 wt% burnup. Considerably fewer observable fission gas bubbles were found in the interfacial layers. Fuel meat swelling ranged between 3.4 and 4.5 vol.% after 57 wt% burnup, and taking into consideration scatter in the data, it was concluded that all three fuel materials swell at roughly the same rate with burnup [43].

(b) High burnup confirmation

In the Exp-FZZ-909B irradiation, mini-elements containing USiAl and USi*Al (3.15 Mg U/m^3) were irradiated up to 93 wt% burnup [44, 45]. The elements were non-destructively examined at various intervals, either under water or by neutron radiography. Definitive swelling measurements were made on the mini-elements by chemically stripping the oxide off the cladding [44]. This removed the uncertainty about oxide composition from immersion density calculations. PIE showed that both dispersions behaved similarly. The meats were swollen by 5.9–7.6 vol.% after 82 atom% burnup and by 6.6–7.8 vol.% after 93 atom% burnup [45]. These are important results because they show that swelling remained approximately linear up to 93 atom% burnup, and confirmed that the NRU (and MAPLE) composition silicide dispersion fuels exceeded NRU design terminal burnups of 80 atom% without exceeding the threshold of breakaway swelling observed elsewhere.

Post-irradiation metallography revealed features similar to those previously observed in experiment Exp-FZZ-905: interfacial layers formed around the silicide particles in the fuel meats and fission gas bubbles were observed within the particles. The interfacial layers were thinner near the fuel meat periphery and their edges were more sharply defined than at the fuel meat centre. The fission gas bubbles were about the same diameter (max. 5 μm) in both locations, but thicker interfacial layers were observed near the fuel meat centre. There was evidence that silicide particles had coalesced during irradiation, and more coalescence occurred at the fuel meat centre than at the periphery [45].

(c) Dispersions with high U loading and fine particles

In Exp-FZZ-909A, dispersions containing USiAl and USi*Al were tested at the higher 4.5 Mg U/m^3 loading required for NRX [44]. The mini-elements were irradiated to a terminal burnup of 82 atom%. Post-irradiation immersion density measurements taken after the oxide layer was chemically stripped from the aluminium cladding showed that the meats had swollen by 7.8–9.3 vol.%, most of which took the form of diameter increases. NRX composition fuel meat swelling was marginally over the envelope of 1 vol.% per 10 wt% burnup previously observed for NRU composition fuel meats, which had swollen by 5.9–7.6 vol.% after 82 atom% burnup. However, when results are normalized using the ratio of NRU and NRX fuel uranium

densities (i.e., 3.15/4.50) the values are comparable, indicating that the two fuel materials behave similarly [45]. Post-irradiation metallography revealed features similar to those described in the preceding section. Interfacial layers around reacted silicide particles reached a thickness of 7.5 μm near the fuel periphery and 25 μm near the centre. There was widespread evidence that silicide particles had also coalesced.

The subsequent experiment, Exp-FZZ-910, was designed to test the effect of fine silicide powder distribution on the swelling rate at high silicide loadings [45]. Mini-elements containing USiAl and USi*Al were irradiated to a terminal burnup of 62 atom% in NRU. The mini-elements contained predominantly fine silicide powders. The results from interim and final PIE showed that the finer silicide powders led to greater amounts of swelling in the Exp-FZZ-910 test compared to the Exp-FZZ-909A test on material of the same overall composition, but in which fuel particles were coarser. In both experiments the linear powers were in the range 50–60 kW/m. Detailed metallographic examinations of the Exp-FZZ-910 mini-elements, which swelled by 6.8 vol.% after 43 wt% burnup, revealed that the silicide particles had reacted extensively with the aluminium matrix and coalesced in about the centre two-thirds of the meat. The matrix in the central region was no longer aluminium but reacted silicide, likely UAl_3 with dissolved Si. The thermal conductivity of the transformed central meat would have been reduced to 63 W/m·K on transformation to UAl_3, compared with 124 W/m·K for Al–73.4 wt% USi*A1. It has been calculated that the meat temperature would have risen from 450 K to about 500 K, and this would have accelerated thermal reactions and increased the diffusion rates of fission gases [45]. The results of Exp-FZZ-910 indicate that silicide particle size distributions need to be closely controlled to ensure good performance at high U loadings equivalent to that required for NRX.

(d) Defective fuel behaviour: In-reactor corrosion

In-reactor corrosion behaviour of uranium silicide dispersion fuels has been investigated in Exp-FZZ-911 and the Exp-FZZ-915 irradiations [45 and 46]. The mini-elements for both tests contained USiAl and USi*Al. Exp-FZZ-911 was performed in two parts; the first part is described here and the second in the next section. In the first part of the test, four mini-elements had 1.2 mm diameter holes drilled into the cladding midsection and were irradiated in NRU at linear powers in the range of 60–87 kW/m. The first two mini-elements were removed from the reactor after reaching 21 and 39 atom% burnup (29 and 62 full power days, respectively), and the third and fourth element after reaching 57 atom% burnup (98 full power days).

Post-irradiation metallography and neutron radiography revealed that shallow ellipsoidal cavities had developed in the fuel under holes in the cladding, and cavity size increased with increasing burnup. The cavities in the mini-elements correspond to 1.1 mg and 3.2 mg of U-235 lost to coolant after 21 and 39 atom% burnup, respectively. After 57 atom% burnup, one mini-element lost 9.8 mg while the other lost 48.0 mg U-235. Metallography revealed that the mini-element with the greatest fuel loss had developed a crevice between the fuel meat and cladding, and thus exposed a larger surface area of the fuel meat to hot coolant than the companion mini-element, in which the meat was well bonded to the cladding. The results show that to a first approximation, fuel loss was proportional to surface area exposed [46]. The test indicates that the corrosion rate of the purposely defected FEs is sufficiently low to prevent sudden catastrophic failures.

The Exp-FZZ-915 experiment was similar to the Exp-FZZ-911 experiment, the major difference being that the mini-elements had been previously irradiated to a variety of burnups in the range of 22.3 to 82.6 atom% before the 1.2 mm diameter holes were drilled in the clad midsections [46].

These elements were further irradiated in the NRU reactor, to evaluate performance of the defected FEs, and during the additional 38 days in the reactor no increase in coolant activity above the normal background was detected. Neutron radiography and metallographic examinations revealed that the cavities were typically 0.7 mm deep by 1.3 mm across; i.e., only marginally larger than the original cavity made by the drill tip.

These results are very encouraging, indicating that the corrosion resistance of LEU fuel is possibly increased by previous burnup, probably because irradiation or thermal effects improved fuel to cladding bonding cohesion, and a corrosion resistant UAl_3 interfacial layer formed around fuel particles.

(e) Assessment of manufacturing variable

(i) Effect of silicide particle size distribution on core swelling

The original objective of the second part of experiment Exp-FZZ-911 was to evaluate the performance of intact mini-elements having slight as-manufactured or deliberately introduced imperfections in the meat surface [42, 46 and 47]. Twelve mini-elements were tested: four contained defects that occurred during fuel meat extrusion (cold shuts and axial gouges up to 150 μm deep), four contained axial grooves up to 0.2 mm deep by 2 mm wide, and four had circumferential grooves up to 0.4 mm deep by 2 mm wide machined in the surface to simulate large defects. The mini-elements contained Al–61.5 wt% USiAl; however, the fuel contained the same fine silicide particle size distribution which had caused enhanced swelling in Exp-FZZ-910. Therefore, examinations were also carried out to determine the effects of fine particle size on meat swelling when loading is 3.15 Mg U/m^3 compared to 4.5 Mg U/m^3 in Exp-FZZ-910. The mini-elements were initially irradiated at a linear power output of 106 kW/m. The power output decreased as burnup increased, to 25 kW/m at 93 atom% burnup.

The mini-elements were removed for interim and final visual examinations, immersion density measurements and metallography. PIE revealed that the aluminium cladding had flowed into and filled surface defects in fuel meats during extrusion, effectively increasing local cladding thickness. More importantly, immersion density measurements show that the meat volume of Exp-FZZ-911 mini-elements had only increased by approximately 4.9 vol.% after 60 atom% burnup, compared with 7.0 to 17.5 vol.% swelling in the Exp-FZZ-910 meats at about the same burnup. The mini-elements with axial and circumferential grooves machined in the meat surface showed volume increases of 7.3-7.4 vol.% after 93 atom% burnup, while the mini-elements with as-manufactured surface defects up to 150 μm deep showed about 8.3 vol.% swelling after 93 atom% burnup. Clearly, surface defects had no detrimental effect on fuel performance. The fuel removed from the machined grooves effectively lowered local fissile loading, and this, combined with the additional restraint provided by thicker cladding (over the grooves), reduced overall swelling of the mini-elements with large simulated surface defects. The results indicate that swelling performance is acceptable even when fine particles size distributions are used with lower silicide loading (3.15 Mg U/m^3) as required for the NRU.

(i) Fuel core surface imperfections

In Exp-FZZ-918, 16 mini-elements containing Al–61.4 wt% uranium silicide were irradiated at NRU to help establish limits on particle size distribution to be used in fuel manufacturing specifications [47]. The 16 mini-elements were divided into four groups, each group containing different particle size distributions. Group 1 contained the highest fraction of fines while Groups 2, 3 and 4 had progressively lower fractions of fines.

It was originally intended that the fuel would contain only U_3Si particles (compared to USiAl and USi*Al used in previous mini-element tests). A broad spectrum of size distributions ranging from predominantly coarse to predominantly fine was desirable so that optimum size distribution could be selected. However, the new comminution process at CRL produced U_3Si powders with only a small fraction having less than 44 μm, which is highly desirable for normal production but resulted in too few fines for the experiment. Further reduction of coarse fractions (88–150 μm) into fines proved difficult; only medium sizes fines (44–88 μm) broke up readily. Since there was a large supply of suitably sized USiAl powders from previous experiments, these were included to make up the difference in fines.

The mini-elements were installed in the NRU in January 1986, and irradiated under normal driver fuel coolant conditions, but at power ratings more severe than that of a typical full length fuel rod. Four mini-elements, one from each group, were removed for interim examinations after approximately 40, 60, and 80 wt% burnup. The remaining four pins were removed in June 1987 after reaching 93 wt% burnup. Immersion density measurements indicated that fuel swelling was proportional to the percentage of fines contained in the mini-elements and, since the percentage of particles in the medium range was held constant, inversely proportional to the fraction of coarse particles.

PIE revealed no evidence of gross linking up of bubbles, indicating that the fuel was well away from the onset of breakaway swelling. Indeed, swelling remained linear up to a very high burnup of 93 wt%. For all of the particle size distributions tested, swelling remained well below the 1 vol.% per 10 wt% envelope previously

established. Results confirmed the silicide particle size distribution required to minimize swelling at the 3.15 Mg U/m^3 loading for NRU.

II.3.4. Out of pile tests

Several out of pile studies were performed to support the fuel qualification programme, including:

— Fuel alloy density measurements;
— Microstructural studies;
— Corrosion tests;
— Thermal compatibility studies;
— Thermal conductivity measurements;
— Bend tests of irradiated mini-elements;
— Fission gas release (FGR) studies. FGR tests are highlighted below as an example.

II.3.4.1. Fission gas release from thermal ramp heating tests

Thermal ramping tests were conducted in hot cells to determine the effects of temperature excursions on dimensional stability and fission product activity release from previously irradiated silicide dispersion fuel [46]. Whole mini-elements and short segments of mini-elements with the fuel meat exposed were chosen, having fuel burnups of either 23 or 93 atom%. Half the samples contained A1–USiAl and half contained Al–USi*A1 with 3.15 Mg U/m^3. Test conditions were as follows:

— Argon gas flow rate: 1.66×10^{-6} m^3/s;
— Heating rate: 0.2 and 0.4°C/s;
— Temperature: 530° to 720°C;
— Holding time: temperature held constant (±4°C) for an hour.

In the thermal ramp tests, a whole mini-element irradiated to 93 atom% burnup developed small localized blisters, some with pinhole cracks releasing fission products (^{85}Kr and ^{137}CS) after 0.25 h at 530°C. This behaviour prevented gross pillowing or ballooning. A mini-element irradiated to 93 atom% burnup and ramped to 640°C developed radial cracks, which tore the cladding and released fission products from the meat. Recorded gamma activity indicated ^{85}Kr was released at 580°C during the temperature rise, peaking (2.9×10^8 Bq) just before reaching the 640°C plateau, and once again (7.6×10^6 Bq) during cooling. Krypton release patterns from the mini-elements were similar to those obtained from short (cut) fuel specimens which had been irradiated to 23 atom% burnup and heated to 720°C, notwithstanding test temperature differences. Even at high test temperatures (530 and 640°C), the mini-element maintained considerable structural integrity. This behaviour is interpreted to mean that coolant channels would not become blocked even if the fuel were subjected to some hypothetical abnormal event with the potential to cause overheating of the meat (e.g. 530°C, compared to the normal operating maximum of 200°C).

II.3.5. NRU full size fuel element test results

The satisfactory performance of mini-elements containing USiAl and USi*Al dispersion fuels led to the fabrication and testing of full size, 12-element NRU fuel bundles (rods) [42, 45–47]. Seven rods (designated FL-001 to FL-007) were irradiated at NRU: three containing Al–62.4 wt% USi*Al and four containing Al–6 1 wt% U_3Si. LEU rods were installed in NRU in 1984, replacing the reference HEU–Al alloy fuel, and were irradiated under typical fuel operating conditions [42]. Electrical conductivity of the coolant ranged from between 25 and 70 μmho/m during irradiation. The pH was not routinely measured, but it typically ranged from between 5.5 and 7.0. The coolant inlet temperature ranged from between 30 and 37°C, while the outlet temperature was between 60 and 70°C; inlet pressure was ~552 kPa (80 psig) and outlet pressure was ~207 kPa (30 psig).

Each NRU rod typically occupies five or six core positions during its lifetime (~340 d residence time at 70% efficiency) as it is moved from a low flux site at the outside of the core to a high flux site in the centre, then back to a low flux site at the outside. The NRU rods thus experience an increasing and then decreasing power history. During irradiation of the first seven prototype LEU rods at the NRU, average element linear power ranged from 30–50 kW/m, with the maximum reaching approximately 80 kW/m [42].

After irradiation, the fuel rods were transferred to underwater cooling bays, where the fuel bundles were removed from the flow tubes. Visual examinations showed that the FEs were in good condition, almost identical in appearance to HEU elements, with the normal aluminium oxide layer coating the surface. Before irradiation, selected elements had special V notches made 274 cm apart in the fins to act as benchmarks for length measurements after irradiation. Underwater measurements indicated that length changes after irradiation were small.

Destructive post-irradiation examinations in hot cells confirmed that the performance of both USi*A1 and U_3Si dispersion fuels with the 3.15 Mg U/m^3 loading required for the research reactors at CRL was acceptable up to high burnups beyond the NRU design burnup of 80 atom%. Thinner interfacial layers around the U_3Si seem to indicate more stable and marginally better performance than for USi*Al. However, comparisons of high burnup performance may be compromised somewhat by the fact that the USi*Al particles were finer and therefore provided a larger surface area for a reaction to take place with the aluminium matrix. The U_3Si has an additional advantage in that its properties make it easier to manufacture powders within a desired size range and it has a higher uranium density, which simplifies fabrication in comparison to USi*Al. Based on its satisfactory performance, Al-U_3Si was selected as the reference LEU fuel for the NRU.

II.3.6. Lead test assemblies and conversion of the NRU

A full scale Al–U_3Si fuel campaign was undertaken to manufacture the equivalent of half a NRU core to qualify the manufacturing process and to provide lead assemblies for irradiation under prototypical handling and refuelling conditions. LEU rods were substituted as HEU rods were discharged until the NRU reactor was partially converted. Approximately 50 sites were fuelled with prototype LEU fuel rods. This demonstrated on a large scale that LEU fuel was acceptable for use in research reactors and that the transition from HEU to LEU would not be problematic. It also served to qualify the Al–U_3Si fuel manufacturing process.

Successful fuel qualification irradiation, PIE, and supporting physics assessments formed the basis for the safety case and subsequent approval by Canadian regulators to reduce fuel enrichment in AECL research reactors. The NRU reactor was converted from HEU to LEU fuel between 1991 and 1993.

II.3.7. LEU fuel development for MAPLE reactors

The MAPLE class of research reactors was designed by AECL for fuels and materials testing, isotope production, and neutron beam experiments. The MAPLE concept features an H_2O-cooled and D_2O-reflected core. The design offers high thermal neutron flux, small core volume, and space for multiple beam tubes and in-core experimental sites, as well as spectrum specific facilities in the reflector (cold neutron source, fast neutron flux trap, etc.).

When AECL began to design the MAPLE reactors, a decision was made to base the LEU fuel design on the well tested and highly successful NRU FE design, scaled appropriately for the smaller reactors. Two bundle designs were developed for MAPLE 1 and 2, an 18-element cylindrical bundle and a 36-element hexagonal bundle with a 60 cm fuel length. MAPLE fuel met requirements under the Canadian Standards Association (CSA) standard N286.2, Design Quality Assurance for Nuclear Power Plants.

MAPLE fuel qualification was based on a comprehensive programme that included:

— Irradiation testing of the Al–U_3Si fuel under the LEU silicide fuel development programme;
— Heat transfer measurements, onset of nucleate boiling (ONB), onset of steam vapour (OSV) and critical heat flux (CHF) measurements;
— Hydraulic measurements of the bundles, including vibration measurements using accelerometers;

— Endurance testing of the bundles in a full scale hydraulic test rig (about eight months of flow testing showed no evidence of fuel fretting);
— Hydraulic load analysis, encompassing loads from seismic events.

The LEU silicide fuel is made in a modern fuel manufacturing facility designed, constructed, and commissioned at Chalk River Laboratories between 1987 and 1990. The facility is licensed to manufacture Al–U_3Si dispersion fuel for the NRU, MAPLE 1 and 2, and HANARO reactors.

II.4. DEVELOPMENT OF TRIGA® ($UZRH_X$) FUEL (GENERAL ATOMICS, USA)

II.4.1. Fuel development, qualification and licensing

The information contained in this section has been extracted, in some cases verbatim, from Refs [48, 49 and 50].

In 1976, the General Atomic Company (renamed General Atomics in 1986 and referred to hereinafter as GA), located in San Diego, California, recognized the need to replace the 8.5 wt% uranium content FLIP fuel (70% enriched) used in its 38.1 mm diameter fuel rods and the 10 wt% uranium content 93% enriched fuel used in its 12.7 mm diameter fuel rods with a higher density fuel containing <20% enriched uranium to make the fuel more readily exportable from the USA. In early 1977, it embarked on a programme in conjunction with the US Department of Energy, to develop, or at least prove, the capability to increase uranium content in $UZrH_x$ fuel to at least 45 wt% (3.7 Mg U/m^3). GA was both the fuel developer and the commercial fuel manufacturer.

Research and development was carried out for both 30 wt% and 45 wt% alloys; 20 wt% fuel had been previously developed during the US SNAP Reactor programme. In 1977, a number of out of pile studies were performed, leading to the following results:

— Extensive metallography and electron microprobe analysis showed that the fuel microstructure was very similar for all uranium loadings;
— Thermal expansion and thermal conductivity measurements showed little change in these values;
— Fission product release measurements showed little change;
— A quench test from 1200°C showed the same behaviour as that of existing fuels;
— A thermal cycling test conducted through the uranium phase change temperature of near 315°C showed that the continuous zirconium matrix continued to stabilize the fuel.

Manufacturing studies at the GA manufacturing plant showed that the existing process could be used with no modification; in fact, manufacturing became somewhat easier because hydriding was simpler. Full size fuel rods were manufactured for irradiation testing, including:

— FLIP prototypes containing either 20 wt% or 45 wt% uranium fuel with erbium burnable poison clad in Incoloy-800;
— 1.27 mm diameter prototypes containing 20 wt%, 30 wt% or 45 wt% uranium fuel with erbium burnable poison, also clad in Incoloy-800.

A two part irradiation test programme was planned:

— Tests in GA's TRIGA Mark F reactor began in 1978 on FLIP prototype rods. Thermal cycling from ambient temperatures showed no apparent detrimental effect after more than 3000 cycles. In addition these rods were pulsed a number of times to peak reactor powers of over 2000 MW (over 1300 times steady state power) and at fuel temperatures of over 600°C during the thermal cycling test programme;
— High burnup tests began at ORNL in 1979 as part of the US RERTR programme. Irradiations began at the ORR in December 1979, using 12.7 mm prototype rods contained in a 16-rod test assembly; 19 fuel rods were irradiated. Subsequent PIEs were performed at the High Radiation Level Examination Laboratory (HRLEL) at ORNL, including:

- Irradiation of the 20 and 30 wt% rods ended in May 1982 after reaching average burnups in the 45–57% range, well in excess of the goal burnups of 35 and 40%, respectively;
- Metallographic examinations of one section each from the highest burnup 20 and 30 wt% rods and one intermediate burnup with 45 wt% rods were performed during the summer of 1983. The examinations showed normal performance with no anomalies;
- Irradiation of 45 wt% rods ended in August 1984 after reaching average burnups in the 60–66% range, well in excess of the goal burnup of 50%. The failure of two 45 wt% rods occurred during irradiation, both of which were subsequently shown to have resulted from cladding breaches — one from an original flaw in the cladding material and the other from mechanical fretting of the cladding against a grid spacer in the test assembly following a rod configuration change. The former failure led GA to change its manufacturing quality control procedure to require 100% ultrasonic testing of all cladding used for fuel rods for higher power reactors with forced flow cooling;
- Comprehensive PIEs were performed on the 45 wt% rods beginning in 1985. The results of the PIEs demonstrated that the fuel had behaved as expected and that no limiting values had been reached. The PIEs included:
  - Gamma scanning of the highest burnup rod to determine peak to average burnup;
  - Diametral change and bowing measurements on all of the 45 wt% rods;
  - Radiochemical burnup analysis from the peak burnup region of the highest burnup rod;
  - Measurement of pressure and identification of gases in the plenum of the highest burnup rod;
  - Physical testing of cladding segments from the highest burnup rod;
  - Metallography of sections from the cladding of the highest burnup rod.

Based upon the results of the tests at GA and at ORNL, GA applied to the USNRC for a license amendment allowing the 20 and 30 wt% fuels to be considered standard fuels for the Mark F reactor and to be approved generically for use in all licensed TRIGA reactors in the USA. The license application was supported [48–50]. Other references, including [51], contain many results of the tests performed by GA.

After reviewing GA's license application and request for generic approval, the USNRC issued NUREG-1282 in August 1987, approving GA's request [52]. The generic approval of the two fuels for use in USNRC licensed TRIGA reactors carried the proviso that "case-by-case analyses discuss individual reactor operating conditions in applications for authorization to use them".

II.4.2. New manufacturer qualification

TRIGA-type fuel was designed and built exclusively by GA for more than forty years beginning in the 1950s. In 1994 and 1995 a 50%–50% joint venture was formed between GA and AREVA CERCA that transferred manufacturing of the TRIGA fuel to the AREVA CERCA facility in Romans, France. The new joint venture, called TRIGA International, is the only facility currently operating that manufactures TRIGA fuel.

The fabrication of TRIGA fuel requires special skills and know-how gained by GA during the 40 years it was sole manufacturer. Therefore transferring the technology to AREVA CERCA required special training of AREVA CERCA technicians before closure of the GA workshop, and then full integration of GA and AREVA CERCA technicians during the transfer and startup of fabrication at the AREVA CERCA facility. This integration was carried out over a period of several years, which assured that by the end of the period the technology transfer for the manufacture of fuel meats and FEs was complete. Quality of the fuel is assured through AREVA CERCA's International Organization for Standardization (ISO) and American Society of Mechanical Engineers (ASME) NQA-1 certification. AREVA CERCA is now fully qualified to perform the fabrication of all types of TRIGA fuel, using U–ZrH and U–ZrErH alloys of up to 45 wt% uranium with 20% U-235 enrichment.

II.5. DEVELOPMENT OF THE IRT-4M FUEL ASSEMBLY (RUSSIAN PERSPECTIVE)

The development and launching into manufacture of the IRT-4M FA with UO_2-based fuel (19.7% U-235 enrichment) is shown here to demonstrate that the system of development and launching into manufacture of

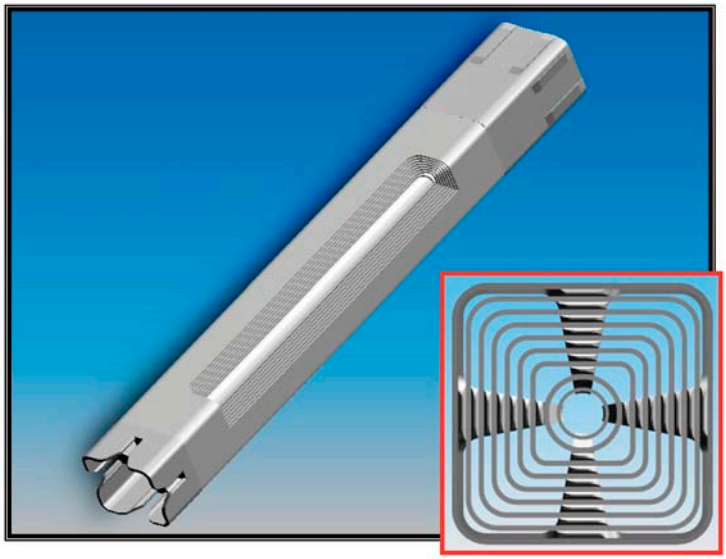

FIG. II.3 . IRT-4M FA (8-tube FA).

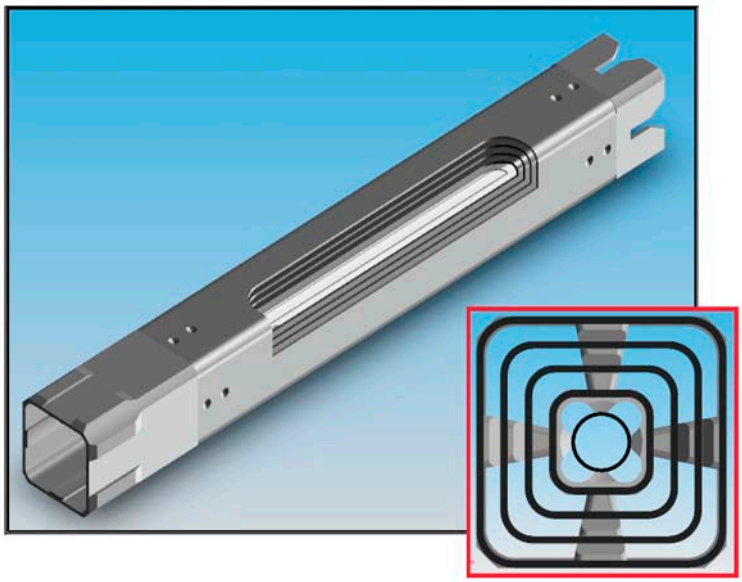

FIG. II.4. IRT-2M FA (4-tube FA).

FAs for research reactors is in force in the Russian Federation. IRT-4M FAs are designed for operation in research reactors in the Czech Republic, Bulgaria, Uzbekistan, and the Libyan Arab Jamahiriya. A general view of the IRT-4M FA is shown in Fig. II.3.

The design of the IRT-4M FA is similar to the design of the IRT-2M FA with UO_2-based fuel (36% U-235 enrichment), shown in Figure II.4, and to the design of the IRT-3M FA with UO_2-based fuel (36% and 90% U-235 enrichment). Fuel based on UO_2 dispersed in an Al matrix has been widely used in the FEs of research reactors for many years. Therefore, during IRT-4M FA development, the 'Research Work' stage was not carried out.

In the Russian Federation, IRT-4M FA development started with development of the TA–DW in 1994. JSC 'TVEL' was the customer who ordered the work performance, FGUP 'NIKIET' was the development centre, and JSC 'NCCP' was the manufacturer.

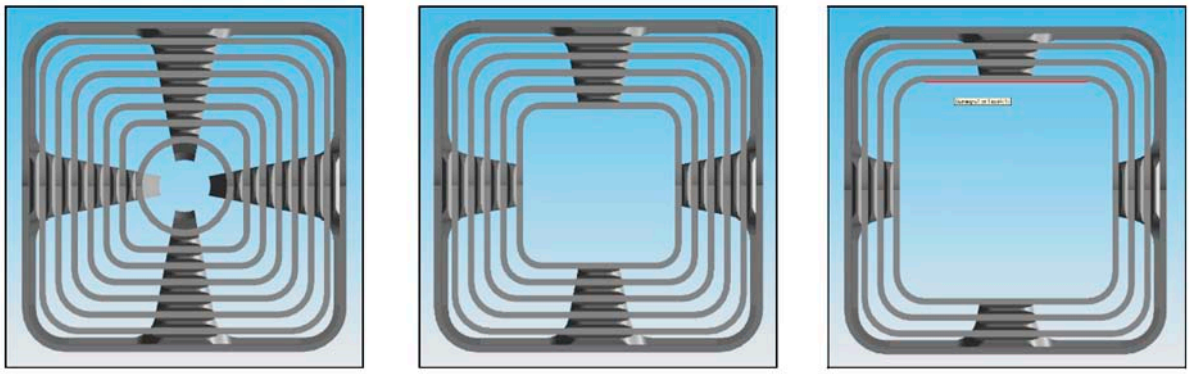

FIG. II.5. Cross-sections of IRT-4M FA modifications: a) 8-tube FA, b) 6-tube FA, c) 4-tube FA.

- For the above mentioned reasons, the stages of technical offer and conceptual design were also not carried out. These stages are not obligatory and are undertaken only if required.

In 1998, at a stage of detailed design according to customer agreement and based on TA–DW requirements, the manufacturer worked out design documentation on experimental prototypes for three modifications of the IRT-4M (8-tube FA, 6-tube FA, and 4-tube FA). Cross-sections of these FA modifications are shown in Fig. II.5.

The manufacturer also developed technological documentation on FE and component manufacturing methods and on FA assembly. During the following two years the manufacturer was involved in designing and manufacturing machining attachments used for the manufacture of FE and components, as well as in the development of manufacturing methods for eight types of IRT-4M FEs.

In 1999, according to an agreement with the customer, the manufacturer made four experimental IRT-4M FAs (two 8-tube FAs and two 6-tube FAs) which differed in their FE cladding material. The FAs were delivered to the Institute of Nuclear Physics (Tashkent, Uzbekistan) for in pile (life) tests in the VVR-SM reactor.

At the end of 2000, in pile tests of the FAs at INP started in the VVR-SM reactor. The reactor core consisted of 16 FAs (12 IRT-3M FAs with 36% U-235 enrichment and four IRT-4M FAs with 19.7% U-235 enrichment). In pile tests of IRT-4M FAs were carried out in accordance with the document "Programme and procedures of in pile (life) tests of experimental IRT-4M FAs in VVR-SM reactor (The Institute of Nuclear Physics, Uzbekistan)" issued by the development centre.

Results of the in pile tests were positive. In experimental IRT-4M FAs, average burnup was more than 60%. Based on in pile test results, the development centre prepared a technical report.

Because there was no possibility for post-irradiation examination of the FEs at the Institute of Nuclear Physics (INP) in Uzbekistan, the customer decided to manufacture additional FEs with UO_2+Al fuel composition for post-irradiation examination in The Russian Federation in order to substantiate the operational capability of the fuel composition. In 2000, according to an agreement with the customer, the manufacturer manufactured two 'combined' IVV-2M FAs with hexagonal cross-sections. The FAs had two experimental FEs with UO_2–Al fuel composition and 3.0 MgU/m^3 uranium concentration. The FAs were installed in the IVV-2M reactor (Zarechniy, the Russian Federation) for irradiation. In pile tests were carried out in accordance with the document "Programme and procedures of in pile service tests in IVV-2M reactor for 'combined' IVV-2M FAs", issued by the development centre.

The results of in pile tests of the 'combined" IVV-2M FAs were positive. In the two experimental FEs with UO_2+Al fuel composition (3.0 Mg U/m^3 uranium concentration) in the two 'combined' FAs, average burnup was 40% and 60%, respectively. One FA containing experimental FEs with 60% burnup was subjected to post-irradiation examination. The examination outcome confirmed the positive results of FA in pile tests.

Technical reports were prepared based on in pile test and post-irradiation examination results. Technical reports were included in the explanatory note of the detailed design for the IRT-4M FA.

In 2004, according to customer agreement, the development centre issued the detailed design of IRT-4M FAs for research reactors LWR-15 (Czech Republic) and VVR-SM (Uzbekistan). The detailed design was agreed to by the Surveillance Authority, Rostekhnadzor.

The detailed design consisted of the following documents:

— Detailed design sheet;
— Draft of FA and FE specifications;
— General view of the FA;
— Explanatory note, which in turn consisted of:
  • Neutron physics, thermal–hydraulic design and stress calculation of reactor core;
  • Report on the results of bench and in pile tests of an FA experimental prototype (mock-up);
  • Report on the results of post-irradiation examination of an FA experimental prototype (mock-up);
  • Report on patent research.

Based on the developed detailed design, the manufacturer prepared design documentation on the developmental prototypes of three IRT-4M FA modifications (8-tube FA, 6-tube FA, and 4-tube FA) and technological documentation on developmental FA manufacture. The manufacturer also tooled up for production, prepared FE and FA quality control procedures, and equipped the production complex with FE quality control units.

To perform a preliminary conformity assessment of the IRT-4M FA development batch with TA–DW requirements to assess design and manufacturing quality, and determine the possibility of presenting developmental FAs for acceptance tests, a preliminary test was carried out in 2004. This was done in accordance with the programme and procedures of preliminary tests developed by the manufacturer for an IRT-4M FA development batch (2 FAs). Test results were acknowledged as successful. The FAs were approved by the Acceptance Committee. Based on test results, a report confirming fulfillment of the test programme was drawn up. The FA development batch was acknowledged as meeting TA–DW requirements. Recommendations regarding presentation of the manufactured FAs for acceptance tests were given. Based on preliminary test results, the letter 'O' was assigned to the design documentation (DD) and the technical documentation (TD).

At the end of 2004, the customer appointed an acceptance committee for IRT-4M FA acceptance tests. Representatives of the chief designer of the reactor core, the designer–technologist of the FEs, the manufacturing plant Rostekhnadzor, and the research manager of the reactor core were appointed to be members of the acceptance committee. The development centre issued the programme and procedure of acceptance tests and presented the following documents to the acceptance committee for review:

— TA–DW "Development and manufacture of IRT-4M FAs with UO_2-based fuel (19.7% U-235 enrichment)";
— Programme and procedure of acceptance tests;
— Approved detailed design for IRT-4M FAs.

The manufacturer presented the following to the acceptance committee:

— Two developmental IRT-4M FAs which had passed the preliminary test;
— Report and protocols of preliminary tests for two developmental IRT-4M FAs;
— The DD and the TD on manufacturing developmental IRT-4M FAs;
— FE and FA specifications.

Based on acceptance test results, the acceptance committee acknowledged that two developmental IRT-4M FAs met TA–DW requirements. Based on test results, a report confirming the fulfilment of the test programme was drawn up. The report was signed by the representative of the Russian Surveillance Authority, Rostekhnadzor. Based on the acceptance test results, the letter 'O_1' was assigned to the DD and TD.

Signing of the IRT-4M FA acceptance test report containing recommendations regarding the letter 'O_1' assignment to the DD and TD is considered to be the completion of FA launching into manufacture (FA manufacture licensing). Manufacture licensing denotes the manufacturer's readiness to supply IRT-4M FAs to customers according to contracts made.

Appendix III

ACRONYMS

Activities, components, descriptive terms

BOL	beginning of life
CD	conceptual design
CHF	critical heat flux
DD	design documentation
DRG	détection de rupture de gaine (loss of cladding integrity)
DW	research work
EDX	energy dispersive X-ray
EFPD	effective full power days
EOL	end of life
EMP	electron microprobe
FA	fuel assembly
FE	fuel element
FLIP	fuel lifetime improvement programme
HEU	highly enriched uranium
LEU	low enriched uranium
LTA	lead test assembly
NQA	nuclear quality assurance
ONB	onset of nucleate boiling
OSV	onset of steam vapour
PIE	post-irradiation examination
R&D	research and development
R/B	fission product atoms released/total atoms burned
RR	research reactor
RW	research work
SEM	scanning electron microscope
TA–RW	technical assignment for research work
TECDOC	technical document
TD	technical documentation
TEM	transmission electron microscope
TO	technical offer
TM–DW	technical assignment for DW
WDD	working design documentation
WDX	wavelength dispersive x-ray

Research laboratories or institutes, companies, organizations, reactor names

AECL	Atomic Energy Canada Limited, Canada
ANL-E	Argonne National Laboratory-East (Illinois), United States of America
ANL-W	Argonne National Laboratory-West (Idaho); now part of INL
ANSTO	Australian Nuclear Science and Technology Organisation
AREVA/CERCA	Compagnie pour l'Etude et la Réalisation de Combustibles Atomiques, France
ASME	American Society of Mechanical Engineers
ATR	Advanced Test Reactor (Idaho)
BR2	Belgian Reactor-2, CEN/SCK Centre, Mol, Belgium
B&W	Babcock & Wilcox
CEA	Commissariat a l'Ènergie Atomique (France)
CANDU®	CANada Deuterium Uranium

CNEA	Comisión Nacional de Energia Atómica
FRM-II	Forschungsreaktor München II (German for Research Reactor Munich II)
GA	General Atomics
HANARO	High Flux Advanced Neutron Application Reactor, South Korea
HFIR	High Flux Isotope Reactor (United States of America)
HFR	High Flux Reactor
HRLEL	High Radiation Level Examination Laboratory, ORNL
INL	Idaho National Laboratory
JHR	Jules Horowitz Reactor (or RJH, RR in construction), CEA–Cadarache, France
JMTR	Japanese Material Test Reactor
KAERI	Korea Atomic Energy Research Institute, Republic of Korea
OPAL	Open Pool Australian Light-water reactor
ORNL	Oak Ridge National Laboratory
ORR	Oak Ridge Research Reactor
OSIRIS	French MTR-type reactor at CEA-Saclay Centre
MAPLE	Multipurpose Applied Physics Lattice Experiment (Canadian Research Reactor)
MIR	Research Reactor, Dimitrovgrad, Russian Federation
NRC	Nuclear Regulatory Commission, USA
NRU	National Research Universal Reactor, Canada
NRX	National Research Experimental Reactor, Canada
NUKEM	German Research Reactor Fuel Fabricator
NUREG	U.S. nuclear regulation of the NRC
PNPI	Petersburg Nuclear Physics Institute, Gatchina, Russian Federation
RERTR	Reduced Enrichment for Research and Test Reactor
RHF	Reactor High Flux, Grenoble, France
RIAR	Research Institute of Atomic Reactors, Dimitrovgrad, Russian Federation
RJH	see JHR
TRIGA®	Training, Research, and Isotope Reactors, General Atomics
VNIINM	A.A. Bochvar All-Russia Research and Development Institute of Inorganic Materials

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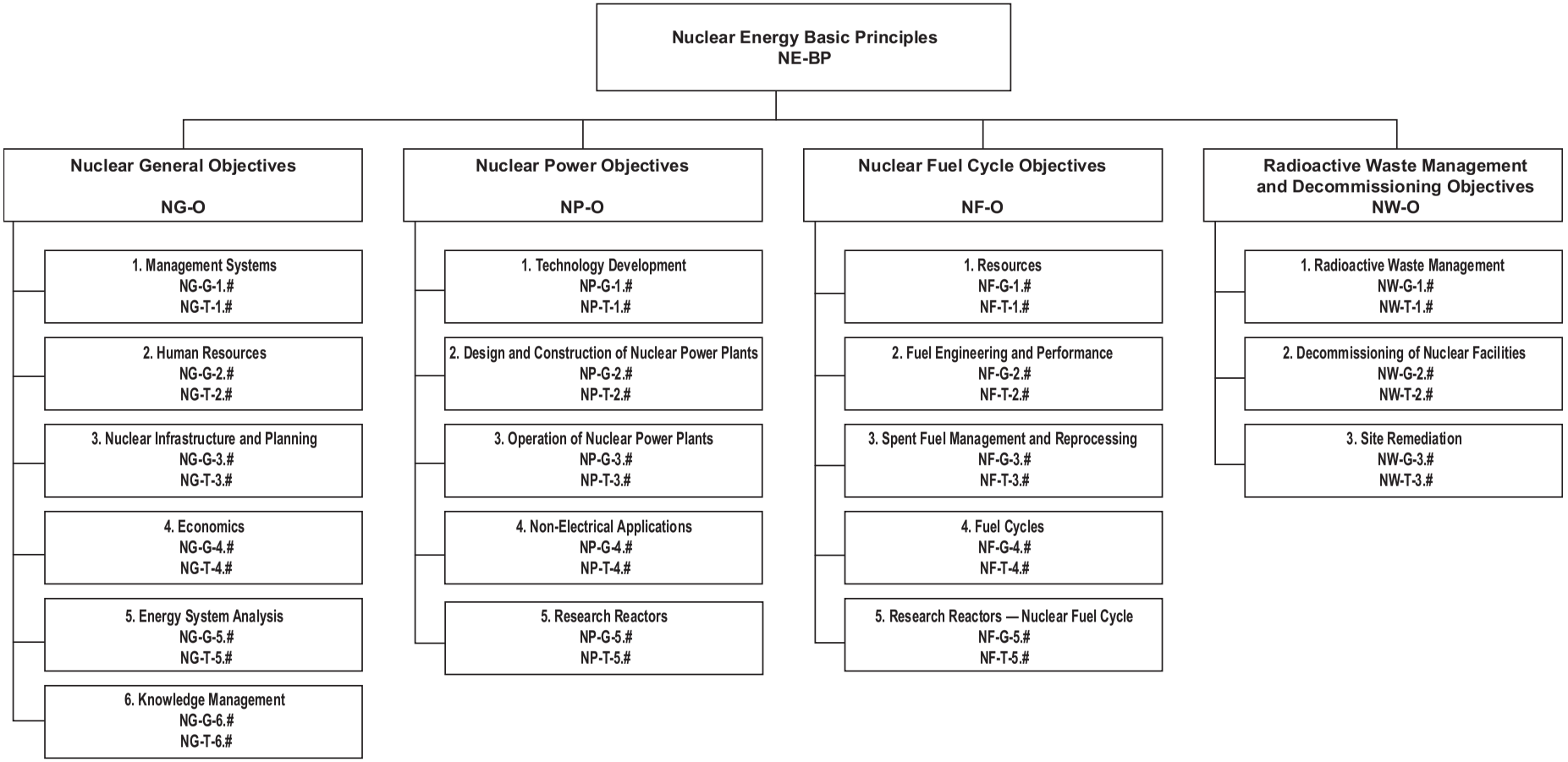

Key

BP: Basic Principles
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