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SUBJECT: DRAFT REPORT ASSESSMENT OF THE ADEQUACY OF METALLIC FUEL QUALIFICATION TO SUPPORT THE LICENSING OF SMALL MODULAR SODIUM-COOLED FAST SPECTRUM REACTORS (4-S, ARC-100, PRISM)

In 2010, the NRC's Advanced Reactor Program (ARP) in the Office of New Reactors contracted with Brookhaven National Laboratory (BNL) to provide support to preparation activities for future licensing reviews of various advanced reactor designs. At that time, the NRC was preparing for reviews of various advanced reactor designs, including but are not necessarily limited to integral pressurized-water reactors (iPWRs), high-temperature gas-cooled reactors (HTGRs), and sodium-cooled fast reactors (SFRs). A task within the BNL statement of work included assessing the available literature and data related to qualification tests and experiments of the metal fuel designs proposed for SFRs and preparing a report on the history of fuel qualification for SFR metal fuel designs. The staff also requested that the report include an assessment of the testing performed at facilities such as EBR-II in regards to the qualification of these fuel designs.

The NRC received the attached draft final report from BNL but changing conditions related to potential licensing applications for SFRs decreased the interest in and resources available for completing the report. The NRC staff did not review the report in sufficient detail to support completing the activity and BNL did not formally issue the report. However, revived interest in SFRs and other non-light water reactor designs means that the information collected and assessments performed as part of the 2010 ARP may be useful – even though this information should be viewed as draft or preliminary. The NRC staff takes no position on the data collection, assessments, findings, or recommendations contained in the report. In addition, information related to the specific designs included in the draft paper, even if correct at the time, may have changed in the years after the report was prepared.

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ASSESSMENT OF THE ADEQUACY OF METALLIC FUEL QUALIFICATION TO SUPPORT THE LICENSING OF SMALL MODULAR SODIUM-COOLED FAST SPECTRUM REACTORS (4-S, ARC-100, PRISM)

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ACRONYMS

ALFUS	Alloyed Fuel Unified Simulator
AOO	Anticipated Operational Occurrences
COL	Combined Licenses
CRBR	Clinch River Breeder Reactor
DC	Design Certification
DFR	Dounreay Fast Reactor
EBR	Experimental Breeder Reactors
ECCS	Emergency Core Cooling System
FCCI	Fuel Clad Chemical Interaction
FCMI	Fuel Clad Mechanical Interaction
FFTF	Fast Flux Test Facility
GDC	General Design Criteria
GNEP	Global Nuclear Energy Partnership
GRSIS	Gas Release and Swelling in Isotropic fuel matrix
IFR	Integral Fast Reactor
IRACS	Intermediate Reactor Auxiliary Cooling System
ITAAC	Inspection, Test, Analyses, and Acceptance Criteria
KAERI	Korea Atomic Energy Institute
LEU	Low-Enriched Uranium
LOCA	Loss of Cooling Accident
LOF	Loss of Flow
LWR	Light-Water Reactors
MACSIS	Metal fuel performance Analysis code for Simulating the In-reactor behavior under Steady-state conditions
NASA	National Aeronautics and Space Agency
NEAMS	Nuclear Energy Advanced Modeling and Simulation
NRC	U.S. Nuclear Regulatory Commission
ORT	Operational Readiness Test
RVACS	Reactor Vessel Auxiliary Cooling System
SFR	Sodium-cooled Fast-Spectrum Reactor
SMR	Small Modular Reactor
SRP	Standard Review Plan
TOP	Transient Overpower
TREAT	Transient Reactor Test Facility
TRL	Technical Readiness Level

1.0 INTRODUCTION

The emergence of Small Modular Reactor (SMR) concepts as potential sources of nuclear energy has resulted in the design of a variety of SMR systems. Included in this selection of designs are three sodium-cooled fast-spectrum reactors (SFR): (1) the 4S design being proposed by Toshiba [1]; (2) the PRISM design proposed by General Electric-Hitachi [2]; and (3) the ARC concept proposed by Advanced Reactor Concepts LLC [3]. These concepts have many features in common; in particular, all three are based on utilization of metallic fuel. Metallic fuel generally contains both a fissile and non-fissile fuel and matrix components; for the above concepts, the reference fuel assumes low-enriched uranium (LEU) and zirconium, respectively. The relative weights of these components are based on satisfying criticality and operations and safety considerations subject to fuel fabrication and performance constraints. For all three concepts, the metallic fuel alloy is assumed to contain approximately 10 wt% zirconium, with the remainder, LEU.

In all cases, the metal fuel is bonded to the inner clad surface by sodium, and the clad material is generally a ferritic stainless steel, such as HT9, although a number of alternate cladding materials have been investigated historically and more advanced "modern materials", e.g., oxide-dispersion strengthened (ODS) steels may be attractive/suitable options. The ability of both the fuel and clad material to accommodate the material damage due to the high energy neutron flux is an important factor in the selection of the cladding material. Furthermore, the fuel-clad combination must be able to survive the design basis events without release of any fission products. Finally, the behavior of the fuel-clad combination during a beyond design basis event must minimize the release of fission products, and minimize the consequences of the event.

The use of metallic-based fuel in sodium-cooled fast-spectrum reactors dates back to the first reactors constructed in the US (i.e., Experimental Breeder Reactor (EBR)-I, EBR-II, Fermi-1, etc.). The SMR concepts currently being proposed use fuel that is based on the experience gained in these early reactors. Indeed, the proponents of all three concepts attest that the available data base effectively qualifies metallic fuel for the fuels-related issues of the licensing process, including operations and safety. The objective of the present study is to review the available knowledge/experience base related to metallic fuels and assess its applicability and its suitability to support the licensing process for the 4S, PRISM, and ARC-100 sodium fast reactor concepts.

Recognizing that there has effectively been extremely limited work on metallic fuel in the US for many years, especially in the context of implementation in a commercial reactor, these objectives present a considerable challenge. Several key issues need to be addressed, including: (1) the current capability to fabricate the fuel and cladding on a commercial basis; (2) how closely do the characteristics of the fuels that were tested match those in the proposed concepts (e.g., fabrication details such as microstructure, cladding, etc.); and (3) how closely do the environmental conditions under which the "historic" testing (especially irradiations) was performed (temperature, flux level/power density and spectrum, etc.) match the conditions relevant to the proposed concepts. The situation is further complicated by the fact that currently a regulatory framework does not exist for either metallic fuels or sodium-cooled fast-spectrum reactors and is exacerbated by the absence domestically of experimental facilities with the

requisite "environmental" capabilities and especially irradiation conditions. Therefore, the option of requiring a confirmatory series of experiments at desired prototypic conditions as a precursor to submitting a licensing package entails significant practical and economic hurdles (e.g., utilizing the limited facilities available internationally). Therefore, from a "practical" perspective, there are two possible scenarios: (1) the existing knowledge/experimental data is adequate to provide the requisite level of confidence to U.S. Nuclear Regulatory Commission (NRC) either "as is" or with "modest" additional work, that the proposed reactors can be built and operated safely, or (2) "significant" additional work is required to support a licensing decision.

The structural framework for performing the present assessment was based on consideration of:

- what is a logical/reasonable way to address the technical maturity of a completely new fuel and its implementation? And

- what information will NRC likely require from each licensee to support the licensing process. In these cases the situation is complicated by the fact that a new reactor AND a new fuel are involved, and both are significantly different from current commercial light-water reactors (LWRs).

The first bullet is well aligned with the concept of "Technical Readiness Level (TRL)," which was originally developed by the National Aeronautics and Space Agency (NASA) and subsequently modified and adopted/adapted by the DOE-NE Advanced Fuel Cycle Initiative and Global Nuclear Energy Partnership (GNEP) programs to assess the technical maturity of components of advance fuel cycles, including fuel [4]. The TRL level corresponding to the existing fuels experience establishes the current baseline relative to what is desirable for a deployed system.

Addressing the second issue is more complicated. At a minimum, the NRC would be expected to require information that would be considered in the Standard Review Plan, suitably modified for the metallic fuel-SFR combination. Since the operational and safety performance of an SFR is especially closely linked to the fuel form, documentation would likely include topical report(s), as well as the various iterations of the Safety Analysis Report.

The TRL approach and potential NRC requirements are discussed in Section 2. The key characteristic parameters of the proposed fuels are described in Section 3. The experimental database and the status of computational models for predicting fuel performance are described in Section 4 and 5, respectively. The discussion will include both "historic" data as well as more recent relevant work performed under the DOE-NE Fuel Cycle Technology program and its predecessors, including the Nuclear Energy Advanced Modeling and Simulation (NEAMS) efforts. Section 6 will describe the results of the assessment, addressing the applicability and adequacy of the state-of-knowledge on metallic fuels to support the licensing of the three proposed concepts.

2.0 CHARACTERISTICS OF PROPOSED METALLIC FUELS

The key fuel parameters and operating conditions for the 4S, PRISM, and ARC-100 SMRs are summarized in Table 2.1. It is important to recognize that the data for the 4S and ARC-100 are obtained primarily from the presentations made at the kick-off meeting for this task held at the NRC on August 26, 2010. The General Electric Hitachi presentation at the meeting did not include any details on the fuel, and to complicate matters further, the data supplied to NRC in support of this review was very high-level, with minimal details. Based on the presentation from that meeting, the "reference" design is 840 MWt (311 MWe), with several fuel options, the most relevant for the present study being a U-20Pu-10Zr metallic fuel. During the course of the presentation, GEH stated that they would begin operations with a U-10Zr fuel, eventually ***potentially*** transitioning to fuel containing plutonium, and possibly other transuranics. As a result, several of the parameters for the PRISM concept in Table 2.1 are based on information submitted by GEH in connection with the GNEP and may not be totally consistent with the PRISM-SMR that is being proposed, even though it was the only additional data provided by GEH to support the current evaluation.

Based on the available information, the following observations relevant to the fuel, and the reactor thermal and nuclear environment can be made:

- the temperature regimes for all concepts are similar

- the 4S concept is significantly different in several respects:
  - longer active fuel length and overall fuel rod length
  - an axial power distribution that varies with burnup/life due to reflector control of the reactivity (cf. Figure 2.1) [1]
  - grid spacers instead of the more traditional wirewrap
  - lower burnup (resulting in a smaller plenum)
  - long-life core
  - higher smear fuel density
  - larger diameter fuel rod and fuel slug
  - thicker clad

- The ARC-100 clad fast fluence is higher than for the other concepts and exceeds the "conventional/accepted" limit for HT-9 clad of $< 4 \times 10^{23}$ n/cm^2.

Table 2.1 Characteristics of Proposed Metallic Fuels

Key Parameter	4S	Standard/ S-PRISM	ARC-100
Peak Burnup, 10^4MWd/t	< 5.5	18	14.3
Max. linear power, kW/m	8		25.5
Cladding hotspot temp., ^oC	609	572	556
Peak center line temp., ^oC	<630		686
Peak radial fuel temp. difference, ^oC	< 30		104
Cladding fast fluence, n/cm^2	2×10^{23}	3×10^{23}	5.06×10^{23}
Cladding outer diameter, mm	14	7.366/7.442	13
Cladding thickness, mm	1.1	0.559/0.559	
Fuel slug diameter, mm	10.4	5.41/5.476	
Fuel length, m	2.5	1.19/1.02	1.5
Plenum/fuel length ratio	1.08	1.48/1.88	
Fuel residence time, years	30	5	
Smeared density, %	78	75/75	
Nominal Fuel Composition (BOL)	U-10Zr	U-26Pu-10Zr	U-10Zr
Uranium Enrichment	17% - 19%		10.1% - 17.2%
Clad Material	HT-9	HT-9/HT-9	HT-9(?)
Vol. Frac.			
Fuel		28.22/28.3	42.9
Sodium Bond		9.41/9.43	14.3
Structural		37.31/36.57	17.7
Coolant		25.06/25.7	25.6

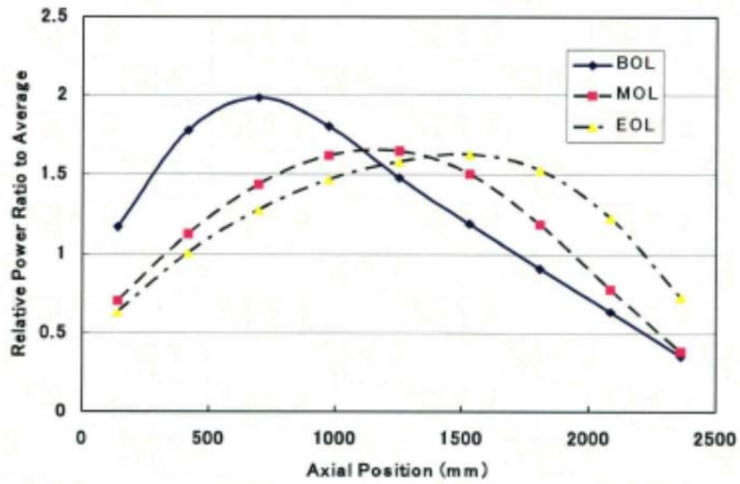

Figure 2.1 – Axial Power Profile of 4S Fuel at Different Stages of Operation. (BOL, MOL, and EOL are beginning, middle, and end of life, respectively) [1]

3.0 APPROACHES TO FUEL QUALIFICATION

3.1 Qualification Objectives

A fuel design that is proposed for use in a reactor developed to the implementation stage must typically provide the following:

1. A specification for the fuel, including chemical composition, geometry, and related design aspects. The latter includes cladding and assembly hardware.

2. A database of fuel properties and irradiation behavior to support an assessment of the ability of the fuel to satisfy operations, safety and reliability objectives.

Achieving the first of these objectives requires a fuel design that can be manufactured by a fuel vendor. The second objective requires convincing evidence that the fuel design is such that operations, safety and reliability performance can be demonstrated with a sufficient/appropriate degree of assurance to regulatory agencies, and the operational risks are acceptable to a reactor operator.

Two approaches to demonstrating that a new fuel is qualified for operation in a reactor can be envisaged. In the first, the fuel performance data is compared to NRC requirements and the degree to which the data addresses those requirements allows a judgment of its adequacy and provides guidance on what additional work may be necessary.

Since the current NRC requirements for a fuel that is significantly different from that for current LWRs are of necessity not specific, an alternative approach is to consider how totally new fuel has been historically qualified for operation. This view was taken in the GNEP program, which adopted an approach based on the concept of Technical Readiness Level originally developed by NASA, and modified it for the development of the new fuels envisioned for use the program. Since metallic fuels in a SFR were one of the GNEP options, the proposed approach has relevance to the current review, with appropriate recognition/modification that the GNEP fuels would in some cases be significant extensions in terms of the compositions of the fuels used in previous SFRs whereas the compositions considered in the SFR concepts considered in the present review are basically the same as those employed in earlier US SFRs (EBR, Fast Flux Test Facility (FFTF), Fermi).

Both of these approaches are discussed in the following sections.

3.1.1 Current NRC Fuel Qualification Requirements

The NRC guidance/procedure for performing safety reviews for a reactor is outlined is outlined in the Standard Review Plan (SRP) and given in NUREG-0800, specifically Section 4.2 for Fuel System Design [5 and Appendix A]. This review provides assurances that the fuel is not damaged as a result of normal operation and anticipated operational occurrences (AOO). In addition, this review will assure that if the fuel system is damaged, it is not so severe as to prevent control rod insertion, that the number of failed fuel rods is not underestimated in the case

of postulated accidents, and that coolability is always maintained. The specific areas to be reviewed according to the SRP are:

1. Design Bases. In the design bases, the safety analysis will address fuel system damage mechanisms and limiting values for important parameters to prevent damage from exceeding acceptable values. In addition, established designs will be evaluated, together with their associated specific acceptable fuel design limits, to determine if both are valid for the new fuel type.

2. Description and Design Drawings.

3. Design Evaluation. In this step, the fuel system performance during normal operation, AOOs, and postulated accidents is determined to ensure that the design bases are met. In addition, operating experience, direct experimental comparisons, detailed analyses (fuel performance codes), and any other pertinent information will be reviewed at this stage.

4. Testing, Inspection, and Surveillance Plans. This review covers all the activity in the steps from new fuel fabrication to installation in the reactor and finally to post-irradiation examination.

5. Inspection, Test, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the applicant's proposed ITAAC associated with the structures, systems, and components as they relate to the fuel system will be reviewed.

6. COL Action Items and Certification Requirements and Restrictions. For a DC application, the review will address COL action items and requirements and restrictions (e.g., interface requirements, etc.).

Furthermore, SRP Section 4.2 outlines the review of the fuel system and the associated damage criteria. However, there are related reviews that depend on the outcome of this review. These include the radioactive inventory in the fuel pins, nuclear design of fuel assemblies, fuel/clad interaction, control rod drive mechanism, and cooling under emergency conditions.

The SRP Section 4.2 also addresses the following which are of particular relevance to the current review:

- Operating Experience
- Prototype Testing
- Analytical Predictions
- Post Irradiation Surveillance

In these sections, it is recognized that significantly different new fuels present challenges with respect to having the desired level of confidence/assurance with respect to safe operations. The guide notes that:

"No definitive requirements have been developed regarding those design features that must be tested before irradiation…"

However, several out-of-reactor tests are recommended to "serve as a guide to the reviewer." In addition,

"When prototype testing cannot be performed, a special detailed surveillance program should be planned for the first irradiation of a new design."

Additional relevant areas are highlighted in Appendix A.

The current acceptance criteria are LWR-centric and will need to be modified to reflect the specific characteristics of SFRs and their fuel. The acceptance criteria are outlined in various 10 CFR chapters, including 50.46, 50.67, 100, and 52.47(b). In addition, general design criteria (GDC) 10, 27 and 35 apply to the acceptance criteria of fuel system designs.

Briefly, the current acceptance criteria outlined in the fuel system SRP are given below:

1. 10CFR50.46, 10CFR50.34, and 10CFR50.67, as they apply to cooling performance of the ECCS.

2. 10CFR Part 100 and 10CFR50.67, as they apply to determining acceptability of a reactor site based on exposure to an individual as a result of fission product release following a major accident scenario.

3. GDC-10, as it relates to assuring that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including AOOs.

4. GDC-27, as it relates to reactivity control system being designed with sufficient margin and in conjunction with Emergency Core Cooling System (ECCS) being capable of controlling reactivity and core cooling under post-accident conditions.

5. GDC-35, as it relates to providing an ECCS to transfer heat from the core following a loss of cooling accident (LOCA) at a rate such that fuel and clad damage that could interfere with continued effective core cooling is prevented, and clad metal-water reactions are limited to negligible amounts.

6. 10CFR52.47(b)(1), which requires that a DC application contain the proposed ITAAC.

7. 10CFR52.80(a), which requires that a COL application contain the proposed inspection, tests, and analyses.

The interface reviews and acceptance criteria are LWR centric and concentrate on specific accidents sequences (i.e., LOCA, etc.), and systems (i.e., ECCS, poison injection system, etc.) that do not apply to SFRs. The criteria should be modified to reflect the new accident sequences possible with an SFR, and the different safety systems that are included in its design. These will

determine the transient response of the reactor core and thus affect the power and temperature transients that the fuel will be subjected to.

The proposed SFR concepts that are the subject of this review are comparatively low pressure (compared to conventional LWRs) designs, using a "pool" type concept for the primary system. Thus, the traditional LWR LOCA event is not credible in these designs. Furthermore, these designs do not have an ECCS as used in LWRs. Instead, in the case of a primary system coolant leak the coolant flows into a guard vessel and decay heat is removed from the primary system by means of the reactor vessel auxiliary cooling system (RVACS). In addition, there is an intermediate reactor auxiliary cooling system (IRACS) that can perform the same heat removal task if the intermediate loop is available.

In view of the differences outlined above, criteria that are specific to ECCS operation would be revised or eliminated entirely in the case of an SFR fuel system review. This comment applies to acceptance criteria 1, 4, and 5. GDC-27 would need to be re-written recognizing that there must be sufficient negative reactivity, due to thermal expansion and distortion of a SFR core to assure cold shutdown with the highest worth rod stuck out. GDC-10 can be carried over without any changes, and GDC-35 probably does not apply to a SFR system as currently envisioned.

Modifications to the above criteria that were made during the Clinch River Breeder Reactor (CRBR) review [6] are instructive and should be included to the extent that they still apply. It should be noted that the CRBR was a "loop"-type design, rather than the "pool"-type design being proposed in the current designs. Any re-write of the criteria should thus be generic enough to cover the characteristics of both types of designs. The specific changes made in the CRBR review, as they relate to the fuel system are:

1. Inspection of the residual heat removal system. This discussion is covered in GDC-36 and GDC-37. However, they are very specific to LWR systems; thus, it will be necessary to remove all reference to ECCS, water, and requires a revised list of important components. Recognition of the RVACS and IRACS decay heat removal systems is necessary, since they will control the temperature transient that the fuel will be subjected to.

2. Protection against fuel rod failure propagation. In GDC-10, it is important to recognize that on-line fuel failure monitoring and post-irradiation examination of fuel are necessary to detect anomalies or confirm expected fuel system performance.

3. Protection against coolant flow blockage. A GDC to specifically eliminate the possibility of fuel assembly flow blockage is required since the fuel assemblies are ducted and inlet flow blockage (Fermi-1 event) is possible.

3.2 Technical Readiness Levels

In this section, the TRL scale, as developed by NASA to judge the technical maturity of space related technologies and modified for gauging the degree of development of reactor fuel [4], will be outlined. An adaptation of this methodology as applied to fuel development is described in

Table 3.1. As can be seen, there are nine TRL levels. In the current application, four fuel development phases are outlined on the right hand side of the table.

The following subsections elaborate on the general requirements of the four phases. Subsequent sections describe the technology status of metallic fuels according to these phases.

Table 3.1 Proposed Application Of Technical Readiness Levels To Reactor Fuel Development And Qualification

<table>
<tr><th>TRL</th><th>TRL Function</th><th>Generic Definition</th><th>Fuel Development-Specific Definition</th><th>Fuel Dev. Phase</th></tr>
<tr><td>1</td><td rowspan="3">Technology Down-Selection</td><td>Basic principles observed and formulated</td><td>Technical review leading to identified technical options. Identification of criteria for candidate selection</td><td rowspan="2">1</td></tr>
<tr><td>2</td><td>Technology concepts and/or applications formulated</td><td>Fuel candidates selected from options, based on selection criteria</td></tr>
<tr><td>3</td><td>Analytical and experimental demonstration of critical function and/or proof of concept</td><td>Calculational analysis and lab-scale experimentation and characterization addressing feasibility, including: fabrication process development, property measurement, and ex-pile tests</td><td rowspan="2">2</td></tr>
<tr><td>4</td><td rowspan="2">Final Process Selections and integration</td><td>Component and/or bench-scale validation in a laboratory environment</td><td>Establish proof of concept. Fabrication of irradiation testing samples in accordance with QA requirements. Design parameters and features established Performance phenomena identified with proof-of-concept irradiation testing</td></tr>
<tr><td>5</td><td>Component and/or breadboard validation in a relevant environment</td><td>Irradiation testing of prototypic rods/compacts under nominal representative conditions (e.g., fission densities, fuel and cladding temperatures, cladding damage rates) is performed and assessed</td><td rowspan="2">3</td></tr>
<tr><td>6</td><td rowspan="2">Full-scale integrated testing</td><td>System/subsystem model or prototype demonstration in relevant environment</td><td>Prototypic rod/compact and assembly/element irradiation in representative environment, under full range of relevant normal and off-normal conditions. Representative compositions Design parameters investigated Information is sufficient to support a Fuel Specification and a Fuel Safety Case (which, in turn, support larger System Demonstration to achieve TRL7)</td></tr>
<tr><td>7</td><td>System prototype demonstration in prototypic environment</td><td>Fabrication of reference fuel derived from production supply sources irradiated to design conditions and utilization Irradiation in representative environment Prototypic design. Prototypic fabrication processes. Representative compositions</td><td rowspan="2">4</td></tr>
<tr><td>8</td><td rowspan="2">Full-scale demonstration</td><td colspan="2">Actual system completed and qualified through test and demonstration</td></tr>
<tr><td>9</td><td colspan="2">Actual system proven through successful mission operations</td><td></td></tr>
</table>

3.2.1 Phase 1 – Fuel Candidate Selection

The objective of this phase is to identify fuel types and concepts with the potential to satisfy the reactor operating requirements. In general, a successful fuel must be amenable to fabrication, have acceptable thermo-mechanical, thermo-physical, and physio-chemical properties, have acceptable in-service performance, and must be compatible with an acceptable disposal or re-cycling technology. Criteria for selection include the following:

- Ability of proposed fuel form to accommodate desired fuel compositions,
- Experience with similar fuel types,
- Suitability of established fabrication techniques,
- Anticipated performance capabilities (temperature, burnup, etc.),
- Anticipated safety-related behavior,
- Suitability to design considerations, such as fuel-clad and fuel-coolant compatibility,
- Compatibility with back-end fuel cycle technology, and
- Cost of fabrication.

Achieving Phase 1 goals corresponds to a TRL 2, as described in the above table

3.2.2 Phase 2 – Concept Definition and Feasibility

The next phase in fuel development and qualification requires the establishment of a reference fuel concept. Research and development efforts are thus directed at determining the viability of the selected fuel forms with regard to fabricability, acceptable properties, and whether there are problematic performance issues. The fabrication process development effort has the following objectives:

1. The process should be based on identifiable techniques, and produce samples for characterization and irradiation testing,

2. Evaluate process improvements through the application of innovative techniques, and perform conceptual designs of full-scale fabrication facilities.

Key physical properties to be measured and assessed in laboratory experiments include:

1. Thermo-physical, physical and mechanical properties such as thermal conductivity, heat capacity, density, and hardness,
2. Phase equilibria, such as liquidus, solidus, and dissociation temperatures,
3. Interdiffusion and compatibility of fuel constituents and fission products with cladding and coolant, and
4. Thermo-mechanical properties of cladding.

The initial irradiation testing will be carried out to differentiate between fuel concepts and to identify potential life limiting phenomena. These will be investigated under steady-state and transient conditions and include:

1. Fuel dimensional change due to swelling or irradiation growth,
2. Gas behavior in the fuel. Gas could be generated within the fuel as a result of (n,f), (n,α) and (n,p) reactions,
3. Fuel constituent migration and phase stability,
4. Interdiffusion and chemical interaction of fuel or fission products with cladding, and
5. Dimensional change of cladding due to exposure to high energy neutrons.

The laboratory-scale experiments outlined above are sufficient to achieve a TRL 3; a TRL 4 can be achieved by carrying out simple proof-of-concept performance testing.

3.2.3 Phase 3 – Fuel Design and Improvements Evaluation

The objective of this phase is to optimize the reference fuel design, prepare a fuel specification and a licensing and safety case, and develop a fuel performance code. In order to prepare a licensing and safety case, a fabrication process will need to be demonstrated. This activity should address the following points:

1. Develop a pilot scale process and parameters that meet the specific fabrication requirements,
2. Design and construct fabrication tools and equipment, and
3. Demonstrate repeatability of fuel fabrication within specification tolerances.

The thermo-physical and mechanical properties measurements mentioned above should be repeated on the as fabricated fuel.

Irradiation testing during this phase is a large effort and should include the following:

1. Provide performance data to inform the design improvements effort,
2. Provide data to support the licensing and safety case,
3. Establish fuel lifetime limits,
4. Identify safety related behavior during off normal transient conditions, such as TOP, LOF etc.,
5. Determine the behavior of the fuel to variations in the as fabricated fuel.

This effort will require many fuel samples, irradiated in a reactor with an appropriate (prototypic) neutron flux, and extensive post-irradiation examinations.

A predictive computer model of fuel behavior is necessary to determine the fuel behavior under conditions that have not been obtained experimentally. These codes are validated against experimentally obtained data and are used to support the safety case for the design. Specific objectives of the fuel modeling process are:

1. Demonstration of an understanding of the fuel fabrication process,
2. Accurate prediction of the fuel material properties as a function of burnup, and
3. A fuel performance code that can be used as a predictive tool.

Following successful irradiation of the reference fuel design a TRL 5 is achievable. This can be increased to TRL 6 with completion of a defendable safety case.

3.2.4 Phase 4 – Fuel Qualification and Demonstration

During the final phase of the fuel qualification, the following objectives will be pursued:

1. Demonstrate engineering scale fuel fabrication capability,
2. Qualify production line fuel for performance within bounds of licensing safety case,
3. Confirm fuel behavior under design basis events conditions,
4. Demonstrate safe and reliable performance of a core consisting of reference fuel elements, and
5. Validate predictive code.

The fuel qualification program entails the irradiation and post-irradiation examination of a set of lead fuel rods and/or assemblies exposed in a reactor with the appropriate neutron flux environment, coolant type, and geometric dimensions. The fuel used in these tests must be produced in accordance with the specifications determined in the previous phases. The irradiation testing will encompass selected design basis accident conditions, which are selected with the concurrence of the licensing authority. If the fuel behavior is within the bounds specified in the licensing safety case, then the fuel design can be considered to be qualified for use in that specific reactor design. Successful completion of these tests results in a TRL 7 if lead test assemblies are used. If the tests are carried out using entire cores, or substantial portions of a core, then a TRL 8 is possible. Finally, following several cycles of power reactor operation, which will result in a reduction in financial risk and an increase in reliability, will result in a TRL 9 rating.

Typically, to carry out the above sequence of experiments and analyses for a completely new fuel would take approximately 20 years. However, in the case of the present review, the basic proposed fuel form has been used in a fast reactor environment, and thus the qualification time should be substantially shorter. However, it must be demonstrated that the new fuel and clad sufficiently close to that for which the experience exists for the experience base to be applicable. The impact of the differences between the proposed fuel and that for which experimental data exists must be assessed in order to determine the need for, and scope of any additional out-of-core and/or irradiation data needed. The absence of a domestic ability to test fuel under prototypic SFR conditions presents significant challenges depending on the degree of testing needed. Note from the previous section that the possibility of developing desired confirmatory/supporting data in the course of operating the actual reactor is recognized as a possible approach.

4.0 OPERATING EXPERIENCE FOR METALLIC FUEL

The discussion in this section is based in large part on a review [7] of sodium-cooled fast-reactor fuels. This review included all possible fuel types that have been proposed for fast reactors, including metal-based fuels.

4.1 Evolution of Metallic Fuel

Metal-based fuels have been used in US reactors for many decades. These include the Production Reactors at Hanford and Savannah River, various research reactors (e.g., High Flux Beam Reactor), and the Experimental Breeder Reactors (EBR-I and EBR-II). In addition, internationally metal fuel was used in the Dounreay Fast Reactor (DFR) in the UK [8, 9, 10]. The reason for selecting metal-based fuel include: the relative ease of fabrication, high thermal conductivity and high heavy metal content. The heavy metal (uranium or plutonium) are generally alloyed with other metals to increase dimensional stability, corrosion resistance, and to optimize the melt temperature of the alloy to result in desired values for operation and fabrication.

Early metal-based fuel alloys exhibited unusual growth patterns following irradiation. These patterns were attributed to the micro-structural phase separation brought about by anisotropic crystal growth during the fabrication process. To overcome these phenomena, the as-fabricated fuel was heat treated to remove all crystallographic texture, resulting in an isotropic crystal structure. Fuels used in EBR-I include unalloyed uranium, and U-Zr and Pu-Al alloys, while U-Cr and U-Mo were used in the DFR. EBR-II was started with a U-5Fs (Mark-I,-IA, and -II) fuel alloys. The designation Fs ("fissium") represents a mixture of noble metals commonly found in fission products (5Fs: 2.4 wt% Mo, 1.9 wt% Ru, 0.3 wt% Rh, 0.2 wt% Pd, 0.1 wt% Zr, and 0.1 wt% Nb). In the 1980s, the EBR-II was converted to using a U-Zr metal alloy (Mark-III and –IIIA) in the driver section of the core [8, 11, 12].

The early EBR-II metal fuel designs were not able to meet the burnup requirements desired for fast-reactor fuel cycle designs. This lead to an adaptation of existing LWR oxide fuels to fast-reactor designs, and the EBR-II was used as an irradiation facility for the development of these fuels. However, in an effort to optimize costs, the metal driver fuels used in EBR-II were developed with an eye to increasing the burnup capability of the fuel. The early U-5Fs fuel designs were capable of burnups of ~ 2.6 at% [8, 9, 13]. The fuel pin failure mode was generally a cladding breach caused by stresses in the clad resulting from fuel swelling, fission gas release, and by stresses induced in the fuel liftoff restraints. These problems were addressed by increasing the fuel-clad gap, and the addition of an impurity-level amount of Si. The gap size was increased to accommodate the fuel swelling to the point where the fuel porosity interconnects to allow fission gas release into the fission gas plenum [9, 14, 15]. This release of gas to the plenum reduces or eliminates the driving force of the swelling mechanism, thus reducing the loads on the clad. The fuel design was further developed by using a thicker clad and increasing the volume of the fission gas plenum. Finally, the clad material was changed from type 304L stainless steel to a type 316 stainless steel. At this stage, the Mark-II fuel design was capable of reliable operation up to 10 at% burnup [11, 10].

4.2 Development of U-Zr and U-Pu-Zr Fuels

While the use of metallic fuel in EBR-II (and eventually FFTF) continued, the CRBR project was initiated. Oxide fuel was selected for the CRBR for a number of reasons, including international experience, and industry familiarity with this fuel form. However, the CRBR project was terminated, and the Integral Fast Reactor (IFR) project was initiated with multiple purposes, one being the continued development of metal-based fuel. These fuels potentially provide increased inherent reactor safety characteristics compared to oxide fuels due to more desirable feedback characteristics, compatibility with the sodium coolant in the case of a clad breach, and more acceptable transient over power behavior. In addition, the metal fuel was compatible with the pyro-metallurgical re-cycling process proposed by the IFR project [16, 17].

Phenomena that had been investigated and understood and that were known to control metal fuel lifetime, such as swelling, and fission gas release in the U-5Fs fuels were reinvestigated in the proposed new fuel alloys.

Fuel swelling is driven by growth of fission product gas bubbles which continue to grow as the irradiation progresses. This growth continues until a release path is created by interlinkages of porosity within the fuel pin. Figure 4.1 shows the fraction of fission product gas that is released from the fuel as a function of burnup [18]. It is seen that initially there is a rapid increase in release, until the burnup reaches approximately 2–3 at%. At this point, it remains essentially constant at approximately 75% up to a burnup of 20 at%. This characteristic variation is very similar for U-10Zr, U-8Pu-10Zr, and U-19Pu-10Zr metal alloy fuels. In addition, the variation of fuel volume increase (swelling) with fission product gas release is shown in Figure 4.2 [19]. It is seen that initially there is no gas release, but after approximately an increase in volume of approximately 25%, the gas release increases rapidly with volume increase. Presumably, this rapid increase is because the porosity interlinkages have occurred allowing the gas to leave the fuel pin. Furthermore, the fuel pin contacts the inner surface of the clad at this point, which inhibits the rate of gas release. The design requirement that enables the swelling and gas release characteristics shown in Figures 4.1 and 4.2 is a smear density of approximately 75% for the fuel. This value of the smear density allows sufficient swelling to facilitate gas release by interlinkage of porosity due to gas bubbles before the fuel pin contacts the clad. Following contact with the clad, the continuous release of gas to the fission gas plenum has been established, and thus reduces the potential for fuel clad mechanical interaction (FCMI). At high burnups, the additional accumulation of solid fission products creates the potential for FCMI with increasing burnup due to swelling.

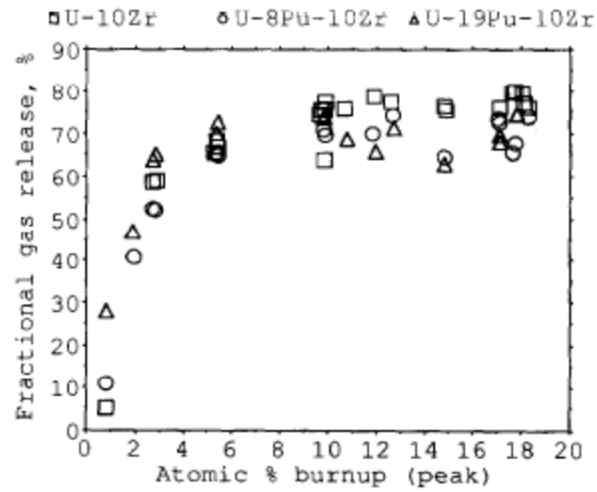

Figure 4.1 – Fractional gas release as a function of Burnup for typical IFR Metallic Fuel.

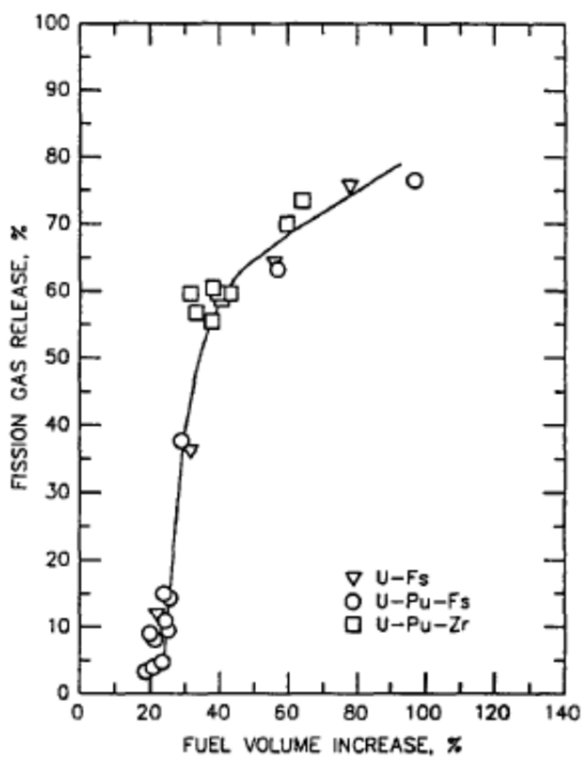

Figure 4.2 – Fission gas release as a function of Fuel Volume Increase for typical Metallic Fuel.

In addition to the radial swelling described above, the fuel swells axially. This swelling has been found to be anisotropic [13]. In particular, the axial swelling is consistently lower than the radial swelling. Furthermore, this anisotropy increases with increasing plutonium content of the metallic fuel alloy. The primary reason for this phenomenon is due to the difference in swelling behavior between the hotter central zones of the pin, and the cooler peripheral zones of the pin. In the central zones, γ-phase of uranium dominates, while on the outer periphery the α-phase of uranium dominates. These phases have different swelling characteristics, which leads to the anisotropic behavior. This behavior is shown in Figure 4.3 [13]. It is seen that the alloys with no plutonium (maximum uranium content) have the largest axial growth, while those with the highest plutonium (minimum uranium content) content have the lowest axial growth.

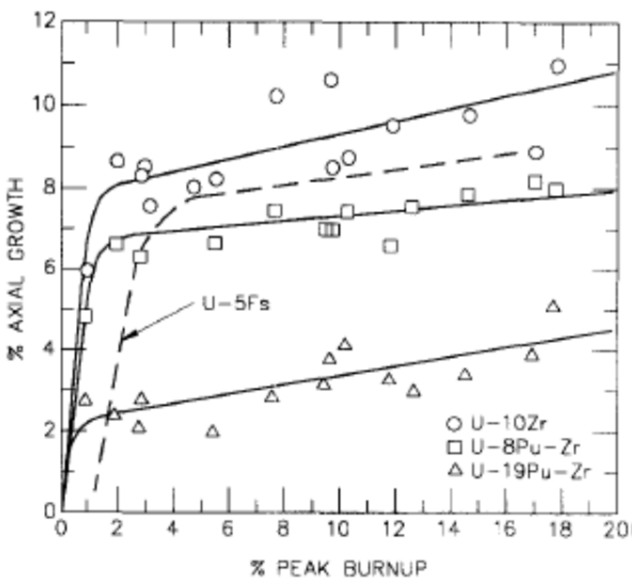

Figure 4.3 – Fuel Axial Growth as a function of Burnup for various Metal Fuel Alloys. (All fuels had similar initial smear density)

In addition to swelling of the fuel alloy due to gas generation and radiation damage, the clad material is also subject to swelling. In the case of the clad, gas production is due to the interaction of fast neutrons ($E_n > \sim 1$ MeV) with the isotopes in the clad material alloy via (n,p) and (n,α) reactions. The clad materials are primarily composed of chromium, iron, and nickel, and the nickel isotopes are a significant source of gas via the above reactions.

The new clad alloys that showed improved swelling and irradiation creep behavior compared to the type 316 stainless steel were also used in the proposed fuel designs. An experimental program was initiated to develop U-Zr and U-Pu-Zr metal alloy fuel designs clad in alternate cladding alloys. Briefly, this program demonstrated that it was possible to achieve close to 20 at% burnup with either a U-10Zr or a U-19Pu-10Zr fuel clad in either a 20% cold worked austenitic stainless steel, such as D9, or a ferritic-martensitic stainless steel, such as HT9. Figure 4.4 [13] shows the sequential improvement in swelling and creep behavior for various cladding alloys. Both D9 and HT9 demonstrated low swelling behavior, and of the two cladding materials tested, HT9 was the preferred option.

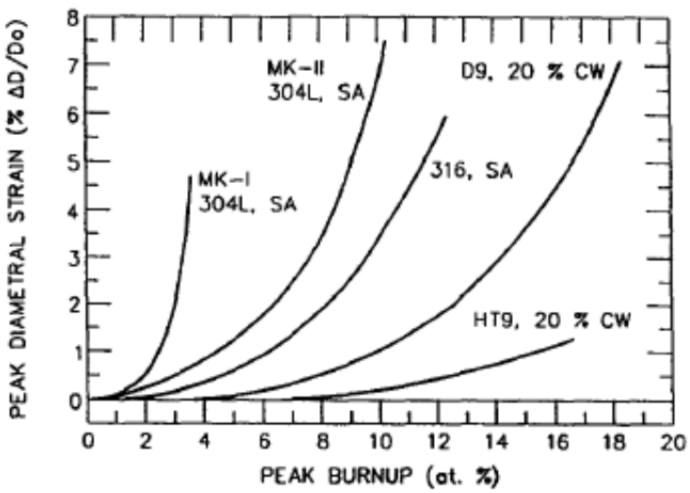

Figure 4.4 – Progressive Improvement in the Deformation (Swelling and Creep) of the Cladding of Metallic Fuel Elements. EBR-II Irradiation.
(SA=Solution Annealed; CW=Cold Worked)

A combination of the effects due to fuel smear density, fission gas release, and cladding strain, for a fuel pin clad with HT9 is shown in Figure 4.5 [13]. The data shown in this figure is for U-19Pu-10Zr fuel alloy. It is seen that as the fuel smear density increases beyond 75%, the maximum strain increases and the fission gas release fraction decreases.

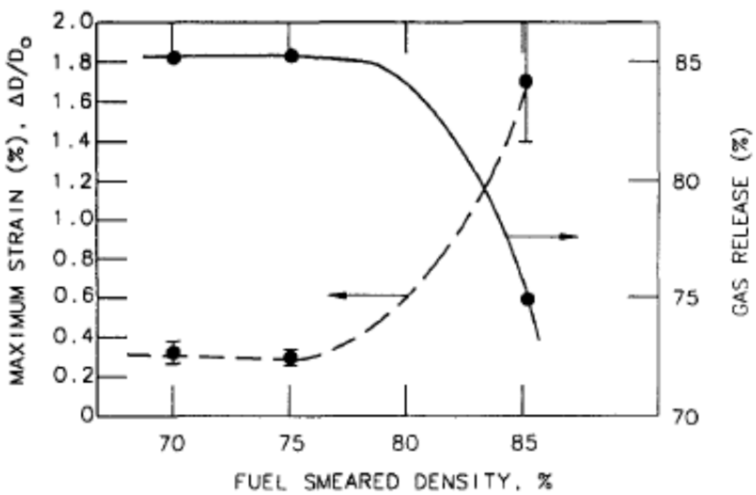

Figure 4.5 – Peak Cladding Diameter Increase and Gas Release Fraction for HT9 Clad U-19-Pu-10Zr Fuel of Various As-built Smeared Densities, at 12.5 at% Burnup.

In addition to the above mentioned metal fuel phenomena that limit its burnup (swelling, fission gas release, etc.), there are phenomena that need to be addressed that might play a role under

accident conditions. These phenomena include fuel re-distribution within the fuel rod to create Zr depleted zones within the fuel rod, and fuel-clad inter-diffusion and fuel-cladding chemical interaction (FCCI), which are enhanced by lanthanide fission products. The lanthanide fission products increase with burnup, thus increasing the phenomena at the end of life. The result of this inter-diffusion and FCCI is to lead to clad thinning and possibly failure. The uranium and zirconium redistribution boundaries follow isotherms in the radial fuel temperature distribution. These isotherms determine the various phase boundaries of the fuel alloy. The peak fuel temperature generally occurs near the outlet end of the core. In high power density pins (high temperature and steep thermal gradient), the redistribution pattern may extend to the top of the core. Examples of this redistribution are shown in Figures 4.6 [13] and 4.7 [20]. It is seen that in both cases, there is a zirconium-rich zone in the center, followed by a uranium rich ring. On the outer periphery, the composition is essentially unchanged. The plutonium composition is largely unaffected by the redistribution of uranium and zirconium. The implication of this redistribution can be understood by considering the U-Zr phase diagram, given in Figure 4.8 [20].

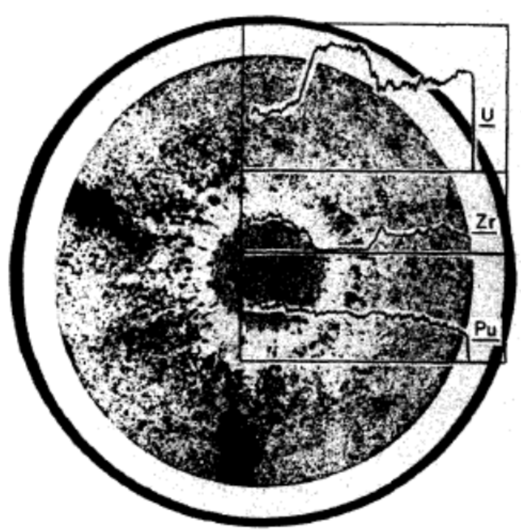

Figure 4.6 – Transverse Metallographic Section from the High-Temperature Region of a U-19Pu-10Zr Element at 3 at% Burnup; Superimposed Mocroprobe Scans showing Zone Formation, Cracking and Zr-U Redistribution.

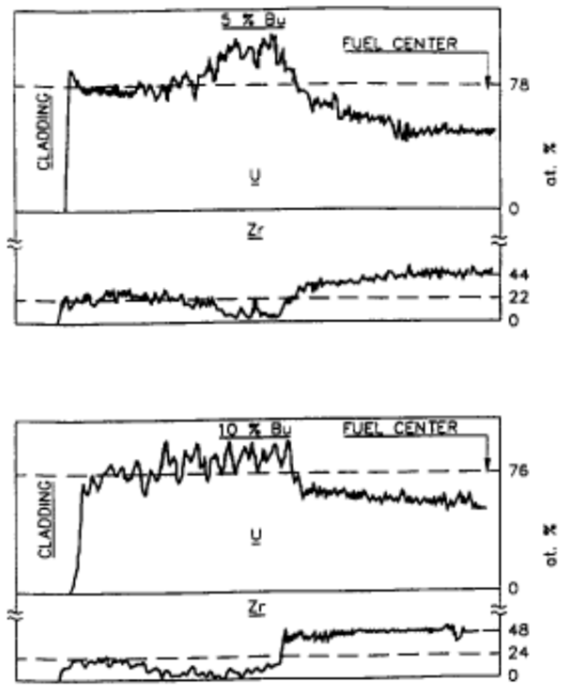

Figure 4.7 – Radial Microprobe Scans at Top of Fuel Pins DP-81 (5 at% peak burnup) and DP-11 (10 at% peak burnup).

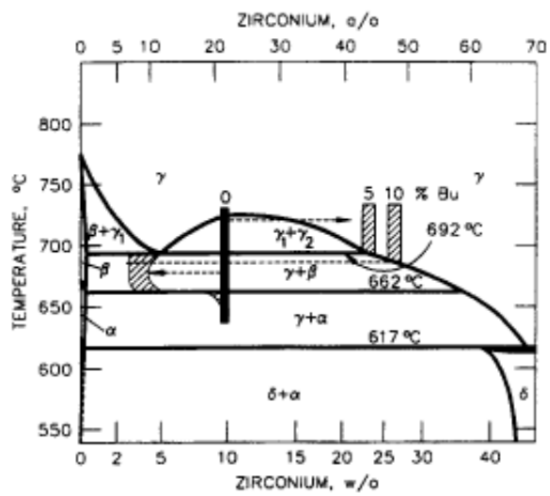

Figure 4.8 – Partial U-Zr Phase Diagram with approximate Radial composition at the Top of the Fuel pin at 0, 5, and 10 at% Peak Burnup.

With increasing zirconium content, the melt temperature decreases over the range of interest. Fresh fuel has a melt temperature of approximately 730º C, which drops to 692º C at 5 at% burnup and then keeps dropping with increasing burnup. The fuel thus takes on a radially varying pattern of lower melt point zones. Under normal operating conditions, the temperature would not exceed the local solidus temperature even at the highest burnups, and thus this phenomenon is important only under certain accident conditions.

Furthermore, it was established that U-Pu-Zr and clad inter-diffusion leads to low melt eutectics that could penetrate the clad under accident conditions. The formation of low melt eutectics is illustrated by the following two phase diagrams. Figure 4.9 [21] shows the diagram for U-Fe, and Figure 4.10 [22] shows the diagram for Pu-Fe. Figure 4.9 shows a low melt eutectic at an atom fraction of uranium of 65% at 721º C for a uranium-iron alloy. This alloy could exist when the fuel contacts the clad, which is a ferritic stainless steel. However, the implied temperature is high, and may only be encountered under accident conditions. In the case of Pu-Fe alloys, the phase diagram shown on Figure 4.10 shows low melt alloys at much lower temperatures (410º C), but at high fractions of plutonium (approximately 90 at%). The eutectic formation at the fuel clad interface is further complicated by the presence of lanthanide fission products. These fission products play a role in the formation of low melt eutectics also with the clad alloy, which will have different temperatures than the eutectics associated with uranium or plutonium and the clad alloy. The effect of the lanthanide fission products will increase with burnup, and will contribute to the formation of low melt compositions at the fuel/clad interface at end-of-life.

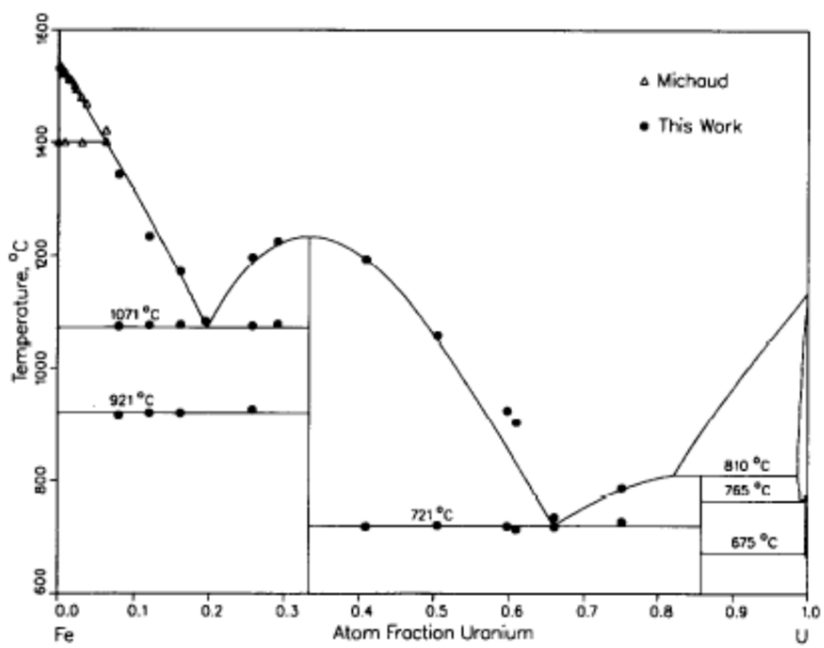

Figure 4. 9 – U-Fe Phase Diagram

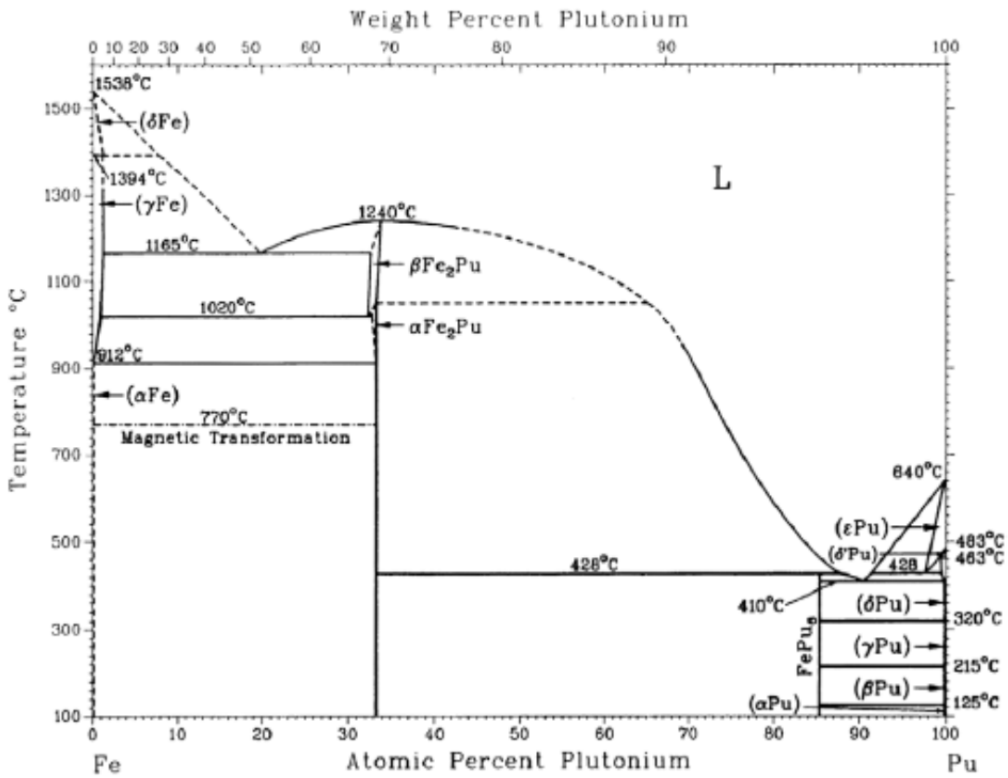

Figure 4.10 – Pu-Fe Phase Diagram.

The alloy formation at the interface of the fuel and clad is very complicated and can best be understood by carrying out appropriate experiments at elevated temperatures. Figure 4.11 shows an experimentally determined envelope of clad penetration as a function of time at 800º C [13]. Since the penetration rate increases with temperature, the experimental results were acquired at higher than normal operating temperatures in order to collect the data in a practical time frame. In addition, the higher temperature also simulates possible accident conditions. It is seen that three different fuel types (U-10Zr, U-19Pu-10Zr, and U-26Pu-10Zr) and three different clad alloys (HT9, D9, and 316) are covered. The data shows that the deepest penetration occurs when U-10Zr is in contact with HT9 or D9 clad, whereas if 316 stainless steel is used the penetration is significantly lower. Furthermore, the penetration decreases with increasing plutonium content of the fuel. However, even for the worst case, it is seen that the deepest penetration is approximately 250 μ, after 2 hours at 800º C. Under normal operating conditions, the clad/fuel interface temperature is expected to be well below the eutectic temperature shown on the U-Fe phase diagram (721º C), and this effect should not be significant at BOL. The above data is contrary to data collected [13] using couple experiments, in which fuel alloys in contact with clad materials are heated. In these couple experiments, it is generally found that the penetration into the clad increases with increasing plutonium content in the fuel. However, these experiments did not include the effects of rare-earth elements (lanthanides) in the fuel alloy. Briefly, it can be concluded that prior to the accumulation of lanthanide fission products the FCCI depends on the depletion of nickel in the case of austenitic steels, and of carbon in the case of the martensitic steels at the fuel clad interface. Once the lanthanides are created in sufficient

quantities and have migrated to the fuel/clad interface, their presence dominates the FCCI process. Their radial migration increases with increasing burnup and temperature gradient and is more pronounced for U-Pu-Zr fuels than for U-Zr fuels. However, the data shown in Figure 4.11 shows less penetration for the cases using U-Pu-Zr fuel than those using U-Zr fuel alloys. This indicates that a complete understanding of the behavior will require more time to develop a satisfactory fundamental model for the observations in Figure 4.11. In the meantime, empirical correlations based on this data can be used to predict FCCI behavior. Finally, the use of austenitic stainless steels as a clad material is probably practical in the standard form for burnups less than 10 at% only, due to its unacceptably high rate of swelling caused by interaction with fast neutrons. Thus, martensitic stainless steels are of primary interest for higher burnups, despite their larger susceptibility to penetration during the FCCI process for higher burnup applications.

These phenomena were used to set Limiting Conditions of Operation for operations at the EBR-II reactor using U-Pu-Zr alloy fuel.

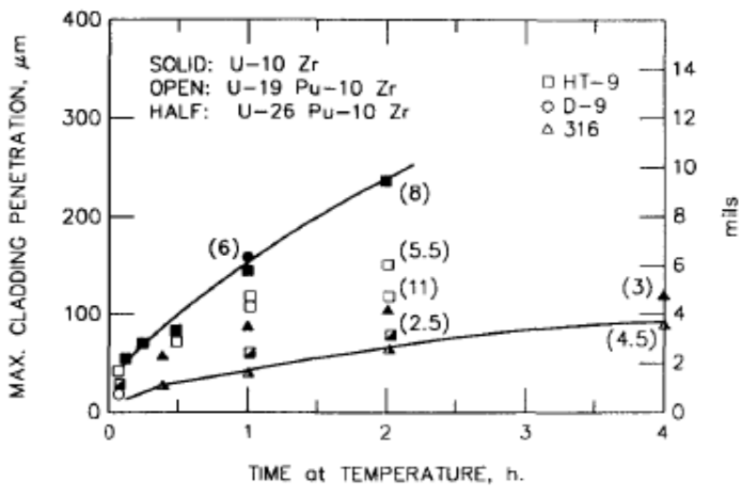

Figure 4.11 – Envelope of Depth of Penetration by Fuel/Cladding Melt into the Cladding as a Function of Time at 800º C during Post-Irradiation Heating Tests. (In parentheses: percent burnup of test samples.)

4.3 Steady-State Burnup Performance and Reliability

The early experience regarding metal fuel steady-state performance is closely coupled to the operation of the EBR-II and the performance of its driver fuel. This experience included both the fabrication of the fuel and the subsequent exposure in the reactor. In total, over 130,000 driver fuel rods were fabricated and irradiated in the EBR-II. Of these 120,000 rods were of the earlier Mark-1/1A and Mark –II designs. Of primary interest to this study are the remaining 13,600 rods (Mark-III/IIIA/IV), these included 13,000 fuel rods composed of a U-10Zr alloy, and over 600 rods composed of a U-Pu-Zr alloy.

The Mark-II driver fuel rods were qualified for a maximum burnup of 8 at%, while the Mark-IIIA driver rods were qualified for a maximum burnup of 10 at%. U-Zr and U-Pu-Zr rods in the experimental positions in the core achieved maximum burnup values of 15-20 at%. The cladding material used in these rods was either a type 316 stainless steel, D9 or HT9. Rods tested in high temperature (2-sigma) assemblies achieved 11-12 at% burnup, which is slightly lower than that achieved under normal operating temperature conditions. In addition to the irradiations carried out at the EBR-II, several irradiations were carried out at the FFTF. These rods were longer than those used at the EBR-II and thus are useful in quantifying the effect of geometry on rod performance. In addition, these tests were carried out to qualify the U-Zr alloy rods as a possible driver fuel design for the FFTF. In all, 1050 U-10Zr fuel rods and 37 U-Pu-Zr fuel rods were exposed in the FFTF to burnup values above 14 at% and 9 at%, respectively. The results of these tests indicate that within the limitations of the test conditions:

1. There were no fuel rod length dependent effects on the fuel rod behavior, and
2. Since the outlet temperatures of the two reactors is equal, it implies that the axial thermal gradient in the EBR-II is steeper, and furthermore the burnup in the case of the EBR-II is higher (15–20 at%) implying that the EBR-II tests were a more severe test of the fuel even considering that the fluence-to-burnup ration is higher in FFTF than in EBR-II.

An aspect of the fuel acceptability is its compatibility with the coolant, and the implications of operation of the reactor beyond clad breach. In this case, it is possible for the primary coolant to come in contact with irradiated fuel rod material, and thus any fission product or fuel release would end up in the coolant circuit. Of the 13,600 U-Zr and U-Pu-Zr fuel rods irradiated in the EBR-II, 22 breached under irradiation. Of these, 16 breaches were due to defective welds (this problem was identified and solved); 3 of the remaining breaches occurred in the gas plenum region, with no known cause given. The remaining 3 breaches occurred in the fueled section of the fuel rod. One breach occurred in a U-Pu-Zr rod clad with D9, and the breach occurred ~ 2/3 up the height of the rod at an approximate burnup of 16.4 at%. This breach was possibly due to rod-rod interaction, which was possible at this height. Two breaches occurred at burnups of 6.5 at% and 10 at%, respectively, in rods with a plenum-to-fuel ratio of one. Subsequent to this experience, the plenum size was increased, and the revised ratio is now generally recommended be 1.4. In addition to the naturally occurring breaches, 7 rods were artificially breached by machining a weakness into an irradiated fuel pin and operating it until it breached. An example of such a rod is shown in Figure 4.12 [22]; the weakness and the failure location are clearly visible. It is also interesting to note the co-axial zone structure caused by redistribution of zirconium and uranium. This pattern indicates that this failure is close to the hottest spot in the pin, possibly due to the cladding having breached at that location. A summary of the seven intentionally breached rods and a natural breached rod are given in Table 4.1 [7]. The experiments were carried out in EBR-II using pre-irradiated rods. The reactor was operated beyond breach in order to assess the post-breach behavior. As indicated in the table, the fuel continued to be irradiated at normal power levels for many days beyond failure.

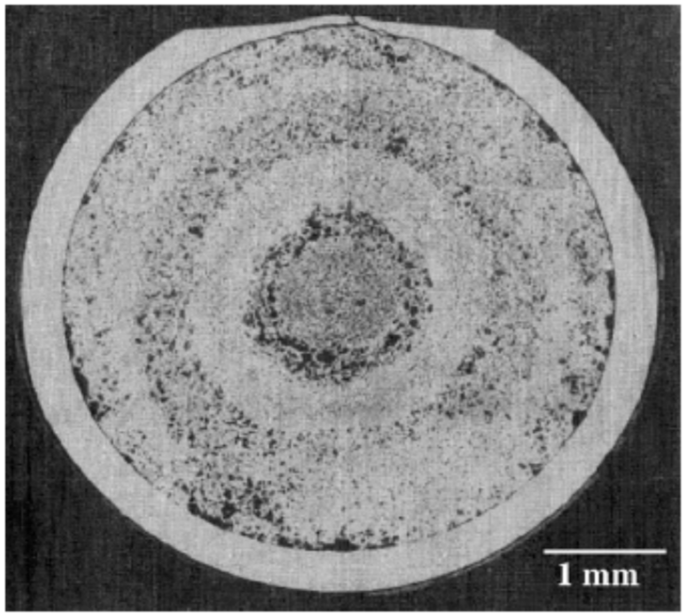

Figure 4.12 – Example of a Run Beyond Clad Breach Test of Metal Fuel.

Table 4.1 Summary of Run Beyond Cladding Breach Experiments in EBR-II

Case	Fuel Type	Clad Type	Clad OD (cm)	Linear Power (kW/m)	Clad Temp. (ºC)	Burnup (at%)	No. of Breached Rods	Days Irr'd Beyond Breach
1	U-5Fs	316 SS	0.44	24	573	7.9	0	N/A
2	U-5Fs	316 SS	0.44	25	593	9.3	1	54
3	U-19Pu-10Zr	316 SS	0.44	21	541	7.6	1	233
4	U-8Pu-10Zr	316 SS	0.44	23	520	6.0	2	131
5	U-19Pu-10Zr	D9	0.58	39	600	14.4	1	168
6	U-10Zr	D9	0.58	36	600	13.5	1	100
7	U-19Pu-10Zr	HT9	0.58	36	600	14	1	150
8*	U-19Pu-10Zr	D9	0.58			17	1	34

*Natural Breach, all other breaches were created at a machined weakness.

An evaluation of the results gathered from the breached rod operating cycles indicate that the coolant is compatible with the metal fuel alloy and does not react chemically with the fuel, thus eliminating the possibility of reaction products from further aggravating the breach size or increasing fission product release. Post-test analysis indicated that the bulk of the mass loss was due to the expulsion of bond sodium, fission product gas, and cesium. In the case of low swelling clad, there is the possibility of fuel-clad-mechanical-interaction, because the clad is closer to the fuel. The amount of fuel washed out of the breach was found to be minimal. The conclusion from these tests indicates that a sodium cooled reactor using metal fuel can operate with breached fuel.

4.4 Transient and Off-Normal Performance

Transient tests were carried out on metal based fuels in both the EBR-II reactor and the Transient Reactor Test Facility (TREAT). The difference between these tests is primarily the rate of reactivity insertion and the subsequent loss of fuel and clad integrity. In EBR-II, the tests were characterized as slow transients, while the TREAT tests are characterized by more rapid reactivity additions, and tested the margins to failure during a transient overpower (TOP) event.

The first set of transients involved Operational Reliability Tests (ORT) [9, 23] carried out on test fuels in the EBR-II. These tests included the driver fuel used at EBR-II, which was the Mark-II fuel design. Two reactivity ramp rates were considered, the first increased power at 1.6% per second, and the second increased the power at 4 MW per second [24]. Although this was not the fuel under discussion in the present report, being U-Fs alloy, the results were extended to U-Zr alloy driver fuel since their behavior is similar. In both cases, the low ramp rate tests showed no indication of fuel failure. The higher ramp rate also resulted in no failure, despite the fact that selected assemblies had experienced the lower ramp rate power increases prior to these tests, and had thus operated at higher temperatures.

As part of the IFR safety program a set of passive safety tests was carried out on the EBR-II, designated as Shutdown Heat Removal Test [25]. These included a loss of flow without scram and a loss of heat sink without scram. Both these transients were carried out on the EBR-II that had sufficient negative feedback to terminate these transients [22]. The power variations were relatively slow, and thus no core damage was detected. However, as with the ORT program, selected assemblies were tested at higher temperature following the transients described above to gauge their limits to failure. A test assembly was irradiated while experiencing clad temperatures reaching 800º C for ~ 42 mins. Post-test examination showed molten-phase attack on the clad but not sufficient to breach the clad. A subsequent longer term irradiation demonstrated a clad breach following a burnup of 10 at%. This breach was a stress rupture breach, typical of an end of life breach. In general, an assessment of the driver fuel following the transient tests showed incremental damage to the fuel following each test, and the probability of the driver fuel reaching its burnup limit was high.

Finally, a safety assessment of the fuel requires a determination of how the fuel rods will behave under a TOP event. These tests were carried out using the TREAT facility, and determined the margin to fuel failure, pre-failure axial fuel expansion, and fuel and coolant behavior following clad breach. A summary of the TREAT test is shown in Table 4.2 [7, 13, 26, 27]. In all, 15 rods

were tested with various combinations of U-Fs, U-Zr, U-Pu-Zr fuel, clad with either type 316 stainless steel, D9, and HT9. The results of these tests indicated that:

1. The tests showed that metal fuel rods consistently fail at approximately 4 times nominal power, under the relatively fast transients implied by these tests. The clad breach occurred at the top of the fuel column in all cases, and were attributed to at-temperature pin-plenum pressure and clad thinning due to eutectic formation of a molten fuel/clad phase that penetrates the clad wall.

2. Pre-failure fuel expansion for the U-Zr and U-Pu-Zr was similar and less than that observed for the higher burnup U-Fs. In all cases, the expansion was greater than that attributable to thermal expansion alone.

3. Post-failure behavior of failed fuel in the case of all tests indicated that the tests were characterized by a rapid expulsion and dispersal of fuel.

Table 4.2 Summary of TREAT Tests on Metal Fuel

Fuel/Cladding	Burnup (at%)	Overpower attained in test	Comments
U-5Fs/316 SS (Mark II)	0.3	4.1	16% axial expansion, fuel damage, but intact.
U-5Fs/316 SS (Mark II)	4.4	4.2	Cladding breached.
U-5Fs/316 SS (Mark II)	7.9	4.1	3% axial expansion, cladding breached
U-5Fs/316 SS (Mark II)	0.3	4.1	18% axial expansion, fuel damage, but intact.
U-5Fs/316 SS (Mark II)	4.4	4.0	4% axial expansion, fuel damage, but intact.
U-5Fs/316 SS (Mark II)	7.9	3.4	4% axial expansion, fuel damage, but intact.
U-5Fs/316 SS (Mark II)	0	3.8	4% axial expansion, fuel damage, but intact.
U-5Fs/316 SS (Mark II)	2.4	4.1	7% axial expansion, cladding breached.
U-5Fs/316 SS (Mark II)	4.4	3.8	4% axial expansion, fuel damage, but intact.
U-19Pu-10Zr/D9	0.8	4.3	1% axial expansion, fuel damage, but intact
U-19Pu-10Zr/D9	1.9	4.3	2% axial expansion, fuel damage, but intact.
U-19Pu-10Zr/D9	1.9	4.4	2-3% axial expansion, fuel damage, but intact.
U-19Pu-10Zr/D9	5.3	4.4	3% axial expansion, cladding breached
U-19Pu-10Zr/D9	9.8	4.0	3% axial expansion, cladding breached.
U-10Zr/HT9	2.9	4.8	2-4% axial expansion, fuel damage, but intact.

The pre-failure axial expansion of the fuel and the rapid expulsion and dispersal of fuel following pin failure, add negative reactivity during these phases of a postulated accident. This is potentially a beneficial effect in the case of reactivity insertion transients or accident scenarios.

Finally, a series of out-of-pile tests were carried out to simulate a loss of flow (LOF) event [7]. These tests were carried out using irradiated U-Pu-Zr fuel clad in HT9. The results from these tests were consistent with those obtained in the TREAT tests described above. These data are used to validate analytic models that are necessary to predict fuel behavior.

4.5 Summary of Experimental Data

The development of metal-based fuels used in the driver assemblies of the EBR-II test reactor has been evolutionary in nature. The initial fuel design was based on the U-Fs alloy, which was then converted to a U-Zr alloy. This conversion resulted in the utilization of approximately 13,000 driver fuel rods with a burnup limit of 10 at%. The addition of Pu to the alloy did not seem to change the mechanisms that control the fuel lifetime, although some of the effects were exacerbated with this addition. These include fuel clad inter-diffusion, and fuel constituent migration as a function of temperature. Thus, in summary, the U-Fs and U-Zr alloy fuels were qualified to be used as driver fuel in the EBR-II, and the process of qualifying U-Pu-Zr fuel for the same application was underway prior to its shutdown.

Based on the historic experience described above (summarized in Table 4.3), the life-limiting phenomena associated with metal fuel are generally understood; however, the data base is focused primarily on application to the EBR-II reactor. A limited number of tests have been carried out in the FFTF, but the experience base and hence confidence level is lower for utilization than for EBR-II.

Table 4.3 Summary of Metal Fuel Experimental Experience Base

Parameter	Experience
Driver fuel operation	~ 30,000 U-Fs rods clad in 316SS to 8 at%. ~ 13,000 U-Zr rods clad in 316SS to 10 at%.
Qualification	U-Zr in 316SS, D9, HT9 rods to 10 at% in EBR-II and FFTF.
Demonstrated burnup capability and experiments	U-Pu-Zr in D9 and HT9 to 10-20 at% in EBR-II and FFTF.
Safety and operability	6 run-beyond-clad-breach tests U-Fs and U-Pu-Zr/U-Zr. 6 TREAT tests U-Fs in 316SS (9 rods) and U-Zr/U-Pu-Zr in D9/HT9 (6 rods)

In addition, the opinion of a panel of experts that was convened to identify the gaps in fuel and materials research prior to licensing SFR designs will be considered below. The results of the discussions and conclusions of this panel have been included in a report [28]. The panel discussed various physical phenomena and then evaluated the state-of-knowledge using the following assessment definitions:

- High (H) – Implied that a physics based model is available, and that the database covers all possible applications.
- Medium (M) – Implies that the model covers most of the relevant phenomena over a considerable portion of the parameter space. Furthermore, the database is not necessarily complete, allowing only moderately reliable assessments.
- Low (L) – Implies no model exists, or that it is speculative, and the database does not exist. Assessments cannot be made reliably.

The following categories, relevant to the current discussion, that have regulatory concern were rated by the panel:

1. Fresh fuel at 10 at%, 20 at% and greater than 20 at% burnup. This fuel includes U-Zr, and U-Pu-Zr.
2. Life-limiting phenomena for HT-9 cladding,
3. Life-limiting phenomena for 316 cladding,
4. Life-limiting phenomena for advanced materials for cladding e.g., 9Cr-1Mo,
5. Life-limiting phenomena for 316 and HT-9 ducts.

Not all of the above categories will be summarized to the same level of detail, since the primary objective of the current report concerns the behavior of the fresh fuel. However, cladding and duct materials will be discussed since their behavior will impact the fuel performance. The regulatory concern column in Tables 4.4 and 4.5 is a measure of the importance, as perceived by the panel, of the phenomena to the licensing process.

Table 4.4 Potential Life-Limiting Phenomena for Fresh Fuel

Phenomena	Regulatory Concern	Bu<10 at%	10 at%<Bu<20 at%	Bu>20 at%
Axial Growth	L	H	M	L
Swelling+ FCMI	H	H	M	L
Gas Release	H	H	H	L
Re-distribution	M	H	M	L
FCCI	H	H	M	L
Fuel/Coolant Compatibility	L	H	H	H

Table 4.5 Phenomena and Properties for HT-9 Cladding

Phenomena	Regulatory Concern	Low dpa* Low PCT**	Low dpa High PCT***	High dpa+ Low PCT	High dpa High PCT
Creep rate	H	H	M	H	L
Swelling	M	H	M	H	L
Fracture Toughness	M	H	M	H	L
Yield Strength	M	H	M	H	L
Carbon Transport	L	N/A	N/A	N/A	N/A
FCCI	M	H	M	H	M

*Low dpa < 100
**Low Peak Clad Temperature (PCT) = 550 - 560ºC
***High PCT ~ 630˚C
+High dpa ~ 200

The phenomena will be discussed below:

1. Axial growth was deemed to have low regulatory concern. U-Zr based fuels grow considerably more than U-Pu-Zr based fuels, and the phenomenon has a bigger impact on the reactor operations, since the growth implies negative reactivity input, which must be compensated for by additional control-rod reactivity. Extensive data is available on this phenomenon [29].

2. Fuel swelling was deemed to have a high regulatory concern. Most swelling occur during the first few at% burnup (~ 2-3 at%) at which point the voids causing the swelling inter-connect and the subsequently generated fission gas is released to the fission gas plenum. This is achieved by having an adequately low smear density (~ 75%). A sufficiently large fission gas plenum is required to accommodate the released gas. Extensive data has been collected on this phenomenon [29].

3. Gas release is deemed to have a high regulatory concern. The need for a large fission gas plenum to contain approximately 80% of the fission gasses is required. Too small a plenum can result in clad creep rupture, thus limiting the life, and too large a plenum has economic implications. Data has been reported on this phenomenon that can be used to help guide the design of an appropriate plenum size [29].

4. Fuel constituent re-distribution was deemed to have medium regulatory concerns. This phenomenon is caused by thermal gradients that result in alternatively zirconium rich and zirconium depleted zones. The zirconium content influences the fuel melt temperature, but the fuel design must demonstrate that sufficient margin exists under all conditions to prevent any fuel melting. Thus, the regulatory concern is considered not to be high. Data exists on this phenomenon, but it is difficult to obtain; in addition, theoretical models exist that explain the phenomenon [30, 31].

5. FCCI is considered of high regulatory concern. This phenomenon results primarily from lanthanide fission products that migrate through the fuel to the clad and react with the stainless steel alloys. There is an initial incubation period associated with the creation of the fission products and the subsequent transport to the fuel/clad interface. The reaction at the clad inner surface results in a brittle interaction layer that grows with burnup and is considered wastage. Thus, FCCI acts to thin the clad, increasing the stress, and potentially resulting in creep rupture. However, for burnups up to 10 at% and peak temperature of approximately 600˚C, FCCI has not been a life-limiting phenomena. Extensive data exists on this phenomenon [32].

6. Fuel/Coolant Compatibility was deemed of low regulatory concern. This phenomenon is of low consequence, since the fuel is bonded by sodium and is thus completely compatible with sodium.

Currently, there are two alloy classes that have sufficient radiation data to be considered as cladding and structural materials in a SFR. First, the austenitic stainless steels (Type 316 or D9), which are limited to ~ 100 dpa and by implication a burnup of ~ 10 at%. This limit is due to

embrittlement and distortion caused by swelling and irradiation induced creep. However, if the exposures are limited to ~ 100 dpa, there are essentially no gaps in the data and knowledge base.

Second, an alloy class consisting of ferritic and ferritic-martensitic steels, which are much more resistant to swelling, but suffer from loss of creep strength at higher temperatures. In the US, HT-9 is the leading candidate steel to fill this role. If the operating conditions of the steel are limited to those outlined in Table 4.5, then there are no significant gaps in the data base.

Table 4.5 is specific to HT-9, and it is seen that for dpa values as high as 200 and peak clad temperatures limited to 550 - 560ºC, the data base is sufficient to address all the regulatory concerns. However, it should be noted that HT-9 does not maintain its strength at higher temperatures. In the case of the austenitic stainless steels (Type 316 or D9), the dpa limit is approximately 100, but they are able to operate at higher temperatures (~630˚C). Reviews of these processes and their consequences in austenitic stainless steels are presented in the literature [33, 34].

Ducts are manufactured of the same two alloys, and the comments regarding these materials made above apply to ducts as well. In addition, in the case of ducts, the dimensional distortion, which leads to bundle and neighboring duct interaction are of high regulatory concern.

The conclusions of this panel [28] and the application of this data (summarized in Table 4.3) to support the licensing of the small modular SFRs that are the subject of this review are addressed in Section 6.

5.0 FUEL PERFORMANCE MODELING

In this section, a selection of currently available codes for evaluating the performance of metallic fuel will be described and their level of validation discussed. The summary below, therefore, reflects potential tools that are "publically" available for this purpose to the vendors but does not necessarily reflect the current status of the capabilities of these codes.

It is also important to note that development of "first principles" advanced fuel performance code(s) is actively being pursued by the DOE-NE Fuel Cycle Technologies and NEAMS efforts. These codes may become available in a time frame to support licensing of these reactors. Validation of these codes against separate effects and integral experiments is being actively pursued.

5.1 LIFE-METAL [35]

LIFE-METAL is a stand-alone, fuel performance code for predicting metal-based fuel pin behavior. It is an evolution of similar codes that have been developed for ceramic-based fuels. Although it recognizes both axial and radial variations in key parameters, it is primarily a one-dimensional code with detailed treatment in the radial direction. Radially, the code has a detailed thermo-mechanical model for the fuel/gap/clad system. The radial mechanical analysis uses a generalized plane strain assumption, with separate thermal and mechanical radial rings. Each thermal ring contains several mechanical rings. The gap has only one ring, and the clad has several rings. In addition, rings can be added to represent the possibility of fuel clad chemical interaction that might take place at elevated temperatures. This interaction is due to the inter-diffusion of selected clad alloy components into the fuel volume and vice versa.

Other metal-based fuel phenomena that impact fuel performance are swelling, axial fuel growth, and constituent redistribution. The constituent redistribution results in regions that are depleted in zirconium and as a result have lowered melting points. This reduction in melting point becomes significant during over temperature accident scenarios, since a reduction in the margin to fuel melting is possible, and should be determined. The redistribution also impacts the swelling and axial growth, since it has been found that the swelling is not symmetric and is composition dependent. Of these phenomena the code models the axial growth, but does not account for the constituent redistribution at this time. The axial growth is accounted for by a correlation that is burnup dependent. The inter-diffusion coefficients related to the constituent redistribution have been measured and can be used to validate an advanced code.

Swelling and fission gas release are related phenomena and are functions of fuel burnup. Swelling proceeds rapidly in the initial stages and then slows down as the fission gases are able to escape from the fuel. Virtually all the length increase occurs during this phase, before the fuel contacts the clad due to radial swelling (~ 1% burnup). Both axial growth and fission gas release are represented by experimentally determined correlations in the code. The axial growth is represented by a correlation that is a function of burnup alone. Fission gas release is represented by a correlation which gives the release rate, and is a function of burnup, porosity, temperature, and fuel swelling strain due to the gas bubbles. Swelling from both gaseous (compressible), and

solid (incompressible) fission products are included. This difference in fission product state affects the strain induced by fission products.

Fuel clad chemical interaction is due to an inter-diffusion of fuel and clad material and results in clad thinning. In addition, the interaction layer is a low melt point composition that has the potential of growing in thickness, thus eroding more of the clad material. This phenomenon is modeled using clad wastage data obtained from irradiated fuel experiments at EBR-II. Wastage on the outer surface of the clad is based on sodium corrosion, de-carbonization, and inter-granular attack. Inner surface clad wastage is based on the same phenomena as on the outer surface due to sodium attack, and the additional fuel clad chemical interaction. Wastage data base for HT9/U-xPu-10Zr and HT9/U-10Zr systems form the basis of a correlation for the inner wastage, which is a function of fast flux, clad inner temperature, and time. The wastage process is given as a rate.

The LIFE-METAL code is based in large part on data obtained from the EBR-II data base, and has been validated against experiments carried out on both EBR-II and FFTF.

5.2 **FPIN2** [36]

FPIN2 is a pre-failure metal fuel pin code that is embedded in the SAS4A/SASSYS code package. Stand-alone versions of the code are also available, but its current use is as a part of the code package.

The thermal modeling used in the code is rudimentary, and in the current application, the temperatures are obtained from detailed analyses determined by other modules in the SAS4A/SASSYS code package. The more detailed modeling involves the mechanical behavior of the fuel pin. The mechanical model is based on a rigorous force-displacement formulation and uses an implicit finite element method with linear shape functions. The elements are defined in a (r,z) mesh, but axial symmetry is assumed, so the analysis is essentially one dimensional. The elements are allowed to interact only on the radial boundaries, and displacement within the elements are approximated by linear functions of the radial displacements.

Models specific to metal-based fuels are also included. These models describe the fission gas generation and release, molten fuel cavity formation, the gas plenum, and fuel clad eutectic formation. These models complement the fuel element mechanical calculation. Internal pin pressure is determined from mass and volume balances in the molten cavity and the gas plenum. The molten cavity is described by the axial and radial extent in which the fuel has reached its solidus temperature. The finite elements within the cavity are removed from the stress-strain calculation. In cases where the melt cavity is below the top of the fuel column, the plenum and cavity pressure are de-coupled. Once the melt reaches the gas plenum, the two pressures are assumed to be equal and the plenum/cavity pressure volume equations are solved simultaneously, to result in a common pin pressure and the amount of molten fuel extruded into the plenum.

The overall modeling for this metallic fuel version of FPIN2 was validated by comparison of calculated parameters with measured data obtained from the TREAT tests of EBR-II irradiated fuel that was prototypic of the IFR concept.

5.3 DEFORM-5

The current version of DEFORM-5 is integrated into the SAS4A/SASSYS code package and receives relevant input from modules within the code package. DEFORM-5 is used to predict the margin to clad failure and cladding failure timing and location in fuel pins containing metal fuel clad in stainless steel. The model calculates the mechanical response to transient thermal and pressure loading conditions which is then compared to expected failure characteristics and criteria to yield quantified measures of margin to failure, failure time, and failure location. This model is only valid for irradiated fuel in which the fuel has swelled to the inner clad surface.

Currently, there is no fission gas generation and migration model in DEFORM-5, but it assumes that an estimate of the fission gas content within the fuel and the gas plenum are made by some other code module. Transient fuel element and internal fuel pressures are calculated using the calculated fission gas mass and the transient temperature in the ideal gas formulation.

Uranium and uranium/plutonium-based metal fuels interact with the iron in the clad material to form low melt point alloys. The melt points vary as a function of uranium content in the alloy, and an inspection of the U/Fe phase diagram indicates that it is possible to create liquid material at a temperature of 725º C for an alloy containing 89 wt% uranium (see Figure 4.9). The primary importance of eutectic formation in this application is to determine the clad thinning implied by the melt front progression. This is accomplished by using appropriate correlations for various temperature ranges. These correlations, which are functions of interface temperature, are given as a rate and are based on data acquired from EBR-II experience. The cladding wastage as a function of axial height can be determined as a function of time during the transient. In order to predict the clad failure, an estimate of the clad strain is required. This is based on a stress estimate determined by assuming a thin shell model, and then estimating the strain using a plastic flow model for stainless steels.

The determination of cladding breach, or margin to cladding breach, is based on the eutectic thinning of the cladding and the reduced ability to contain the internal pressure. Besides the eutectic thinning, the cladding wall thickness is reduced as plastic flow takes place. Different accident scenarios and pin conditions lead to different modes of breach, whether through cladding penetration or plastic strain, or a combination thereof. In addition, in DEFORM-5, the time-dependent cladding damage fraction is calculated based on time remaining to breach. This estimate results in a time to rupture. Correlations are used to make these estimates, which are a function of hoop stress, fast neutron fluence, steady state temperature, and transient temperature ramp.

5.4 SAS4A/SASSYS [37]

The SAS4A/SASSYS code package has been developed to analyze thermal, hydraulic, and neutronic power and flow transients in liquid metal cooled nuclear reactors. These codes are

used to analyze severe core disruptive accidents, and SASSYS is additionally used to assess the margins in design basis accidents and to analyze the consequences.

The SAS4A module contains detailed mechanistic models to describe transient thermal, hydraulic, neutronic, and mechanical phenomena associated with the response of a fast reactor core to an accident scenario. The models provide the capability to analyze the initial phases of a core disruptive accident, through coolant heat up and boiling, fuel element failure, and the melting and re-location of fuel and clad. This code can treat both oxide and metal based fuel types.

The SASSYS module has the same core models as the SAS4A module but also provides the capability to model the balance of plant, including primary, secondary and tertiary loops. Models are included for heat exchangers, pumps, valves, turbines, and condensers. In addition, a plant control system model is included.

The SAS4A/SASSYS code package consists of sixteen modules that are linked together in an appropriate manner to solve the problem at hand. A list of these modules and a short description are given below in Table 5.1.

Table 5.1 SAS4A/SASSYS Modules

Module	Purpose
ROOT	Logic path control, data management, and material properties.
TSCL0	Single phase liquid coolant thermal/hydraulic and fuel element heat transfer.
TSPK	Reactor point kinetics and first order perturbation theory reactivity feedback.
PRIMAR-4	Primary and secondary coolant loops and components thermal/hydraulics and heat transfer.
CNTLSYS	Reactor and plant control and protection system simulation.
BOP	Balance of plant systems and components thermal/hydraulics and heat transfer.
DEFORM-4	Oxide fuel/cladding fuel element mechanics
DEFORM-5	Cladding mechanics for metal fuel elements
SSCOMP	Metallic fuel pre-transient characterization and material properties.
FPIN2	Metallic fuel/cladding fuel element mechanics.
TSBOIL	Two-phase (boiling) coolant thermal/hydraulics and fuel element heat transfer.
CLAP	Molten cladding relocation and heat transfer.
PLUTO2	Post-cladding failure oxide fuel/liquid coolant interaction with fuel/coolant thermal/hydraulics.
PINACLE	Molten metallic fuel relocation and heat transfer prior to cladding failure.
LEVITATE	Post cladding failure oxide and metallic fuel relocation with fuel/cladding heat transfer.
D3IF	Interface to DIF3D for TSPK input data generation or DIF3D-K space/time neutronics.

It is seen that both DEFORM-5 and FPIN2 are used as modules in the SAS4A/SASSYS code package.

The code package has been validated for both relatively fast reactivity inputs characterized by fuel melting and motion, and slower reactivity inputs characteristic of core thermal deformation and slow heating of the core. The first set of experiments was carried out using the TREAT reactor, and involved both oxide and metal-based fuel. The second set of experiments were carried out using the EBR-II and FFTF reactors and involved transients without reactor scram. These included LOF and LOHS accidents, and lasted many hundreds of minutes.

5.5 International Codes

In this section, a selection of relevant codes that have either been developed by non-US institutions or those codes that were initially developed in the US and whose subsequent development was continued at non-US institutions will be outlined. Codes that fall into the first category are GRSIS [31] and MACSIS [32], both of which were developed by the Korea Atomic Energy Institute (KAERI), as part of their fast reactor program.

The GRSIS (Gas Release and Swelling in Isotropic fuel matrix) [38] code assumes that fission gas bubbles nucleate isotropically from gas atoms in the metallic fuel matrix. The metallic matrix assumed in this case is U-Pu-10Zr, since this is the fuel of choice for the KAERI fast reactor design. Gas bubbles in the model can grow as a result of gas atom diffusion and coalescence with other bubbles. Once bubble number density or swelling due to bubble growth reaches a threshold value, they are assumed to interconnect to form open channels to the pellet outer surface. The fission gas is then released to the pellet/clad gap. Any subsequent fission gas generation is assumed to be released to the gap and eventually ends up in the fission gas plenum. The model in GRSIS was validated against experimental data obtained from irradiation tests carried out in EBR-II for the metal fuel alloys of interest.

The MACSIS (Metal fuel performance Analysis code for Simulating the In-reactor behavior under Steady-state conditions) [39] code is used for metal fuel rod design and evaluation of operational limits under irradiation conditions. This code is similar in capability to the LIFE-METAL code developed in the US and described above. Briefly, among the most important parameters determined by this code are fuel temperature distribution, fission gas release and subsequent fuel swelling, interaction of the swelling fuel with the clad, and the resultant fuel rod deformation. The MACSIS MOD1 [40] version of the code has upgraded models for fission gas release and swelling, fuel temperature distribution, and rod deformation. MACSIS MOD1 has been validated against experimental data measured at EBR-II, and against calculations carried out using LIFE-METAL.

The ALFUS (Alloyed Fuel Unified Simulator) [41] code, developed in Japan in support of their fast reactor project is used to investigate the behavior of metal based fuels. This code can mechanistically simulate fission gas release and subsequent deformation of the metal fuel. The stress-strain analysis model recognizes the anisotropic strain introduced by cavity formation along grain or phase boundaries in the α-phase uranium, which results in anisotropic deformation in the U-Zr alloy fuel. Other models included in ALFUS are based on experimental data

measured at EBR-II and other facilities. It has been found that if the smear density is greater than ~ 80% there is a significant probability of clad failure due to FCMI.

A code that falls in the second category, i.e., those that were developed in the US with subsequent developments and improvements at non-US institutions, is the SIMMER code [42, 43]. This code was developed to study the consequences of a hypothetical core disruptive accident in oxide and metal based cores. Although this code is not used for operational fuel behavior analysis, it does estimate the behavior of metal fuel under transient conditions. This code models the same phenomena modeled by SAS4A described above. The exception is that it includes models that estimate the slumping of a damaged core and the energetic disassembly following a possible re-criticality event.

6.0 SUMMARY OF ASSESSMENT

The data base necessary to support the use of metal fuel in any of the proposed SMRs needs to consider the specifics of the particular design proposed by the applicant. Therefore, the initial assessment considers how closely key characteristics of the available data match those of the proposed concepts. Based on the presentations made by 4S, PRISM, and ARC-100 at the kick-off meeting at NRC on August 25, 2010 and data submitted by them in support of this assessment, all three argue that experiments with metallic fuel at EBR-II and FFTF are directly relevant **and sufficient** to qualify the use of metallic fuel in their individual concepts. The "relevancy" is indisputable; the issue is the "adequacy/sufficiency" to support licensing. Table 6.1 compares some of the characteristics of these concepts to those of the fuels that were irradiated in EBR-II and FFTF.

Table 6.1 Comparison of Characteristics of Metallic Fuels

Key Parameter	EBR-II/ FFTF	4S	Standard/ S-PRISM	ARC-100
Peak Burnup, 10^4MWd/t	5.0 – 20	< 5.5	18	14.3
Max. linear power, kW/m	33 – 50	8		25.5
Cladding hotspot temp., ^oC	650	609	572	556
Peak center line temp., ^oC	<700	<630		686
Peak radial fuel temp. difference, ^oC	100 - 250	< 30		104
Cladding fast fluence, n/cm^2	up to 4 x 10^{23}	2 x 10^{23}	3 x 10^{23}	5.06 x 10^{23}
Cladding outer diameter, mm	4.4 - 6.9	14	7.366/7.442	13
Cladding thickness, mm	0.38 – 0.56	1.1	0.559/0.559	
Fuel slug diameter, mm	3.33 – 4.98	10.4	5.41/5.476	
Fuel length, m	0.3 (0.9 in FFTF)	2.5	1.19/1.02	1.5
Plenum/fuel length ratio	0.84 to 1.45	1.08	1.48/1.88	
Inter-Pin Spacing	Wire wrap	Grid spacer	Wire wrap	Wire wrap
Fuel residence time, years	1 - 3	30	5	
Smeared density, %	75	78	75/75	
Nominal Fuel Composition (BOL)	U-10Zr* U-19Pu-10Zr**	U-10Zr*	U-26Pu-10Zr	U-10Zr*
Uranium Enrichment		17% - 19%		10.1% - 17.2%
Clad Material	HT-9,D9, CW-316-SS	HT-9	HT-9/HT-9	HT-9(?)

*Qualified as driver fuel in EBR-II and FFTF;
**Demonstrated burnup capability and experimental fuel in EBR-II and FFTF

As noted in Section 2, the parameters for PRISM are "the best available" based on what was provided by GE-Hitachi. The data presented in the table support the following observations:

- The irradiation conditions (e.g., burnup, linear power, temperature, fluence) in the EBR-II and FFTF irradiations generally bound those of the three concepts.

- The nominal/reference fluence for ARC-100 exceeds the experimental data by 25%.

- The fuel compositions (U-10Zr, U-Pu-10Zr) irradiated in EBR-II and FFTF are fundamentally the same as those proposed. Although the number of U-Pu-10Zr irradiations is significantly lower, only PRISM has indicated that this may be considered sometime in the future but it is not the primary option.

- The peak centerline fuel temperature and cladding hotspot temperatures for all three concepts are lower than those experienced in the earlier irradiations.

- The dimensions of the fuel (diameters of slug and rod, fuel and plenum lengths) for the three concepts show the greatest differences relative to the pins that were irradiated. The 4S, PRISM and ARC-100 fuel rods are all thicker, and generally longer, most notably in the case of the 4S.

The results show that there are both similarities and differences between the fuel proposed for the three SMR concepts and the fuels that were irradiated in EBR-II and FFTF and which represents the only currently available experimental data available for the performance of metallic fuel. Given the difficulty in obtaining similar data specific to any of the SMR designs (cost, time, no domestic fast-spectrum, no domestic SFR, etc.), the challenge, therefore, is to assess the degree to which the earlier irradiations can be considered to be "prototypic" of the fuels proposed for the new SMRs, and adequate/sufficient to support future licensing.

The fuel system safety review that will eventually be performed by NRC will be based on the guidance outlined in Standard Review Plan NUREG-800 Section 4.2, and discussed earlier. This design criterion requires that there be assurance in the plant design that fuel design limits will not be exceeded during normal operation and anticipated operational occurrences. Two limits are implied by this criterion, they are:

1. **Fuel design limits** – temperature, burnup, fluence, and clad strain – considering normal and anticipated duty cycles events, which include load following and run beyond clad breach operation, will be established to ensure a failure rate that is acceptably low.

2. **Fuel damage limits** – clad strain, amount of fuel melting, amount of clad deformation or melting, and fractional fuel failure beyond which accident consequences are unacceptable – will be established from a set of design-basis accidents with consequences ranging from insignificant degradation of expected fuel lifetime to maintenance of a coolable geometry.

As described above, these issues were successfully addressed for the fuels used in EBR-II and FFTF, and U-10Zr was "qualified" as driver fuel for both reactors. In light of the differences between this fuel and those proposed for the SMRs, the optimum and unambiguous approach would be to develop a qualification program similar to that performed for EBR-II and FFTF, or based on the TRL approach described in Section 3 employing fuels that were truly "prototypic" of the new fuel designs to enhance and extend the available data. Given the difficulties with this option as discussed earlier, if the differences are relatively modest, the EBR-II/FFTF operation can be applied to the new fuel designs by extrapolating the data if necessary to the new design using the codes described in Section 5. However, some experiments will still likely be necessary to confirm/validate that any extrapolations are reasonable.

6.1 Conclusions

Based on the discussion outlined above, it is clear that either metal fuel form, U-Zr or U-Pu-Zr, represent a fuel development TRL consistent with a Phase of 2 (Table 2.1) when viewed in the context of the technical maturity of the metallic fuels for the SMRs. Achieving Phases 3 and 4 requires that experimental data be gathered using "prototypic" fuel. Thus, if the proposed reactor has a core that is similar in size to the EBR-II or acceptably close in size, a higher development phase would be possible for U-Zr. In the case of U-Pu-Zr based fuel, a larger data base would be required to qualify for a higher fuel development TRL/Phase value. It should be noted that these fuels are based on fresh feed material at BOL, and thus do not contain any fission product or actinide carry over as would be the case if reprocessing was involved. In addition, the uranium is not from a re-cycling plant and only contains U^{235} and U^{238}.

If the chosen cladding material is D9, then the exposure should be limited to 100 dpa and the operating temperature to 630ºC. However, if HT9 is chosen, then the exposure can be increased to approximately 200 dpa, but the operating temperature must be limited to 560ºC. Note that the 4S and PRISM SMRs explicitly refer to HT9 as their cladding of choice.

The experimental data available from EBR-II and FFTF provides relevant data for qualifying metallic fuel for the 4S, PRISM, and ARC-100 SMRs and supporting the licensing process. However, the differences noted above in Table 6.1 make it difficult to argue that the EBR-II and FFTF data address the "prototypic" objective. This is especially true for the 4S and ARC-100 reactors.

A qualified fuel fabrication process must be established, and it must be demonstrated that the fuel fabricated by that process is "equivalent" to that irradiated in EBR-II/FFTF to support the argument that those irradiations satisfy the "prototypic" objective.

It is assumed that the development of a new fuel type suitable for one of the proposed SMRs will proceed in an evolutionary manner. The starting point for this evolution will be the existing data base and parameters that are a reasonable extension of those used in the EBR-II. A series of parameters that are extrapolations of the EBR-II experience are given in Table 6.2. These parameters are deemed close enough to the experience base that they can form a reasonable starting point for a detailed study that is specific to a given small modular reactor application [7]. Alternatively, the designers/vendors might delay introduction of their reactor concept until such

a time that the knowledge base concerning fuel performance evolves sufficiently. This may occur as a result of activities under the DOE-NE Fuel Cycle Technologies program where metallic fuels are actively being studied, including irradiation experiments and first principles modeling that may allow extrapolation with higher confidence over a broader range of parameters. International activities may also provide relevant information, although the near-term focus in most countries is on oxide fuel.

Ex-core experiments for multiple rod bundles or fuel assemblies would be needed to address mechanical and thermal-hydraulic issues to provide the requisite assurances for the 4S given its significantly longer fuel length and use of grid spacers instead of more conventional wirewrap. Data from these experiments will be used to validate thermal-hydraulic codes that are to be used in the licensing process.

In order to develop the safety case, the issues that each concept will need to address for its specific design include:

- Behavior of and extent of fuel restructuring and porosity characteristics of the proposed new fuel alloy as a function of burnup and temperature needs to be confirmed,

- The behavior of prototypical fuel at requisite burnup in appropriate geometry needs to be established. This applies primarily to fuel rods that are longer or have a larger diameter than those used in EBR-II and FFTF,

- Behavior of molten fuel during a power excursion, particularly the extrusion mechanism, needs to be verified by testing. This verification is particularly necessary if different cladding or fuel alloys are proposed than those used in the data base,

- A statistical data base to support the reliability requirements for fuel of prototypic geometry,

- The conclusions drawn from experiments carried out on EBR-II under “slow” transient conditions need to be verified in experiments with fuel assemblies of prototypic geometry, and in addition, TOP transients need to be carried out, and

- The ability to run-beyond-clad-breach needs to be confirmed for the new fuel configuration.

In summary, the data available at the present time regarding the performance/behavior of metal fuels in SFRs is limited primarily to the EBR-II. A limited experience base exists for fuel used in the FFTF. Based on this general observation and the above conclusions, it is recommended that unless the proposed reactor has the same dimensions and operating parameters as the EBR-II, additional work will be required to make a convincing safety case. The amount of additional work required will depend on the magnitude of the deviation from the EBR-II configuration. If the proposed SFR had a configuration covered by the design parameters shown in Table 6.2, a “modest” amount of work would be required. This is because the fuel composition is the same as that used in the EBR-II, and the primary differences are modest dimensional changes. However, deviations beyond those shown in Table 6.2 that would not allow extrapolation with the requisite confidence would imply a “significant” amount of work required to license the fuel.

This conclusion is fundamentally consistent with that reached by a panel of experts convened under the DOE-NE Advanced Fuel Cycle Research and Development program to identify research need for fuels and materials for SFRs [28]. In contrast to the present assessment, the panel did not consider any specific SFR concept. The primary conclusions of the panel regarding the ability to license a SFR using metallic fuel are summarized below:

> "The main conclusion reached by the panel was that an SFR could be designed and licensed based upon the technology base developed from the successful operation of EBR-II and FFTF. However, the design would be constrained within the limitations of the technology base. From a fuels and materials perspective, these limitations are the following:

1. For metal fuel a maximum burnup of 10 at%.
2. A peak cladding temperature of 600ºC.
3. The use of D9 stainless steel cladding and duct material.
4. A peak irradiation exposure of 100 dpa on the cladding and duct.
5. The use of fresh fuel (fuel with neither additions of minor actinides or fission product carry-over from reprocessing)."

Alternatively, if the proposed fuel is such that it is judged as not being adequately covered by the existing data base, and the analytic techniques are not appropriately validated to allow extrapolation to the set of parameter values descriptive of the proposed fuel with the required level of confidence, a more extensive development program may be required. For example, an approach might be to designate the first unit as a "test unit" with instrumented fuel rod/assemblies, and the capability to allow for periodic fuel retrieval for PIE purposes, i.e., "license by operation." This possibility has been recognized in the SRP (Appendix A) and discussed in Section 3 where a "… *special detailed surveillance program should be planned for the first irradiation of a new design"* if true prototype testing is not feasible.

Table 6.2 Enveloping Design Parameters for Metal Fuel from the Existing Data Base.

Parameter	Reference Value
Nominal composition	U-10Zr, U-20Pu-10Zr
Pu/(U+Pu) range	17 – 28 %
Theoretical density	100 %
Smear density (% of TD)	75%
Plenum-to-fuel volume	1.4
Fuel height (cm)	91
Fuel outer diameter (cm)	0.5
Fuel-clad bond	Na
Clad material	HT-9 or 20% cw 316SS D9 if burnup is limited to <10 at%
Clad outer diameter (cm)	0.69
Clad inner diameter (cm)	0.57
Peak linear heat rate kW/m	49 – 52
Peak inner clad wall temperature (ºC)*	620 for D9 560 for HT9
Duct material	HT-9 or 20% cw 316SS
Peak Clad Fast Fluence (n/cm^2)	4×10^{23}
Maximum Burnup (at%)*	10 if D9 ~20 if HT9
Peak Clad Exposure (dpa)*	100 if D9 ~200 if HT9

* See text regarding constraints on D9 and HT9 with respect to maximum burnup, peak clad exposure, and maximum temperature

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APPENDIX A

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
SECTION 4.2, FUEL SYSTEM DESIGN

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

4.2 FUEL SYSTEM DESIGN

REVIEW RESPONSIBILITIES

Primary - The organization responsible for the review of transient and accident analyses

Secondary - None

I. AREAS OF REVIEW

The organization responsible for the review of transient and accident analyses evaluates the thermal, mechanical, and materials design of the fuel system. The fuel system consists of arrays (assemblies or bundles) of fuel rods, including fuel pellets, insulator pellets, springs, tubular cladding, end closures, hydrogen getters, and fill gas; burnable poison rods including components similar to those in fuel rods; spacer grids and springs; end plates; channel boxes; and reactivity control rods. This section discusses the reactivity control elements of the control rods that extend from the coupling interface of the control rod drive mechanism into the core.

The fuel system safety review provides assurance that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs), (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. General Design Criterion (GDC) 10, within Appendix A to 10 CFR Part 50, also addresses item 1 above. Specifically, GDC 10 establishes specified acceptable fuel design limits (SAFDLs) that should not be exceeded during any condition of normal operation, including the effects of AOOs. Therefore, the SAFDLs are established to ensure that the fuel is

not damaged. Within this context, “not damaged” means that the fuel rods do not fail, fuel system dimensions remain within operational tolerances, and functional capabilities are not reduced below those assumed in the safety analysis. The design limits of GDC 10 (i.e., the SAFDLs) accomplish these objectives. In a “fuel rod failure,” the fuel rod leaks and the first fission product barrier (the cladding) is breached. The dose analysis required by 10 CFR Part 100 for postulated accidents must account for fuel rod failures. “Coolability,” in general, means that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat even after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the GDC found in Appendix A to 10 CFR Part 50 (e.g., GDC 27 and 35). In particular, 10 CFR 50.46 provides the specific coolability requirements for the loss-of-coolant accident (LOCA).

Standard Review Plan (SRP) Section 4.2 describes all fuel damage criteria. SRP Section 4.4 provides specific thermal-hydraulic criteria for instances involving limits to the departure from nucleate boiling ratio (DNBR) and the critical power ratio (CPR). The available radioactive fission product inventory in fuel rods (i.e., the gap inventory expressed as a release fraction) is provided to the U.S. Nuclear Regulatory Commission’s (NRC) organization that is responsible for the review of design basis accident radiological consequence analyses for use in estimating the radiological consequences of plant releases.

The specific areas of review are as follows:

1. Design Bases. Design bases for the safety analysis address fuel system damage mechanisms and provide limiting values for important parameters to prevent damage from exceeding acceptable levels. The design bases should reflect the safety review objectives as described above.

   The reviewer should evaluate established (past) design-basis limits and associated SAFDLs to determine whether they remain applicable to the new fuel design (including the introduction of new materials) given the operating conditions (temperature, burnup, and power). If they do not apply, new limits must be established based on appropriate data.

2. Description and Design Drawings. The reviewer examines the fuel system description and design drawings. In general, the description will emphasize product specifications rather than process specifications.

3. Design Evaluation. The reviewer evaluates the performance of the fuel system during normal operation, AOOs, and postulated accidents to determine whether all design bases are met. The fuel system components, as listed above, are reviewed not only as separate components but also as integral units such as fuel rods and fuel assemblies. New fuel designs, new operating limits (e.g., rod burnup and power), and the introduction of new materials to the fuel system require a review to verify that existing design-basis limits, analytical models, and evaluation methods remain applicable for the specific design for normal operation, AOOs, and postulated accidents. The review also evaluates operating experience, direct experimental comparisons, detailed mathematical analyses (including fuel performance codes), and other information.

4. Testing, Inspection, and Surveillance Plans. The licensee performs testing and inspection of new fuel to ensure that the fuel is fabricated in accordance with the design and that it reaches the plant site and is loaded in the core without damage. Online fuel rod failure monitoring and postirradiation surveillance should be performed to detect anomalies or confirm that the fuel system is performing as expected; surveillance of control rods containing B_4C should be performed to preclude reactivity loss. The organization responsible for reactor systems reviews the testing, inspection, and surveillance plans, along with their reporting provisions, to ensure that the important fuel design considerations have been addressed.

5. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this SRP section in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against acceptance criteria contained in this SRP section. Furthermore, the staff reviews the ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

6. COL Action Items and Certification Requirements and Restrictions. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

   For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

Review Interfaces

SRP Section 4.2 describes all fuel damage criteria. SRP Section 4.3 establishes fuel criteria for axial offset anomaly (AOA). For those criteria that involve DNBR or CPR limits, SRP Section 4.4 provides specific thermal-hydraulic criteria. The available radioactive fission product inventory in fuel rods (i.e., the gap inventory expressed as a release fraction) is provided to those organizations that estimate the radiological consequences of plant releases in accordance with SRP Chapter 15. Fuel stored energy, flow blockage, peak cladding temperature, and equivalent cladding reacted (ECR) limits defined in SRP Section 4.2 are provided to those organizations that review Chapter 15.

Other SRP sections interface with this section as follows:

1. Review of the nuclear design of the fuel assemblies, control systems, and reactor core under SRP Section 4.3.

2. Review of the thermal margins, the effects of corrosion products (crud), and the acceptability of hydraulic loads under SRP Section 4.4.

3. Review of the design bases for the emergency core cooling system (ECCS), including GDC and ECCS acceptance criteria, under SRP Section 6.3.

4. Review of the postulated fuel failures resulting from overheating of cladding, overheating of fuel pellets, excessive fuel enthalpy, pellet/cladding interaction (PCI), and bursting under Chapter 15.

5. Review of the control rod drive mechanism design in SRP Section 3.9.4 and the reactor internals design under SRP Section 3.9.5.

6. Review of the estimates of radiological dose consequences under Chapter 15.

The specific acceptance criteria and review procedures are contained in the referenced SRP sections.

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. 10 CFR 50.46, 10 CFR 50.34, and 10 CFR 50.67, as they relate to the cooling performance analysis of the ECCS using an acceptable evaluation model and establishing acceptance criteria for light-water nuclear power reactor ECCSs.

2. 10 CFR Part 100 and 10 CFR 50.67, as they relate to determining the acceptability of a reactor site based on calculating the exposure to an individual as a result of fission product releases to the environment following a major accident scenario.

3. GDC 10, as it relates to assuring that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs.

4. GDC 27, as it relates to the reactivity control system being designed with appropriate margin and, in conjunction with the ECCS, being capable of controlling reactivity and cooling the core under postaccident conditions.

5. GDC 35, as it relates to providing an ECCS to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

6. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;

7. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for the review described in this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

Specific criteria necessary to meet the relevant requirements of 10 CFR 50.46; GDC 10, 27, and 35; Appendix K to 10 CFR Part 50; and 10 CFR Part 100 are as follows:

1. Design Bases

   The fuel system design bases must reflect the four objectives described in Subsection I, Areas of Review. To satisfy these objectives, acceptance criteria are needed for fuel system damage, fuel rod failure, and fuel coolability. These criteria are discussed in the following paragraphs:

   A. Fuel System Damage

      This subsection applies to normal operation, and Section 4.2 of the safety analysis report should contain the information to be reviewed.

      To meet the requirements of GDC 10, as it relates to SAFDLs for normal operation, including AOOs, fuel system damage criteria should be included for all known damage mechanisms.

      Fuel damage criteria should assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. When applicable, the fuel damage criteria should consider high burnup effects based on irradiated material properties data. Complete damage criteria should address the following:

      i. Stress, strain, or loading limits for spacer grids, guide tubes, thimbles, fuel rods, control rods, channel boxes, and other fuel system structural members should be provided. Stress limits that are obtained by methods similar to those given in Section III of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME) are acceptable. Other proposed limits must be justified.

ii. The cumulative number of strain fatigue cycles on the structural members mentioned in item (i) above should be significantly less than the design fatigue lifetime, which is based on appropriate data and includes a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles. Other proposed limits must be justified.

iii. Fretting wear at contact points on the structural members mentioned in item (i) above should be limited. Fretting wear tests and analyses that demonstrate compliance with this design basis should account for grid spacer spring relaxation. The allowable fretting wear should be stated in the safety analysis report, and the stress and fatigue limits in items (i) and (ii) above should presume the existence of this wear.

iv. Oxidation, hydriding, and the buildup of corrosion products (crud) should be limited, with a limit specified for each fuel system component. These limits should be established based on mechanical testing to demonstrate that each component maintains acceptable strength and ductility. The safety analysis report should discuss allowable oxidation, hydriding, and crud levels and demonstrate their acceptability. These levels should be presumed to exist in items (i) and (ii) above. The effect of crud on thermal-hydraulic considerations and neutronic (AOA) considerations are reviewed as described in SRP Sections 4.3 and 4.4.

v. Dimensional changes, such as rod bowing or irradiation growth of fuel rods, fuel assemblies, control rods, and guide tubes, should be limited to prevent fuel failures or a situation in which the thermal-hydraulic limits established in Section 4.4 are exceeded. Irradiation growth can result in a significant interference fit between the rod upper end cap and the tie plate (in a boiling-water reactor (BWR)) or the upper nozzle (in a pressurized-water reactor (PWR)), resulting in rod bowing.

Control blade/rod, channel, and guide tube bow as a result of (1) differential irradiation growth (from fluence gradients), (2) shadow corrosion (hydrogen uptake results in swelling), and (3) stress relaxation, which can impact control blade/rod insertability from interference problems between these components. For BWRs, the effects of shadow corrosion should be considered for new control blade or channel designs, dimensions (e.g., the distance between control blade and channel is important), or materials. The effects of channel bulge should also be considered for interference problems for BWRs. Design changes can alter the pressure drop across the channel wall, thus necessitating an evaluation of such changes. Channel material changes can also impact the differential growth, stress relaxation, and the amount of bulge and therefore must be evaluated. If interference is determined to be possible, tests are needed to demonstrate control blade/rod insertability consistent with assumptions in safety analyses. Additional in-reactor surveillance (e.g., insertion times) may also be necessary for new designs, dimensions, and materials to demonstrate satisfactory performance.

vi. Fuel and burnable poison rod internal gas pressures should remain below the nominal system pressure during normal operation or other limits must be justified based on, but not limited to, the following minimum criteria.

(1) No cladding liftoff during normal operation

(2) No reorientation of the hydrides in the radial direction in the cladding

(3) A description of any additional failures resulting from departure of nucleate boiling (DNB) caused by fuel rod overpressure during transients and postulated accidents (see Subsection II, item 1.B.vii)

vii. Because unseating a fuel bundle may challenge control rod/blade insertion, an evaluation of worst-case hydraulic loads should be performed for normal operation, AOOs, and accidents. These worst-case hydraulic loads for normal operation should not exceed the holddown capability of the fuel assembly (either gravity or holddown springs). Hydraulic loads for this evaluation are reviewed as described in SRP Section 4.4.

viii. Control rod reactivity and insertability must be maintained. This requires that, at a minimum, the following may need to be reviewed:

(1) Changes in control rod configuration

(2) Introduction of new materials

(3) Changes in neutronics and mechanical lifetime

(4) Changes in mechanical design

(5) The ability to exclude water/coolant if water-soluble or leachable materials (e.g., B_4C) are used

Changes in mechanical and neutronics lifetimes need to be calculated using acceptable methods. Safety analyses must specifically account for the reduction in neutron-absorbing capabilities with time in-reactor.

B. <u>Fuel Rod Failure</u>

This subsection applies to normal operation, AOOs, and postulated accidents. Items 1.B.i through 1.B.iii below address failure mechanisms that are more limiting during normal operation; Section 4.2 of the safety analysis report should contain the information to be reviewed. Items 1.B.iv through 1.B.viii below address failure mechanisms that are more limiting during AOOs and postulated accidents; Chapter 15 of the safety analysis report usually contains the information to be reviewed.

To meet the requirements of (1) GDC 10 as it relates to SAFDLs for normal operation, including AOOs and (2) 10 CFR Part 100 as it relates to fission product releases for postulated accidents, fuel rod failure criteria should be provided for all known fuel rod failure mechanisms. Fuel rod failure is defined as the loss of fuel rod hermeticity. Although the staff recognizes that it is impossible to avoid all fuel rod failures and that cleanup systems are installed to handle a small number of leaking rods, the review must ensure that fuel does not fail as a result of specific causes during normal operation and AOOs. Fuel rod failures are permitted during postulated accidents, but they must be accounted for in the dose analysis.

Fuel rod failures can be caused by overheating, PCI, hydriding, cladding collapse, bursting, mechanical fracturing, and fretting. When applicable, the fuel rod failure criteria should consider high burnup effects based on irradiated material properties data.

Complete fuel failure criteria should address the following:

i. Hydriding. Both internal and external sources of hydriding can cause a zirconium alloy component to fail. To prevent failure from internal hydriding (i.e., primary hydriding), the level of moisture and other hydrogenous impurities within the fuel is kept very low during fabrication. Acceptable moisture levels for Zircaloy-clad uranium oxide fuel should be no greater than 20 micrograms per gram (μg/g) (20 parts per million (ppm)). Current specifications of the American Society for Testing and Materials (ASTM),1989 edition, Standard C776-89, Part 45, for uranium oxide fuel pellets state an equivalent limit of 2 μg/g (2 ppm) of hydrogen from all sources. For other materials clad in Zircaloy tubing, an equivalent quantity of moisture or hydrogen can be tolerated. A moisture level of 2 milligrams of water per cubic centimeter of hot void volume within the Zircaloy cladding has been shown to be insufficient for primary hydride formation. External hydriding is caused by waterside corrosion in which the water reaction with the zirconium alloy results in zirconium hydrides as well as zirconium dioxide.

ii. Cladding Collapse. If axial gaps in the fuel pellet column result from densification, the cladding has the potential to collapse into a gap (i.e., flattening). Because of the large local strains that accompany this process, collapsed (flattened) cladding is assumed to fail.

iii. Overheating of Cladding. Traditional practice assumes that failures will not occur if the thermal margin criteria (DNBR for PWRs and CPR for BWRs) are satisfied. SRP Section 4.4 details the review of these criteria. Violation of the thermal margin criteria is not permitted for normal operation and AOOs. For postulated accidents, the total number of fuel rods that exceed the criteria has been assumed to fail for radiological dose calculation purposes. Although a thermal margin criterion is sufficient to demonstrate that overheating from a deficient cooling mechanism can be avoided, it is not a necessary condition (i.e., DNB is

not a failure mechanism) and other mechanistic methods may be acceptable. At present, there is little experience with other approaches, but new positions recommending different criteria should address cladding temperature, pressure, time duration, oxidation, and embrittlement.

iv. Overheating of Fuel Pellets. Traditional practice has also assumed that failure will occur if centerline melting takes place. This analysis should be performed for the maximum linear heat generation rate anywhere in the core, including all hot spots and hot channel factors, and should account for the effects of burnup and composition on the melting point. For normal operation and AOOs, centerline melting is not permitted. For postulated accidents, the total number of rods that experience centerline melting should be assumed to fail for radiological dose calculation purposes. The centerline melting criterion was established to assure that axial or radial relocation of molten fuel would neither allow molten fuel to contact the cladding nor produce local hot spots. The assumption that centerline melting results in fuel failure is conservative.

v. Excessive Fuel Enthalpy. The sudden increase in fuel enthalpy from a reactivity initiated accident (RIA) below fuel melting can result in fuel failure due to pellet/cladding mechanical interaction (PCMI) (see Subsection II, item 1.B.vii). Exceeding the DNBR for a PWR or the CPR for a BWR may result in cladding failure during an RIA. See Appendix B for criteria.

vi. Pellet/Cladding Interaction. No criterion currently exists for fuel failure resulting from PCI or PCMI. The difference between PCI and PCMI is subtle, and it is sometimes difficult to differentiate the two types of failures from visual observation of the failure. PCI is generally caused by stress-corrosion cracking due to fission product (iodine) embrittlement of the cladding, while PCMI is primarily a stress-driven failure. The design basis for PCI and PCMI can only be generally stated.

Two related criteria should be applied, but they are not sufficient to preclude PCI or PCMI failures. The first criterion limits uniform strain of the cladding to no more than 1 percent. In this context, uniform strain (elastic and inelastic) is defined as transient-induced deformation with gauge lengths corresponding to cladding dimensions; steady-state creepdown and irradiation growth are excluded. Mechanical testing must demonstrate that the irradiated cladding ductility at maximum waterside corrosion (hydride embrittlement) is well within the 1-percent strain criterion. Although observing this strain limit may preclude some PCI and PCMI failures, it will neither preclude the corrosion-assisted failures that occur at low strains nor the highly localized overstrain failures introduced by pellet chips on the outer fuel diameter. The second criterion states that fuel melting should be avoided. The large volume increase

associated with melting may cause a pellet with a molten center to exert a stress on the cladding. Avoiding fuel melting can preclude such a PCI. Note that item 1.B.iv above invoked this same criterion to ensure that overheating of the cladding would not occur.

Fuel vendors have introduced fuel design limits on power maneuvering and rate of power ascension to prevent PCI or PCMI. These design limits have primarily been based on power ramp data from test reactors for a specific fuel design. Recently, however, fuel vendors have been relying more on their predictions of cladding strain and less on their power ramp data to verify that PCMI will not occur. Convincing evidence exists that gaseous swelling and fuel thermal expansion is responsible for cladding strains at high burnup levels and perhaps at even moderate burnups. Therefore, PCI or PCMI analyses of cladding strain for AOO transients and accidents should apply approved fuel thermal expansion and gaseous fuel swelling models, as well as irradiated cladding properties.

vii. Bursting. To meet the requirements of 10 CFR 50.46, as it relates to ECCS performance evaluation, the ECCS evaluation model should include a calculation of the swelling and rupture of the cladding resulting from the temperature distribution in the cladding and from pressure differences between the inside and outside of the cladding. Regulatory Guide (RG) 1.157 provides guidelines for performing a realistic (i.e., best estimate) model to calculate the degree of cladding swelling and rupture. Alternatively, Appendix K to 10 CFR Part 50 presents the acceptable features of an evaluation model for predicting the degree of swelling and rupture in the Zircaloy cladding. Although fuel suppliers may use different rupture-temperature vs. differential-pressure curves, an acceptable curve should be similar to the one described in NUREG-0630 based on similar data for a specific material. Cladding burst from non-LOCA accidents also needs to be evaluated and addressed in terms of impact on cladding temperatures and radiological consequences.

viii. Mechanical Fracturing. A mechanical fracture refers to a defect in a fuel rod caused by an externally applied force such as a hydraulic load or a load derived from core-plate motion. Cladding integrity may be assumed if the applied stress is less than 90 percent of the irradiated yield stress at the appropriate temperature. Other proposed limits must be justified. Results from the seismic and LOCA analysis (see Appendix A to this SRP section) may show that failures by this mechanism will not occur for less severe events.

C. Fuel Coolability

This subsection applies to postulated accidents, and Chapter 15 of the safety analysis report will contain most of the information to be reviewed. Item 1.C.v below addresses the combined effects of two accidents, and Section 4.2 of the safety analysis report should include that information. To meet the requirements of GDC 27 and 35 as they relate to control rod insertability and core coolability

for postulated accidents, fuel coolability criteria should be provided for all severe damage mechanisms. Coolability, or coolable geometry, has traditionally implied that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat. Reduction of coolability can result from cladding embrittlement, violent expulsion of fuel, generalized cladding melting, gross structural deformation, and extreme coplanar fuel rod ballooning. This subsection also addresses control rod insertability criteria. Complete criteria should address the following:

i. <u>Cladding Embrittlement</u>. The ECCS performance analysis must satisfy the fuel design criteria specified within 10 CFR 50.46(b). These criteria ensure a coolable core geometry by preserving adequate postquench ductility in the fuel rod cladding. The current criteria require that (1) the peak cladding temperature remains below 2200 °F and (2) the peak cladding oxidation remains below 17 percent ECR. These criteria were originally developed on the basis of unirradiated Zircaloy test specimens. Zirconium alloy composition, manufacturing process, and in-reactor corrosion alter the postquench characteristics of the fuel cladding material. Rulemaking pursuant to 10 CFR 50.46 is planned to implement a performance-based test program that will dictate postquench performance requirements and provide an acceptable means to establish specific limits for new cladding materials. Future cladding alloys must comply with the postquench performance requirements specified by the new rule and provide the empirical database to support any limits assigned to the new alloy.

ii. <u>Violent Expulsion of Fuel</u>. In severe RIAs, such as rod ejection in a PWR or rod drop in a BWR, the large and rapid deposition of energy in the fuel can result in melting, fragmentation, and dispersal of fuel. The mechanical action associated with fuel dispersal can be sufficient to destroy the cladding and the rod-bundle geometry of the fuel and produce pressure pulses in the primary system. (See Appendix B for criteria.)

iii. <u>Generalized Cladding Melting</u>. Generalized (i.e., nonlocal) melting of the cladding could result in the loss of rod-bundle fuel geometry. Criteria for cladding embrittlement in item 1.C.i above are more stringent than melting criteria. Therefore, additional specific criteria are not used. However, this may not always be the case for newer alloys or reactor types.

iv. <u>Fuel Rod Ballooning</u>. To meet the requirements of 10 CFR 50.46 as it relates to ECCS performance during accidents, the analysis of the core flow distribution must account for burst strain and flow blockage caused by ballooning (swelling) of the cladding. RG 1.157 describes acceptable models, correlations, data, and methods that can be used to meet the requirements for a realistic calculation of ECCS performance during a LOCA. Alternatively, Appendix K to 10 CFR Part 50 outlines the acceptable features of a conservative evaluation model to consider burst

strain and flow blockage. Burst strain and flow blockage models must be based on applicable data to (1) properly estimate the temperature and differential pressure at which the cladding will rupture (see item 1.B.vii above), (2) avoid underestimating the resultant degree of cladding swelling, and (3) avoid underestimating the associated reduction in assembly flow area.

The flow blockage model evaluation is provided to the organization responsible for the review of transient and accident analyses for incorporation in the comprehensive ECCS evaluation model to demonstrate that the criteria in 10 CFR 50.46(b) are not exceeded. The reviewer also determines whether the analysis of AOOs and other accidents should include fuel rod ballooning. The possibility of ballooning during an AOO transient or accident increases as the fuel rod pressure exceeds the system pressure. Those non-LOCA accidents that result in clad ballooning should examine the possibility of DNB propagation resulting from ballooning. The impact of ballooning on non-LOCA accidents should not be underestimated. A limit on ballooning (circumferential strain) may be required to prevent DNB propagation for these accidents.

v. Structural Deformation. Appendix A discusses the applicable analytical procedures.

2. Description and Design Drawings

The reviewer determines that the fuel system description and design drawings provide an accurate representation and supply the information needed in audit evaluations. Completeness is a matter of judgment, but the following fuel system information and associated tolerances are necessary for an acceptable fuel system description:

- Type and metallurgical state of the cladding
- Cladding outside diameter
- Cladding inside diameter
- Cladding inside roughness
- Pellet outside diameter
- Pellet roughness
- Pellet density
- Pellet resintering data
- Pellet length

- Pellet dish dimensions
- Pellet grain size and open porosity
- Burnable poison content
- Insulator pellet parameters
- Fuel column length
- Overall rod length
- Rod internal void volume
- Fill gas type and pressure
- Sorbed gas composition and content
- Spring and plug dimensions
- Fissile enrichment
- Equivalent hydraulic diameter
- Coolant pressure
- Design-specific burnup limit
- Control blade/rod descriptions, dimensions, and lifetime limits
- Fit of control blade/rod interference with surrounding structure (e.g., channel box or guide tube)

The following design drawings and dimensions are also necessary for an acceptable fuel system description:

- Fuel assembly cross section
- Fuel assembly outline
- Fuel rod schematic
- Spacer grid cross section
- Guide tube and nozzle joint
- Guide tube with respect to control rod dimensions
- Control blade/rod assembly cross section
- Control rod assembly outline
- Control rod schematic
- Burnable poison rod assembly cross section
- Burnable poison rod assembly outline
- Burnable poison rod schematic
- Orifice and source assembly outline

3. Design Evaluation

The reviewer will evaluate the methods for demonstrating that the design bases are met. Methods include operating experience, prototype testing, and analytical predictions. Many of these methods will be presented generically in topical reports and will be incorporated in the safety analysis report by reference.

A. Operating Experience

Operating experience with fuel systems of the same or similar design should be described, including the maximum burnup experience. When adherence to specific design criteria can be conclusively demonstrated with operating experience, prototype testing and design analyses that were performed before gaining that experience need not be reviewed. Design criteria for fretting wear, oxidation, hydriding, and crud buildup might be addressed in this manner.

B. Prototype Testing

When conclusive operating experience is not available, as with the introduction of a design change, prototype testing should be reviewed. Out-of-reactor tests should be performed, when practical, to determine the characteristics of the new design. No definitive requirements have been developed regarding those design features that must be tested before irradiation, but the following out-of-reactor tests have been performed for this purpose and will serve as a guide to the reviewer:

- Spacer grid structural tests
- Control rod structural and performance tests
- Fuel assembly structural tests (lateral, axial and torsional stiffness, frequency, and damping)
- Fuel assembly hydraulic flow tests (lift forces, control rod wear, vibration, fuel rod fretting (should account for spacer spring relaxation), and assembly wear and life)

In-reactor testing of design features and lead-assembly irradiation of whole assemblies of a new design should be reviewed. The maximum burnup or fluence experience associated with such tests should also be reviewed and considered in relation to the specified maximum burnup or fluence limit for the new design. The following phenomena have been tested in this manner in new designs and will serve as a guide to the reviewer:

- Fuel and burnable poison rod growth
- Fuel rod bowing
- Fuel rod, spacer grid, and channel box oxidation and hydride levels
- Fuel rod fretting
- Fuel assembly growth
- Fuel assembly bowing
- Channel box wear and distortion
- Fuel rod ridging (PCI)

- Crud formation
- Fuel rod integrity
- Holddown spring relaxation
- Spacer grid spring relaxation
- Guide tube wear characteristics

In some cases, in-reactor testing of a new fuel assembly design or a new design feature cannot be accomplished before operation of the design's full core. The inability to perform in-reactor testing may result from an incompatibility of the new design with the previous design. In such cases, special attention should be given to the surveillance plans (see Subsection II.4 below).

C. Analytical Predictions

Some design bases and related parameters can only be evaluated with calculational procedures. The analytical methods that are used to make performance predictions must be reviewed. Many such reviews have been performed establishing numerous examples for the reviewer. The following paragraphs discuss the more established review patterns and provide many related references.

i. Fuel Temperatures (Stored Energy). Fuel temperatures and stored energy during normal operation serve as input to ECCS performance calculations. Temperature calculations require complex computer codes that model many different phenomena. RG 1.157 describes models, correlations, data, and methods to realistically calculate ECCS performance during a LOCA and to estimate the uncertainty in that calculation. Alternatively, an ECCS evaluation model may be developed in conformance with the acceptable features of Appendix K to 10 CFR Part 50. Phenomenological models that should be reviewed include the following:

  - Radial power distribution
  - Fuel and cladding temperature distribution
  - Burnup distribution in the fuel
  - Thermal conductivity of the fuel, cladding, cladding crud, and oxidation layers
  - Densification of the fuel
  - Thermal expansion of the fuel and cladding
  - Fission gas production and release
  - Solid and gaseous fission product swelling
  - Fuel restructuring and relocation

- Fuel and cladding dimensional changes
- Fuel-to-cladding heat transfer coefficient
- Thermal conductivity of the gas mixture
- Thermal conductivity in the Knudsen domain
- Fuel-to-cladding contact pressure
- Heat capacity of the fuel and cladding
- Growth and creep of the cladding
- Rod internal gas pressure and composition
- Sorption of helium and other fill gases
- Cladding oxide and crud layer thickness
- Cladding-to-coolant heat transfer coefficient
- Cladding hydriding

Because of the strong interaction between these models, overall code behavior should be checked against data (standard problems or benchmarks) and the NRC audit codes. NUREG/CR-6534 (PNNL-11513) Vol. 2, December 1997, FRAPCON-3 NUREG/CR-6534 (PNNL-11513) Vol. 4, May 2005, Babcox & Wilcox Report BAW-10087A, Rev. 1, August 1977, CENPD-139-A, July 1974, Supplement 1 to Technical Report on General Electric Reactor Fuels, December 14, 1973, Technical Report on Exxon Nuclear PWR Fuels, February 27, 1975, and the Letter on Westinghouse Safety Evaluation of WCAP-8720, February 9, 1979, provide examples of previous fuel performance code reviews.

ii. Densification Effects. In addition to its effect on fuel temperatures (discussed above), densification affects (1) core power distributions (power spiking - see SRP Section 4.3), (2) the fuel linear heat generation rate (LHGR) - see SRP Section 4.4, and (3) the potential for cladding collapse. NUREG-0085 and RG 1.126 discuss densification magnitudes for power spike and LHGR analyses. To be acceptable, densification models should follow the guidelines of RG 1.126. Models for cladding-collapse times should also be reviewed. The memorandums on Evaluation of Westinghouse Report, WCAP-8377, January 14, 1975 and on CEPAN-Method of Analyzing Creep Collapse of Oval Cladding, February 5, 1976, provide previous review examples.

iii. Fuel Rod Bowing. The memorandum on Request for Revised Rod Bowing Topical Reports, May 30, 1978, includes guidance for the analysis of fuel rod bowing. The memorandum on Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing in Thermal Margin Calculations for Light Water Reactors, February 16, 1977, presents interim methods that may be used. At this writing, the causes of fuel rod bowing are not well understood and mechanistic analyses of rod bowing have not been approved.

iv. Cladding Collapse. Approved analytical models/methods are used to demonstrate that cladding collapse is not possible within the fuel lifetime. A change in cladding or fuel material (additives or significant changes in fabrication) and/or a reduction in the as-fabricated fuel cladding gap can impact the approved analytical model/methods used for this analysis. A change in fuel material can impact fuel densification and a change in cladding material can impact cladding creep, both of which can impact cladding collapse. If any of these parameters change, they must be evaluated in terms of their impact on the approved analytical models and methods for evaluating cladding collapse.

v. Structural Deformation. Appendix A discusses the acceptance criteria.

vi. Rupture and Flow Blockage (Ballooning). The ECCS evaluation model includes Zircaloy rupture and flow blockage models, which should be reviewed by the organization responsible for reactor systems. The models are empirical and should be compared with relevant data. NUREG-0630, NUREG/CR-1883, and the publication on Burst Criterion of Zircaloy Fuel Cladding in a LOCA, August 4-7, 1980, provide examples of such data and previous reviews. These models should account for the phase transformation in the cladding at high temperatures.

vii. Fuel Rod Pressure. The thermal performance code for calculating temperatures discussed in item 3.C.i above should be used to calculate fuel rod pressures in conformance with the fuel damage criteria of item 1.A.vi in Subsection II. This calculation should account for uncertainties in the estimated rod powers, code models, and fuel rod fabrication. The reviewer should ensure that conservatisms that were incorporated for calculating temperatures do not introduce nonconservatisms with regard to fuel rod pressures.

viii. Metal/Water Reaction Rate. To meet the requirements of 10 CFR 50.46(b) as it relates to the performance of the ECCS during accidents, the rate of energy release, hydrogen generation, and cladding oxidation resulting from the reaction of the Zircaloy cladding with steam should be calculated. Currently this can be calculated in two ways. RG 1.157 allows the use of a best-estimate model, provided its technical basis is demonstrated with appropriate data and analyses. Alternatively, Appendix K to 10 CFR Part 50 specifies that the rate of energy release, hydrogen generation, and cladding oxidation from the metal/water

reaction should be calculated using the Baker-Just equation (Argonne National Laboratory Report ANL-6548, May 1962). For non-LOCA applications, other correlations may be used if justified. These reaction rate models were originally developed based upon unirradiated Zircaloy test specimens. Zirconium alloy composition, manufacturing process, and in-reactor corrosion alter the reaction rate characteristics of the fuel cladding material. Rulemaking pursuant to 10 CFR 50.46 is planned to implement a performance-based test program to provide an acceptable means for establishing specific reaction rate models for new cladding materials. Future cladding alloys must comply with the new rules and the need to provide an empirical database to support applicable reaction rate models.

ix. Fission Product Inventory. The assumptions in RG 1.3, RG 1.4, RG 1.5, RG 1.25, RG 1.77, RG 1.195, and RG 1.196, as they relate to fission product release for existing reactors (i.e., DC applications before January 10, 1997), currently specify the available radioactive fission product inventory in fuel rods (i.e., the gap inventory). RG 1.195 and RG 1.196 can be used in place of RG 1.3, RG 1.4, RG 1.5, RG 1.25, and RG 1.77. RG 1.183 and the requirements of 10 CFR 50.34 apply to fission product release for new reactors. An alternate source term (AST), specified in 10 CFR 50.67, can be applied to existing reactors as an alternative to 10 CFR Part 100 as defined in these documents. American Nuclear Society (ANS) 5.4 presents an approved method for release during non-LOCAs and situations that do not involve accidents in which the fuel temperature exceeds the temperature experienced during normal operation and AOOs. ANS 5.4 also provides an acceptable analytical model for calculating the release of volatile fission products from oxide fuel pellets during normal steady-state conditions. When used with nuclide yields, this model will define the inventory of volatile fission products that could be available for release from the fuel rod if the cladding were breached, sometimes referred to as gap inventory. Recent experimental data from RIA tests in Nuclear Safety Research Reactor (NSRR) and Cabri (Publication on NSRR/RIA Experiments with High Burnup PWR Duels, March 2-6, 1997, Publication on High-Burnup BWR Fuel Behavior Under Simulated Reactivity-Initiated Accident Conditions, Nuclear Technology Vol. 38, June 2002, and Publication on The Role of Grain Boundary Fission Gases in High Burn-Up Fuel Under Reactivity Initiated Accident Conditions, September 2000) suggest that the gap inventory for a BWR rod drop accident specified in RG 1.183 and for a PWR control rod ejection accident may need modification. The NRC has plans to issue new guidelines for gap inventory (fission product release) from these accidents.

4. Testing, Inspection, and Surveillance Plans

Plans must be reviewed for each plant for testing and inspection of new fuel and for monitoring and surveillance of irradiated fuel.

A. Testing and Inspection of New Fuel

Testing and inspection plans for new fuel should verify cladding integrity, fuel system dimensions, fuel enrichment, burnable poison concentration, and absorber composition. Quality control reports should document the details of the manufacturer's testing and inspection programs and should be referenced and summarized in the safety analysis report. The program for onsite inspection of new fuel and control assemblies after they have been delivered to the plant should also be described. When the overall testing and inspection programs are essentially the same as those for previously approved plants, a statement to that effect should be made. In that case, the safety analysis report need not include program details, but an appropriate reference should be cited and a summary (tabular) should be presented.

B. Online Fuel System Monitoring

The applicant's online fuel rod failure detection methods should be reviewed. Both the sensitivity of the instruments and the applicant's commitment to use the instruments should be evaluated. NUREG-0401 and NUREG/CR-1380 evaluate several common detection methods and should be used in this review.

Surveillance is also needed to assure that B_4C control rods are not losing reactivity. Boron compounds are susceptible to leaching in the event of a cladding defect. Periodic reactivity worth tests such as those described in NUREG-0308 are acceptable.

C. Postirradiation Surveillance

A postirradiation fuel surveillance program should be described for each plant to detect anomalies or confirm expected fuel performance. The extent of an acceptable program will depend on the history of the fuel design being considered (i.e., whether the proposed fuel design is the same as current operating fuel or incorporates new design features).

For a fuel design similar to that in other operating plants, a minimum acceptable program should include a qualitative visual examination of some discharged fuel assemblies from each refueling. Such a program should be sufficient to identify gross problems of structural integrity, fuel rod failure, rod bowing, dimension changes, or crud deposition. The program should also commit to perform additional surveillance if unusual behavior is noticed in the visual examination or if plant instrumentation indicates gross fuel failures. The surveillance program should address the disposition of failed fuel.

In addition to the plant-specific surveillance program, a continuing fuel surveillance effort should exist for a given type, make, or class of fuel that can be suitably referenced by all plants using similar fuel. In the absence of such a generic program, the reviewer should expect more detail in the plant-specific program.

For a fuel design that introduces new features, a more detailed surveillance program commensurate with the nature of the changes should be described. This program should include appropriate qualitative and quantitative inspections to be carried out at interim and end-of-life refueling outages. This surveillance program should be coordinated with the prototype testing discussed in Subsection II.3.B. When prototype testing cannot be performed, a special detailed surveillance program should be planned for the first irradiation of a new design.

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this SRP section is discussed in the following paragraphs:

1. 10 CFR 50.46 requires each PWR and BWR to be provided with an ECCS that must be designed so that its calculated cooling performance following a postulated LOCA conforms to acceptance criteria set forth in the regulation. Acceptance criteria in 10 CFR 50.46 establish both fuel system design limits and core cooling requirements. SRP Section 4.2 reviews the performance of the fuel system during postulated LOCAs related to flow blockage and the methods used to establish the initial fuel conditions before the LOCA. RG 1.157 or Appendix K to 10 CFR Part 50 presents acceptable methods to evaluate the performance of the ECCS. RG 1.126 provides an acceptable model for predicting the effects of fuel densification in commercial LWRs. Application of acceptance criteria established in 10 CFR 50.46 significantly reduces the possibility of a violent chemical reaction between the Zircaloy cladding and the coolant, which would result, if it were to occur, in the production of explosive hydrogen gas following an accident. It also ensures that damage to the fuel system in the event of an accident is never so severe as to prevent cooling of the core.

2. 10 CFR Part 100 requires the calculation of the exposure to an individual caused by the release of fission products to the environment during a postulated reactor accident and consideration of the result when determining the acceptability of a reactor site. 10 CFR Part 100 and RG 1.195 and RG 1.196 apply to reactors with DC applications before January 10, 1997, unless the reactor has adopted the AST, as defined in 10 CFR 50.67 and RG 1.183. RG 1.195 and RG 1.196 can be used in place of RG 1.3, RG 1.4, RG 1.5, RG 1.25, and RG 1.77. RG 1.183 and the requirements of 10 CFR 50.34 apply to new reactors with DC applications after January 10, 1997; the source terms for both new reactors and the AST are based on total effective dose equivalent rather than whole body dose as used in 10 CFR Part 100 and RG 1.195 and RG 1.196. This section discusses acceptable fission gas release models to perform radiological dose calculations; these models ensure that doses are not underestimated. RG 1.3, RG 1.4, RG 1.183, and RG 1.195 provide acceptable assumptions that may be used to evaluate the radiological consequences associated with a LOCA for BWRs and PWRs. RG 1.25, RG 1.183, and RG 1.196 provide acceptable assumptions that may be used to evaluate the radiological consequences associated with a fuel-handling accident at a fuel handling and storage facility at reactor sites. RG 1.77, RG 1.183, and RG 1.195 identify acceptable analytical methods and assumptions that may be used to evaluate the consequences of a rod ejection accident in PWRs. Evaluation of the radiological dose consequences associated with a postulated reactor accident, as prescribed in 10 CFR Part 100, provides assurance that nuclear reactors can be operated safely under worst-case conditions.

3. GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. One objective of the fuel system safety review cited in this section is to ensure that the fuel system is not damaged during normal operations or AOOs. SRP Section 4.2 specifies design limits to accomplish this objective, while this section reviews alternative design limits proposed by vendors. Compliance with GDC 10 significantly reduces the likelihood of fuel failures during normal operations or AOOs, thereby minimizing the possible release of fission products. In addition, preventing fuel damage during normal operation and AOOs may also reduce the severity of fuel damage during an accident. For example, an increase in the severity of fuel damage for normal operation may result in an increase in source term consequences, along with a decrease in core coolability and/or control rod insertability for postulated accidents.

4. GDC 27 requires that the reactivity control system be designed with margin to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes. The review of Section 4.2 ensures that fuel system damage is never so severe as to prevent control rod insertion when it is required. Maintaining the ability to insert control rods during postulated accidents minimizes the extent of fuel damage, thus reducing the amount of fission products released to the primary coolant system in the event an accident occurs.

5. GDC 35 requires that a system be provided to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) cladding metal-water reaction is limited to negligible amounts. This section reviews fuel system performance analysis methods under postulated accident conditions to ensure compliance with GDC 35. Application of GDC 35 to the design of the fuel system ensures that fuel rod damage will not interfere with effective emergency core cooling and that cladding temperatures will not reach a temperature high enough to allow a significant metal-water reaction to occur, thereby minimizing the potential for offsite release.

III. REVIEW PROCEDURES

The reviewer will select material from the procedures described below, as may be appropriate for a particular case.

These review procedures are based on the identified SRP acceptance criteria. For deviations from these specific acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

1. For review of a DC application, the reviewer should follow the procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

2. For review of both DC and COL applications, SRP Section 14.3 should be followed for the review of ITAAC. The review of ITAAC cannot be completed until after the completion of this section.

3. For construction permit (CP) applications, the review should ensure that the design bases set forth in the preliminary safety analysis report (PSAR) meet the acceptance criteria given in Subsection II.A. In addition, the CP review should determine, from a study of the preliminary fuel system design, that there is reasonable assurance that the final fuel system design will meet the design bases. This judgment may be based on experience with similar designs.

4. For operating license (OL) applications, the review should confirm that the design bases set forth in the final safety analysis report (FSAR) meet the acceptance criteria given in Subsection II.A and that the final fuel system design meets the design bases.

Much of the fuel system review is generic and is not repeated for each similar plant. That is, the reviewer will have evaluated the fuel design or certain aspects of the fuel design in previous PSARs, FSARs, and licensing topical reports. All previous reviews on which the current review depends should be referenced so that the plant safety evaluation report comprises a completely documented safety evaluation. In particular, the NRC safety evaluation reports for all relevant licensing topical reports should be cited. Staff in the organization responsible for reactor systems has also performed certain generic reviews, the findings of which have been issued as NUREG or WASH series reports. At the present time, these reports include WASH-1236, NUREG-75/077, NUREG-0085, NUREG-0303, NUREG-0401, and NUREG-0418, and they should all be appropriately cited in the plant safety evaluation reports. These reports should also cite the applicable RGs (RG 1.3, RG 1.4, RG 1.25, RG 1.60, RG 1.77, RG 1.126, RG 1.157, and RG 1.183). Deviation from these guides or positions should be explained. After briefly discussing related reviews, the plant safety evaluation should concentrate on those areas in which the application is not identical to one previously reviewed and approved and on areas related to newly discovered problems.

Analytical predictions discussed in Subsection II.3.C will be reviewed in PSARs, FSARs, or licensing topical reports. The validity of analytical models used to predict the performance of the fuel system design, and their applicability up to the design's specified burnup and power limit, should be reviewed. Fuel burnup and power limits should be specified for each fuel type used in the reactor and justified based on irradiated material properties data and prototypic test results. An exception may be made for prototype test assemblies, in which case only an estimate of the maximum burnup and power needs to be provided. When the methods are being reviewed, the staff may perform calculations to verify the adequacy of the analytical methods. Thereafter, audit calculations will not typically be performed to verify the results of an approved method that has been submitted in a safety analysis report. Calculations, benchmarking exercises, and additional reviews of generic methods may be undertaken, however, at any time a clear need arises to reconfirm the adequacy of the method.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

> The staff concludes that the fuel system of the ________ plant has been designed so that (1) the fuel system will not be damaged as a result of normal operation and anticipated operational occurrences, (2) fuel damage during postulated accidents will not be severe enough to prevent control rod insertion when it is required, and (3) core coolability will always be maintained, even after severe postulated accidents, thereby meeting the related requirements of 10 CFR 50.46; GDC 10, 27, and 35 in Appendix A to 10 CFR Part 50; and 10 CFR Part 100 (for existing reactors) or 10 CFR 50.34 (for new reactors) or 10 CFR 50.67 (as an alternative to 10 CFR Part 100 for existing reactors). This conclusion is based on the following:
>
> 1. The applicant has provided sufficient evidence that these design objectives will be met based on operating experience, prototype testing, and analytical predictions. Those analytical predictions dealing with structural response and fuel densification have been performed in accordance with (1) the guidelines of RG 1.60, RG 1.77, and RG 1.126, or methods that the staff has reviewed and found to be acceptable alternatives to those RGs and (2) the guidelines in Appendix A to SRP Section 4.2. Those analytical predictions dealing with control rod ejection (PWR) or drop (BWR) have been performed in accordance with the interim criteria for RIAs in Appendix B to SRP Section 4.2.
>
> 2. The applicant has provided for testing and inspection of new fuel to ensure that it is within design tolerances at the time of core loading. The applicant has made a commitment to perform online fuel failure monitoring and postirradiation surveillance to detect anomalies or confirm that the fuel has performed as expected.
>
> The staff concludes that the applicant has described methods of adequately predicting fuel rod failures during postulated accidents so that radioactivity releases are not underestimated and thereby meets the related requirements of 10 CFR Part 100 or 10 CFR 50.67 or 10 CFR 50.34 (for new reactors). In meeting these requirements, the applicant has (1) used the fission-product release assumptions of RG 1.3 (or RG 1.4), RG 1.25, RG 1.77, and RG 1.183 and (2) performed the analysis for fuel rod failures for the rod ejection accident in accordance with the guidelines of Appendix B to Section 4.2 or with methods that the staff has reviewed and found to be an acceptable alternative to Appendix B.

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this SRP section.

In addition, to the extent that the review is not discussed in other SER sections, the findings will summarize the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable.

V. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications submitted six months or more after the date of issuance of this SRP section, unless superseded by a later revision.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced RGs and NUREGs.

VI. REFERENCES

1. 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities.”

2. 10 CFR 50.34, “Contents of Applications; Technical Information.”

3. 10 CFR 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors.”

4. 10 CFR 50.67, “Accident Source Term.”

5. 10 CFR Part 52, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants.”

6. 10 CFR 52.47, “Contents of Applications.”

7. 10 CFR 52.97, “Issuance of Combined Licenses.”

8. 10 CFR Part 100, “Reactor Site Criteria.”

9. 10 CFR Part 50, Appendix A, General Design Criterion 2, “Design Basis for Protection Against Natural Phenomena.”

10. 10 CFR Part 50, Appendix A, General Design Criterion 10, “Reactor Design.”

11. 10 CFR Part 50, Appendix A, General Design Criterion 27, “Combined Reactivity Control Systems Capability.”

12. 10 CFR Part 50, Appendix A, General Design Criterion 35, “Emergency Core Cooling.”

13. 10 CFR Part 50, Appendix K, “ECCS Evaluation Models.”

14. Regulatory Guide 1.3, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors.”

15. Regulatory Guide 1.4, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors.”

16. Regulatory Guide 1.5, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors.”

17. Regulatory Guide 1.25, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors.”

18. Regulatory Guide 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants.”

19. Regulatory Guide 1.77, “Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors.”

20. Regulatory Guide 1.126, “An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification.”

21. Regulatory Guide 1.157, “Best-Estimate Calculations of Emergency Core Cooling System Performance.”

22. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

23. Regulatory Guide 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors.”

24. Regulatory Guide 1.196, “Control Room Habitability at Light-Water Nuclear Power Reactors.”

25. WASH-1236, “Technical Report on Densification of Light Water Reactor Fuels,” Atomic Energy Commission Regulatory Staff Report, November 14, 1972.

26. NUREG-75/077, “The Role of Fission Gas Release in Reactor Licensing,” November 1975.

27. NUREG-0085, “The Analysis of Fuel Densification,” July 1976.

28. NUREG-0303, “Evaluation of the Behavior of Waterlogged Fuel Rod Failures in LWRs,” March 1978.

29. NUREG-0308, Supp. 2, “Safety Evaluation Report Related to Operation of Arkansas Nuclear One, Unit 2,” September 1978.

30. NUREG-0401, “Fuel Failure Detection in Operating Reactors,” March 1978.

31. NUREG-0418, “Fission Gas Release from Fuel at High Burnup,” March 1978.

32. NUREG-0609, “Asymmetric Blowdown Loads on PWR Primary Systems,” January 1981.

33. NUREG-0630, “Cladding Swelling and Rupture Models for LOCA Analysis,” April 1980.

34. NUREG/CR-1018, “Review of LWR Fuel System Mechanical Response with Recommendations for Component Acceptance Criteria,” September 1979.

35. NUREG/CR-1019, “Pressurized Water Reactor Lateral Core-Response Routine, FAMREC (Fuel Assembly Mechanical Response Code),” September 1979.

36. NUREG/CR-1020, “Technical Evaluation of PWR Fuel Spacer Grid Response Load Sensitivity Studies,” September 1979.

37. NUREG/CR-1380, “Assessment of Current Onsite Inspection Techniques for LWR Fuel Systems,” Vol. 1, July 1980; Vol. 2, January 1981.

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40. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, “Rules for Construction of Nuclear Power Plant Components,” New York.

41. American Society for Testing and Materials, 1989 edition, Standard C776-89, Part 45, “Standard Specification for Sintered Uranium Dioxide Pellets,” Philadelphia, Pennsylvania.

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50. "Technical Report on Densification of Exxon Nuclear PWR Fuels," Atomic Energy Commission Regulatory Staff Report, February 27, 1975.

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52. Memorandum from V. Stello, NRC, to R.C. DeYoung, Subject: Evaluation of Westinghouse Report, WCAP-8377, Revised Clad Flattening Model, dated January 1975.

53. Memorandum from D.F. Ross, NRC, to R.C. DeYoung, Subject: CEPAN - Method of Analyzing Creep Collapse of Oval Cladding, dated February 5, 1976.

54. Memorandum from D.F. Ross, NRC, to D.B. Vassallo, Subject: Request for Revised Rod Bowing Topical Reports, dated May 30, 1978.

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PAPERWORK REDUCTION ACT STATEMENT

The information collections contained in the Standard Review Plan are covered by the requirements of 10 CFR Part 50 and 10 CFR Part 52, and were approved by the Office of Management and Budget, approval number 3150-0011 and 3150-0151.

PUBLIC PROTECTION NOTIFICATION

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

APPENDIX A

EVALUATION OF FUEL ASSEMBLY STRUCTURAL RESPONSE TO EXTERNALLY APPLIED FORCES

I. BACKGROUND

Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. SRP Section 4.2 states that fuel system coolability should be maintained and that damage should not be so severe as to prevent control rod insertion when required during these low probability accidents. This appendix describes the review that should be performed of the fuel assembly structural response to seismic and LOCA loads. NUREG-0609, NUREG/CR-1018, NUREG/CR-1019, and NUREG/CR-1020 provide background material for this appendix.

II. ANALYSIS OF LOADS

1. Input

   Input for the fuel assembly structural analysis comes from the results of the primary coolant system and reactor internals structural analysis, which is reviewed by the organization responsible for the review of mechanical engineering issues. Input for the fuel assembly response to a LOCA should include (1) motions of the core plate, core shroud, fuel alignment plate, or other relevant structures (these motions should correspond to the break that produced the peak fuel assembly loadings in the primary coolant system and reactor internals analysis) and (2) transient pressure differences that apply loads directly to the fuel assembly. If the earthquake loads are large enough to produce a nonlinear fuel assembly response, input for the seismic analysis should use structure motions corresponding to the reactor primary coolant system analysis for the safe-shutdown earthquake (SSE). If a linear response is produced, a spectral analysis may be used in accordance with the guidelines of RG 1.60.

2. Methods

   Analytical methods used in performing structural response analyses should be reviewed. The appropriateness of numerical solution techniques should be justified.

   Linear and nonlinear structural representations (i.e., the modeling) should also be reviewed. Experimental verification of the analytical representation of the fuel assembly components should be provided when practical.

   The applicant should work a sample problem of a simplified nature, which the reviewer will compare with either hand calculations or results generated with an independent code (NUREG/CR-1019). Although the sample problem should use a structural representation that is as close as possible to the design in question (and, therefore, would vary from one vendor to another), the applicant may make simplifying assumptions (e.g., one might use a three-assembly core region with continuous sinusoidal input).

The sample problem should be designed to exercise various features of the code and reveal their behavior. The sample problem comparison is not, however, designed to show that one code is more conservative than another, but rather to alert the reviewer to major discrepancies so that an explanation can be sought.

3. Uncertainty Allowances

The fuel assembly structural models and analytical methods are likely to be conservative; input parameters are also conservative. However, to ensure that the fuel assembly analysis does not introduce any nonconservatisms, two precautions should be taken—(1) if it is not explicitly evaluated, impact loads from the PWR LOCA analysis should be increased by about 30 percent to account for a pressure pulse, which is associated with steam flashing that affects only the PWR fuel assembly analysis and (2) conservative margin should be added if any part of the analysis (PWR or BWR) exhibits pronounced sensitivity to input variations.

Variations in resultant loads should be determined for positive variations in input amplitude and frequency of 10 percent; variations in amplitude and frequency should be made separately, not simultaneously. A factor should be developed for resultant load magnitude variations of more than 15 percent. For example, if +10-percent variations in input magnitude or frequency produce a maximum resultant increase of 35 percent, the sensitivity factor would be 1.2. Since resonances and pronounced sensitivities may be plant dependent, the sensitivity analysis should be performed on a plant-by-plant basis until the reviewer is confident that further sensitivity analyses are unnecessary or it is otherwise demonstrated that the analyses performed are bounding.

4. Audit

The reviewer should perform independent audit calculations for a typical full-sized core to verify that the overall structural representation is adequate. An independent audit code (NUREG/CR-1019) should be used for this audit during the generic review of the analytical methods.

5. Combination of Loads

To meet the requirements of GDC 2, as it relates to combining loads, an appropriate combination of loads from natural phenomena and accident conditions must be made. Loads on fuel assembly components should be calculated for each input (i.e., seismic and LOCA) as described in Subsection II.1 of this appendix, and the resulting loads should be added by the square-root-of-sum-of-squares method. These combined loads should be compared with the component strengths described in Subsection III according to the acceptance criteria in Subsection IV.

III. DETERMINATION OF STRENGTH

1. Grids

All modes of loading (e.g., in-grid and through-grid loadings) should be considered, and the vendor's laboratory grid strength tests should represent the most damaging mode. Test procedures and results should be reviewed to assure that the appropriate failure mode is being predicted. The review should also confirm that (1) the testing impact velocities correspond to expected fuel assembly velocities and (2) the crushing load P(crit) has been suitably selected from the load-versus-deflection curves. Because of the potential for different test rigs to introduce measurement variations, the review of the test procedure will evaluate the grid strength test equipment.

The consequences of grid deformation are small. Gross deformation of grids in many PWR assemblies would be needed to interfere with control rod insertion during an SSE (i.e., buckling of a few isolated grids could not displace guide tubes significantly from their proper location), and grid deformation (without channel deflection) would not affect control blade insertion in a BWR. In a LOCA, gross deformation of the hot channel in either a PWR or a BWR would result in only small increases in peak cladding temperature. Therefore, average values are appropriate, and the allowable crushing load P(crit) should be the 95-percent confidence level on the true mean as taken from the distribution of measurements on unirradiated production grids at (or corrected to) operating temperature. While P(crit) will increase with irradiation, ductility will be reduced. The extra margin in P(crit) for irradiated grids is thus assumed to offset the unknown deformation behavior of irradiated grids beyond P(crit).

2. Components Other Than Grids

Strengths of fuel assembly components other than spacer grids may be deduced from fundamental material properties or experimentation. Supporting evidence for strength values should be supplied. Since structural failure of these components (e.g., fracturing of guide tubes or fragmentation of fuel rods) could be more serious than grid deformation, allowable values should bound a large percentage (about 95 percent) of the distribution of component strengths. Therefore, ASME Code values and procedures may be used when appropriate for determining yield and ultimate strengths. Specification of allowable values may follow the ASME Code requirements and should consider buckling and fatigue effects.

IV. ACCEPTANCE CRITERIA

1. Loss-of-Coolant Accident

Two principal criteria apply for the LOCA—(1) fuel rod fragmentation must not occur as a direct result of the blowdown loads and (2) the 10 CFR 50.46 temperature and oxidation limits must not be exceeded. The first criterion is satisfied if the combined loads on the fuel rods and components other than grids remain below the allowable values defined above. The second criterion is

satisfied by an ECCS analysis. If combined loads on the grids remain below P(crit), as defined above, then no significant distortion of the fuel assembly would occur and the usual ECCS analysis is sufficient. If combined grid loads exceed P(crit), then grid deformation must be assumed and the ECCS analysis must include the effects of distorted fuel assemblies. An assumption of maximum credible deformation (i.e., fully collapsed grids) may be made unless other assumptions are justified.

Control rod insertability is a third criterion that must be satisfied. Loads from the worst-case LOCA that requires control rod insertion must be combined with the SSE loads, and control rod insertability must be demonstrated for that combined load. For a PWR, if combined loads on the grids remain below P(crit), as defined above, then significant deformation of the fuel assembly would not occur and lateral displacement of the guide tubes would not interfere with control rod insertion. If combined loads on the grids exceed P(crit), then additional analysis is needed to show that the deformation is not severe enough to prevent control rod insertion.

For a BWR, several conditions must be met to demonstrate control blade insertability—(1) combined loads on the channel box must remain below the allowable value defined above for components other than grids (otherwise, additional analysis is needed to show that the deformation is not severe enough to prevent control blade insertion) and (2) vertical liftoff forces must not unseat the lower tieplate from the fuel support piece such that the resulting loss of lateral fuel bundle positioning could interfere with control blade insertion.

2. Safe-Shutdown Earthquake

Two criteria apply to the SSE—(1) fuel rod fragmentation must not occur as a result of the seismic loads and (2) control rod insertability must be assured. The first criterion is satisfied by the criteria in Subsection IV.1 of this appendix. The second criterion must be satisfied for SSE loads alone if Subsection IV.1 does not require an analysis for combined loads.

APPENDIX B

INTERIM ACCEPTANCE CRITERIA AND GUIDANCE FOR THE REACTIVITY INITIATED ACCIDENTS

A. BACKGROUND

This appendix provides the interim acceptance criteria and guidance for the reactivity-initiated accident (RIA). RIAs consist of postulated accidents which involve a sudden and rapid insertion of positive reactivity. These accident scenarios include a control rod ejection (CRE) for pressurized water reactors (PWRs) and a control rod drop accident (CRDA) for boiling water reactors (BWRs). The uncontrolled movement of a single control rod out of the core results in a positive reactivity insertion which promptly increases local core power. Fuel temperatures rapidly increase, prompting fuel pellet thermal expansion. The reactivity excursion is initially mitigated by Doppler feedback and delayed neutron effects followed by reactor trip. Standard Review Plan (SRP) Section 15.4.8 and 15.4.9 provide further detail on the CRE and CRDA respectively. The technical and regulatory basis of this interim criteria is documented in a memorandum dated January 19, 2007 (ADAMS ML070220400).

B. FUEL CLADDING FAILURE CRITERIA

The total number of fuel rods that must be considered in the radiological assessment is equal to the sum of all of the fuel rods failing each of the criteria below. Applicants do not need to double count fuel rods that are predicted to fail more than one of the criteria.

1. The high cladding temperature failure criteria for zero power conditions is a peak radial average fuel enthalpy greater than 170 cal/g for fuel rods with an internal rod pressure at or below system pressure and 150 cal/g for fuel rods with an internal rod pressure exceeding system pressure. For intermediate (greater than 5% rated thermal power) and full power conditions, fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g. DNBR and CPR).

2. The PCMI failure criteria is a change in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in Figure B-1 (PWR) and Figure B-2 (BWR).

Fuel cladding failure may occur almost instantaneously during the prompt fuel enthalpy rise (due to PCMI) or may occur as total fuel enthalpy (prompt + delayed), heat flux, and cladding temperature increase. For the purpose of calculating fuel enthalpy for assessing PCMI failures, the prompt fuel enthalpy rise is defined as the radial average fuel enthalpy rise at the time corresponding to one pulse width after the peak of the prompt pulse. For assessing high cladding temperature failures, the total radial average fuel enthalpy (prompt + delayed) should be used.

C. CORE COOLABILITY CRITERIA

Fuel rod thermal-mechanical calculations, employed to demonstrate compliance with criteria #1 and #2 below, must be based upon design-specific information accounting for manufacturing tolerances and modeling uncertainties using NRC approved methods including burnup-enhanced effects on pellet power distribution, fuel thermal conductivity, and fuel melting temperature.

1. Peak radial average fuel enthalpy must remain below 230 cal/g.

2. Peak fuel temperature must remain below incipient fuel melting conditions.

3. Mechanical energy generated as a result of (1) non-molten fuel-to-coolant interaction and (2) fuel rod burst must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity.

4. No loss of coolable geometry due to (1) fuel pellet and cladding fragmentation and dispersal and (2) fuel rod ballooning.

D. FISSION PRODUCT INVENTORY

The total fission-product gap fraction available for release following any RIA would include the steady-state gap inventory (present prior to the event) plus any fission gas released during the event. The steady-state gap inventory would be consistent with the Non-LOCA gap fractions cited in RG 1.183 (Table 3) and RG 1.195 (Table 2) and would be dependent on operating power history. Whereas fission gas release (into the rod plenum) during normal operation is governed by diffusion, pellet fracturing and grain boundary separation are the primary mechanisms for fission gas release during the transient.

Based upon measured fission gas release from several RIA test programs, the staff developed the following correlation between gas release and maximum fuel enthalpy increase:

$$\text{Transient FGR} = [(0.2286 * \Delta H) - 7.1419]$$

Where:

FGR = Fission gas release, % (must be ≥ 0)

ΔH = Increase in fuel enthalpy, Δcal/g

The transient release from each axial node which experiences the power pulse may be calculated separately and combined to yield the total transient FGR for a particular fuel rod. The combined steady-state gap inventory and transient FGR from every fuel rod predicted to experience cladding failure (all failure mechanisms) should be used in the dose assessment. Additional guidance is available within RG 1.183 and 1.195.

FIGURE B-1: PWR PCMI Fuel Cladding Failure Criteria

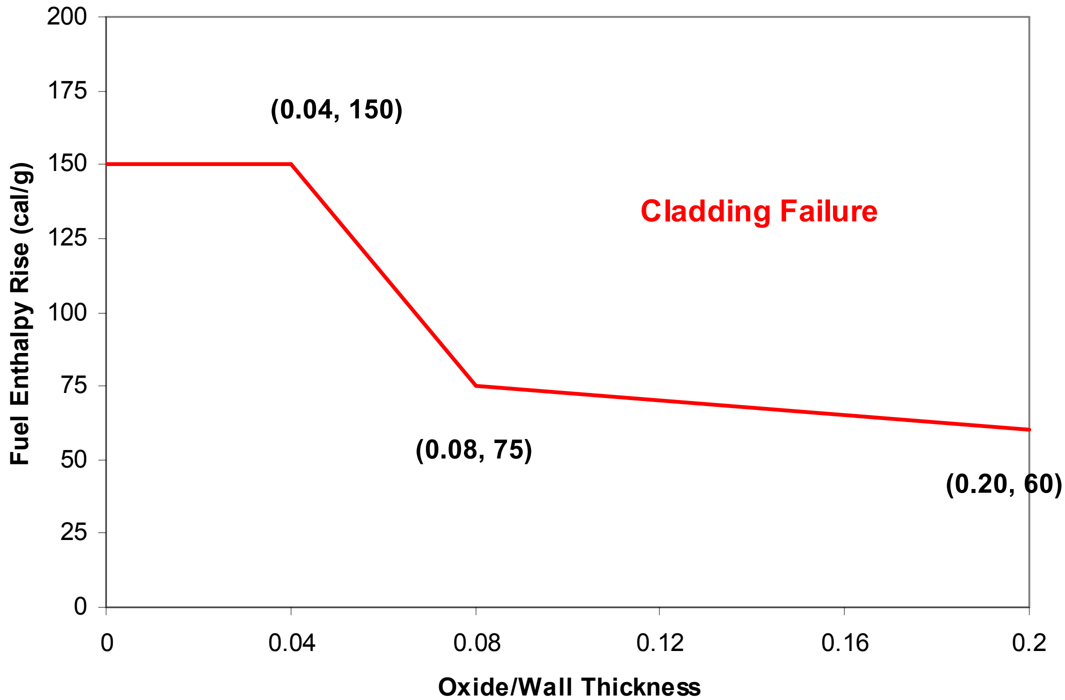

FIGURE B-2: BWR PCMI Fuel Cladding Failure Criteria

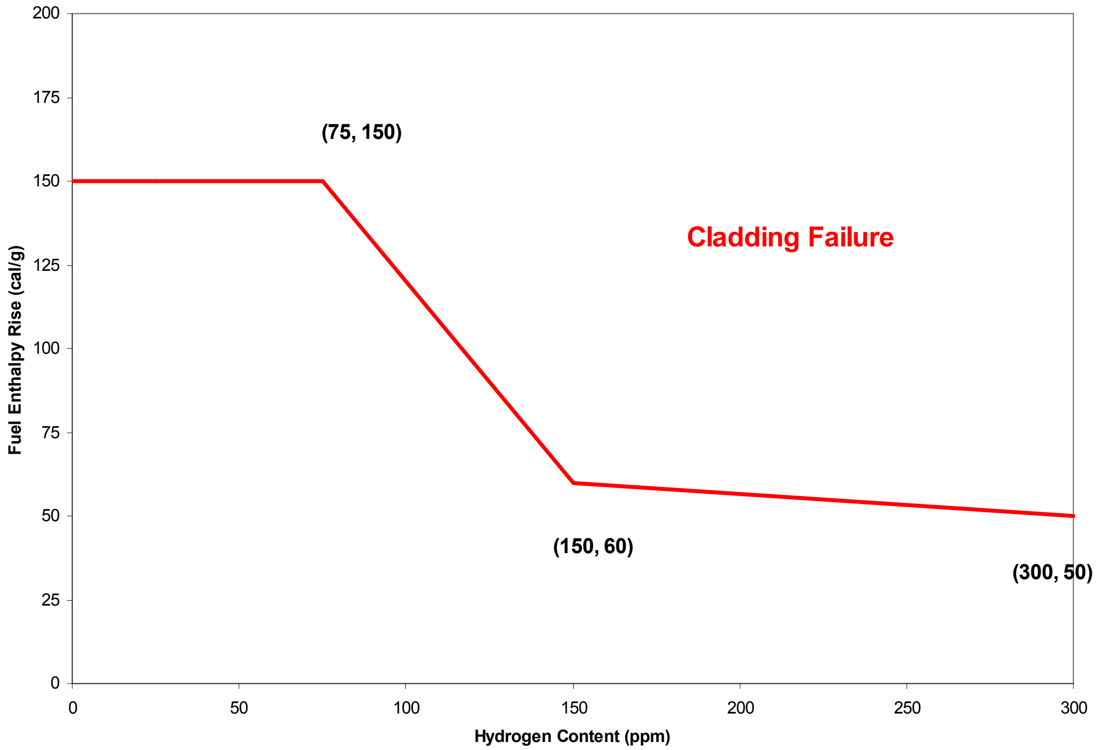

Figure B-2 Note:

The empirical database supporting the BWR fuel cladding failure criteria consists of NSRR tests conducted from initial test temperatures ranging from 20 to 85 °C. Due to temperature and hydrogen solubility effects, application of the BWR cladding failure criteria to higher operating temperatures is conservative. If properly justified, an applicant may adjust the failure criteria to account for hydrogen solubility at initial temperatures above 85 °C.