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COMMITTEE ON NUCLEAR REGULATORY ACTIVITIES

Regulatory Perspectives on Nuclear Fuel Qualification for Advanced Reactors (DRAFT)

Technical Report of the CNRA Working Group on the Safety of Advanced Reactors (Approved December 2020)

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List of abbreviations and acronyms

AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
ASN	Autorité de sûreté nucléaire
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CNSC	Canadian Nuclear Safety Commission
CNRA	Committee on Nuclear Regulatory Activities
DNR	Dounreay Fast Reactor
EBR	Experimental Breeder Reactor
EFA	Experimental Feeder Assembly
ENEA	L’energia e lo sviluppo economic sostenibile
EPRI	Electric Power Research Institute
EU	European Union

FCCI	Fuel/Clad Chemical Interaction
FCMI	Fuel/Clad Mechanical Interaction
FFTF	Fast Flux Test Facility
GFR	Gas-Cooled Fast Reactor
GIF	Generation IV International Forum
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit
HWR	Heavy Water Reactor
HTGR	High Temperature Gas Reactor
IA	Irradiation Assembly
IAEA	International Atomic Energy Agency
IPyC	Inner Pyrolytic Carbon
IRSN	Institut de radioprotection et de sûreté nucléaire
LFR	Lead-Cooled Fast Reactor
LWR	Light Water Reactor

MOX	Mixed Oxide
MSR	Molten Salt Reactor
MSRE	Molten Salt Reactor Experiment
NEA	Nuclear Energy Agency
NRC	Nuclear Regulatory Commission
ODS	Oxygen Dispersion Strengthened
OECD	Organisation for Economic Co-operation and Development
ONR	Office of Nuclear Regulation
OPyC	Outer Pyrolytic Carbon
ORNL	Oak Ridge National Laboratory
PCMM	Predictive Capability Maturity Model
PFR	Prototype Fast Reactor
PGSFR	Prototype Generation IV Sodium Fast Reactor
PIE	Post Irradiation Examination

PT	Pressure Tube
PWR	Pressurized Water Reactor
RBCB	Run Beyond Cladding Breach
SCWR	Super-Critical Water-Cooled Reactor
SEC NRS	Scientific and Engineering Centre for Nuclear and Radiation Safety
SFR	Sodium Fast Reactor
SiC	Silicon Carbide
TRISO	Tristructural-Isotropic
UK	United Kingdom
USA	United States of America
VHTR	Very High Temperature Reactor
WGSAR	Working Group on the Safety of Advanced Reactors

Executive summary

This technical report describes the regulatory perspectives on nuclear fuel qualification for advanced reactors and identifies topics that should be investigated in the frame of advanced reactor fuel regulation, potentially involving additional research and development needs. The present report is based on the input provided by seven countries.

The topic of fuel qualification is recognized as a challenging topic from a regulatory point of view due to a lack of clarity regarding the definition and scope of fuel qualification, and uncertainty associated with the regulatory basis for fuel qualification. This report attempts to address the topic of fuel qualification holistically by providing a definition of fuel qualification, clarifying the scope of fuel qualification, identifying common positions associated with current approaches to fuel qualification, and developing an assessment framework based on regulatory requirements. The development of the assessment framework makes up the bulk of this report and identifies specific criteria that should be evaluated in order to qualify a fuel for use in a nuclear reactor.

Common positions were identified based on a comparison of the information provided by the member countries that contributed to the report. These common positions are presented in Section 2.5, "Common positions", of this report and represent a general consensus among the member countries regarding high-level safety goals and objectives in the area of fuel qualification, as well as safety functions to be applied for Generation IV reactors. The essence of these common positions is mainly the following:

1. Fuel qualification requirements are derived from higher level nuclear power plant requirements (e.g., protection of "first barrier", accident source term, assurance of coolable geometry). However, guidance specific to the fuel qualification process is not generally available.
2. Process for qualifying fuel is generally implicit in the overall licensing of the nuclear facility.
3. An essential part of fuel qualification is to define a test envelope to cover expected operating, transient, and accident conditions to assess fuel performance and validate fuel performance codes.
4. An irradiation testing program, which includes testing of the integral fuel design, is necessary to identify fuel failure and degradation mechanisms. This testing should provide irradiation covering the exposure or burnup limits of the fuel.
5. Fuel qualification requires transient testing to assess fuel performance under transient and accident conditions.

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1. Background

1.1 Purpose

The objective of nuclear fuel qualification is the demonstration that a fuel product fabricated in accordance with a specification behaves as assumed or described in the applicable fuel licensing safety case, and with the reliability necessary for economic operation of the reactor plant (D. Crawford et. al., 2007). Regulatory agencies, responsible for the protection of public health and safety, focus on ensuring that high confidence is established in the licensing safety case for nuclear fuel performance. Advanced reactor designs are being proposed that utilize fuel designs and operating environments (e.g., neutron energy spectra, fuel temperatures, neighboring materials) that are outside of the large experience base available for traditional light water reactor fuel. The purpose of this report is to identify those best practices used by regulatory agencies to provide assurance that nuclear fuel behaves as assumed in the safety case for the proposed nuclear facility.

This technical report has been developed by WGSAR. Nuclear fuel qualification was identified by WGSAR as a topic for further investigation.

1.2 Safety Case

The role of nuclear fuel in the safety case can vary significantly between different reactor designs. For example, in most countries facilities that utilize traditional oxide fuels with metal cladding have been designed with robust barriers (e.g., a containment building) to protect against the release of radioactive material under accident conditions whereas in some countries (e.g., the USA) a facility that utilizes tristructural-isotropic (TRISO) fuel may credit a series of barriers (including barriers within the fuel itself) to protect against the release of radioactive material (i.e., a functional containment). Specifying the fission product retention functions of the nuclear fuel is an essential step in nuclear fuel qualification. The types of fuels considered for Generation IV nuclear energy systems and the associated safety case are further discussed in Section 1.4 of this report.

1.3 Scope

Many aspects of nuclear safety are impacted by nuclear fuel including neutronic performance, thermal-fluid performance (e.g., margin to critical heat flux limits), fuel mechanical performance, corrosion, reactor core seismic behavior, fuel transportation, and storage. Several of these safety considerations are discussed in available review guidance for fuel that is similar to light water reactor fuel (NEA, 2012). The scope of this report focuses on the identification and understanding of fuel life limiting failure and degradation mechanisms that occur as a result of irradiation during reactor operation. Additionally, the assessment criteria developed in Section 3 of this report are informed by regulatory experience licensing solid fuel reactors designs (particularly light water reactor designs). An attempt has been made to develop generically applicable criteria. However, it is recognized that some criteria may not be applicable to liquid fuel forms (e.g., Molten Salt Reactors).

1.4 Fuel for Generation IV Nuclear Energy Systems

1.4.1 Sodium Cooled Fast Reactor (SFR)

Within GIF, recent evaluation of SFR advanced fuel options entailed a comparison of the oxide, metal, and nitride fuel types with respect to fuel fabrication processes and fuel performance during steady-state irradiation and off-normal transients with the purpose of identifying advanced fuel candidates for different applications and fuel cycle missions. Additional research and development efforts were also focused on the minor-actinide-bearing fuels and high burnup capability evaluation.

Historically, the metal fuel in an SFR was first used in the USA in Experimental Breeder Reactor (EBR)-I in 1951 and later on in the UK in Dounreay Fast Reactor (DFR) in 1957, while the oxide fuel was first selected in the BR-5 reactor in 1958 in Russia, and then in Rapsodie in 1967 in France. During the following five decades, these two fuel types have been studied extensively worldwide, with substantial accumulated irradiation and transient testing experience. Additional metal fuel experience comes from EBR-II, FERMI, and the fast flux test facility (FFTF) in the USA. Additional oxide fuel experience comes from BR-10, BOR-60, BN-350, BN-600 and BN-800 in Russia, the prototype fast reactor (PFR) in the UK, Phenix and Superphenix in France, KNK and SNR-300 in Germany, JOYO and Monju in Japan, FFTF in USA, and CEFR in China. Nitride fuel performance was studied in Russia at the pin and sub-assembly level in BR-10 (1983-2002) and in experimental fuel pins and sub-assemblies in BOR-60 and currently in BN-600. Additional nitride fuel experience on a limited number of pins was obtained in experimental fast reactors (RAPSODIE-France, DFR-UK, EBR-II – USA, PHENIX – France, HFR – Netherlands, JOYO – Japan).

The metal fuel is considered to have the following advantages:

- high density with favorable breeding gain, and harder neutron spectrum with improved neutron economy
- high thermal conductivity, allowing the fuel to operate at lower temperature as a favorable safety attribute
- chemical compatibility with sodium coolant
- simple manufacturability of the fuel
- demonstrated capability of its electrochemical reprocessing

The oxide fuel is considered to have the following advantages:

- high melting temperature
- excellent structural and chemical stability at high temperature and high burnup
- generally good retention of fission products, particularly in contact with water
- good Doppler feedback due to low thermal conductivity
- compared to metal fuel, lower fuel swelling under irradiation
- manufacturing and reprocessing processes similar to the Light Water Reactors (LWR) fuel industrial processes, taking advantage of LWR experience and existing facilities

The nitride fuel is considered to have the following advantages:

- high density compared to oxide
- higher thermal conductivity than oxide fuel

- high melting point

The selection of a fuel type for an SFR is strongly related to the core design requirements, past irradiation and testing experience, availability of fuel manufacturing facilities, and the back end of the fuel cycle considerations in each country. Among the current GIF members, China, France, Japan and the European Union (EU) have selected oxide fuel for their main SFR design tracks. The USA and Republic of Korea are pursuing SFR designs with metal fuel. Although current generation of SFRs in Russia are oxide fueled, the nitride fuel is considered for the next generation BN-1200 design. All GIF members recommend starting with ferritic/martensitic or austenitic steel clad and assembly duct, but aim to transition in the longer term to other advanced alloys, such as oxide dispersion strengthened steels, or ceramics.

1.4.1.1 Description

The typical SFR fuel element has the following common characteristics:

- A cylindrical steel clad sealed at both ends by two welds plugs. In addition to the fuel matrix that holds the solid fission products, the sealed cladding constitutes a barrier against the release of radioactivity
- A fissile column, typically consisting of a stack of oxide or nitride fuel pellets, or long metal alloy rods (slugs)
- Possible fertile blankets, reflectors, or shielding located on top and/or bottom of the fissile column
- The fission gas plenum at the top and/or bottom end of the fuel element
- A spacer-wire of the same steel grade as the clad, helically wound with a constant pitch along the full length of the cladding
- Fuels pins are typically arranged in triangular pitch inside a hexagonal fuel assembly duct that controls the flow into each assembly

<u>Metal Fuel</u>

The metal fuel is typically injection cast as binary (U-Zr) or ternary (U-Pu-Zr, U-TRU-Zr) alloys of long rods (slugs). The metallic alloy is stabilized typically using 5 to 30% addition of zirconium to increase the melting point, improve the structural strength, and minimize the potential for fuel/clad chemical interaction (FCCI). The fuel is thermally bonded to the cladding using sodium inside the cladding, providing a high thermal conductivity medium to facilitate almost unimpeded transfer of the heat generated in fuel to the cladding and reactor coolant. The relatively larger fission gas collection plenum (typically located above the fuel) is included to capture the gaseous fission products released from the fuel during irradiation. Most metallic fuel forms maintain a low fuel-smeared density at or below 75% to provide room for early swelling and development of interconnected porosity to allow the escape of fission gasses from the fuel matrix and to accommodate fuel swelling at higher burnup.

<u>Oxide Fuel</u>

The SFR oxide fuel is similar to LWR fuel in the form of uranium- or mixed-oxide sintered (ceramic) pellets stacked inside a cladding tube that is filled with helium to improve conduction of heat between the pellet and the cladding. Due to low thermal conductivity of the fuel and small gap conductance,

oxide fuel operates at relatively higher temperature in comparison to metal fuel. However, its high melting point provides a comparable margin to melting during postulated accidents. Good structural stability of the oxide fuel limits it swelling. Oxide fuel pins typically include a fission gas plenum that is, in some designs, located below the fueled section where the sodium coolant temperature, and therefore the plenum pressure, are lower. The oxide fuel element design can take advantage of LWR fuel fabrication processes and the absence of sodium bond inside the cladding can also simplify the back end fuel cycle processes.

Nitride Fuel

Nitride fuel for fast reactors – more precisely uranium and plutonium nitride (U,Pu)N - is used in the form of sintered pellets stacked in a cylindrical pin with a smeared density of around 80%. Density of pellets already manufactured spanned from 80% to 95% of theoretical density. Cladding is usually made of ferritic-martensitic steel like the T91 grades, but austenitic steels are also suitable. Sodium or even lead bonds are possible, the trend being to use helium which proved better results in terms of cladding internal corrosion.

(U,Pu)N combines three specific physical properties:

- a high melting point (3100 K) of the same order of magnitude as that of mixed oxide (MOX) fuel
- a thermal conductivity (around 18 $W.m^{-1}.K^{-1}$) comparable to that of the carbide or metal and very superior to that of the oxide
- a density of fissile nuclei comparable to that of metal and significantly higher than the oxide

Nevertheless, fuel dissociation into metallic phases and nitrogen occurs at temperatures much lower than the melting point (around 2000 K – depending on nitrogen partial pressure) (IAEA, 2017), (WHC, 1991). To avoid this phenomena, it is recommended to operate below 1650 K. The decomposition limit is thus an important criterion against which margins have to be estimated in case of accidents. However, considering its thermal conductivity and the targeted linear power, the fuel maximum centerline temperature would be lower than 1200 K.

Nitride fuel volume swelling rate is a function of burnup, centerline temperature and initial porosity. Nitride fuel volume swelling is a higher than that of the oxide fuel (for example UN swelling was around 12% for 8.2 at% at 1170 K in the BR-10 core) (Adamov and Orlov, 2001) and may be managed through the optimization of the initial porosity and smeared density. However, adjustments to porosity and smeared density in order to accommodate swelling may also reduce some of the advantages of nitride fuel (e.g., fissile nuclide density). Nitride fuel also retains the majority of fission gasses. It is to be noted that swelling and retention of fission gasses is sensitive to impurities like carbon and oxygen. The hardness and brittleness of nitrides leads to a hard contact of the swelling fuel with cladding through the fuel splinters in the gas gap already at an early stage of irradiation, almost immediately after the reactor power rises to the nominal level. Such contact leads to local deformation, ovalization and elongation of the cladding and would induce a risk of clad punching.

Finally, the use of nitride fuel based on natural nitrogen leads to the production of an abnormally large amount of C14 and tritium as well as to the carburization of the inner surface of the claddings and their embrittlement Accordingly, it is envisaged to enrich nitride fuel in N15 isotope to avoid the production of C14 and H by N14(n,p)C14 reaction, production of tritium by N14(n,T)C12 reaction, and a significant degradation of the neutron balance of the core (especially for high power reactors). Such enrichment may involve the reprocessing of the fuel for economic reasons.

1.4.1.2 Role in Safety Case

Metal Fuel

Metal fuel performance and failure modes depend on various irradiation effects such as fuel-alloy constituent redistribution, porosity formation, fission gas retention and release, irradiation-induced radial and axial swelling of fuel slugs, and formation of low melting temperature eutectic formation at the fuel-cladding interface resulting in a gradual thinning of the cladding. The primary failure mode for the conventional metal fuel forms is fission gas induced breach of the cladding that is weakened due to eutectic thinning. Although FCCI is expected to be a very slow process at the operating temperatures, it can lead to accelerated cladding failure at elevated temperatures (below the fuel and cladding melting points) during postulated accidents. Because eutectic formation is a highly temperature-dependent process, it limits the coolant outlet temperature for an SFR with metal fuel, but it does not impose a major burnup limit like fuel/clad mechanical interaction (FCMI) does.

High thermal conductivity of the fuel and high gap conductance play the most significant role in safety advantages of the metal fuel, resulting in a flatter radial temperature profile within the pin and low operational and transient temperatures. This also results in a small Doppler contribution to feedback during reactivity induced accidents. Accordingly, reactor designs that utilize metal fuel need to ensure that adequate mechanisms are in place to provide sufficient negative feedback during power excursion events and to ensure that the amount and rate of possible reactivity insertions is limited to the capacity of the feedback mechanisms.

Under accident conditions, the load on cladding is typically dominated by the fission gas pressure. Therefore, the larger fission gas plenum in metal fuel pins plays an important role in delaying the cladding failures. Chemical compatibility of the metal fuel with sodium coolant helps avoid the energetic fuel-coolant reactions and resulting potential for flow blockages.

Although the metallic fuel has a relatively low melting temperature, the power level of the reactor can be set to offer a comparable margin to melting during postulated accidents relative to oxide fuel due to proportionally low operating temperature (reactivity faults being the exception). Melting temperature of metal fuel is generally lower than that of the cladding material.

Oxide Fuel

As with all fuel, oxide fuel performance and failure modes depend on various irradiation effects such as fuel restructuring and grain growth, fission gas retention and release impacting the fuel swelling, as-fabricated porosity migration leading to formation of a central cavity, solid fission product induced fuel creep, and fuel-cladding gap condition impacting the fuel temperatures. The typical oxide fuel failure modes include the plastic straining of the cladding due to internal fission gas pressure and FCMI.

Low thermal conductivity and low gap conductance of oxide fuel can result in higher operational and transient temperatures in comparison to metal fuel. But its high melting point also provides a comparably large margin to fuel melting. High operating temperature and availability of oxygen atoms in the fuel (resulting in softer neutron spectrum) leads to a stronger Doppler feedback for oxide fuel relative to metal fuel, reducing the overpower the fuel will experience in rapid reactivity faults (compared to other fuel types). Robustness of oxide fuel against rapid overpower transients up to several times the nominal power has also been demonstrated in tests at the CABRI and TREAT test reactors.

The vast international experience with oxide fuel spanning over more than five decades of experiments in test and prototype reactors makes its assessment and qualification high enough to be used in commercial reactors, allowing its improvement for various missions such as reduced sodium void worthwhile addressing its main drawbacks. Some examples that can be mentioned: a good knowledge concerning the fuel evolution during accidents leading to a cladding failure, and new design improvements based on use an annular fuel pellets to increase the melting margin, raise the limit burnup rate, and radically reduce FCMI during both slow and fast power transients.

Off-normal behavior of MOX fuel is also fairly well understood with main remaining questions being the transient behavior of very high burnup fuel and Run Beyond Cladding Breach (RBCB) behavior of minor-actinide-bearing fuel. The transient behavior of high burnup fuel is not considered to be critical if the fuel-smeared density is appropriate and such designs have already been experimentally verified. Concerning the RBCB behavior, since minor actinides, when in limited quantity, have a somewhat higher oxygen potential than (U,Pu) in the fuel, and formation of Na-TRU-O behavior may not be different from previous test results for MOX fuel. The effects of minor-actinide-bearing will be evaluated experimentally and analytically.

Nitride Fuel

Nitride fuels have a high heavy metal density compared to oxide fuel, which allows a harder neutron spectrum and more attractive breeding ratios at the expense of higher rates of neutron damage to materials. In principle, this would allow better fuel utilization if materials can be found which have acceptable endurance. The breeding ratio would also allow cores to be designed with lower excess reactivity, which could balance to some degree the reduced levels of Doppler feedback. The fuel is chemically compatible with sodium, water, and air, although more work is required on fission product retention.

Based on the understanding of the off-normal behavior of carbide and oxide fuel systems, it is expected that the off-normal behavior of nitride fuel will principally depend on the following:

- Nitride decomposition at high temperature could lead to gaseous nitrogen release and thus a pressurization of the structural pressure boundaries but also the appearance of metallic phases enriched in Pu with a low melting point. The decomposition temperature of mononitrides of uranium and plutonium is above 2000 K. However, mixed nitride fuel may contain several percent U_2N_3 which decomposes below 1600 K and leads to the nitriding of the inner surface of the cladding with an associated decrease in cladding ductility.
- The high thermal conductivity and high melting point of nitride fuel could lead to vapor explosions when the molten fuel comes in contact with the colder coolant. The intense heat transfer between the fuel melt and the sodium coolant may result in rapid conversion of the thermal energy into mechanical work, resulting in shock waves and endangering the integrity of surrounding structures.
- The chemical and structural performance of the fuel including fuel fragmentation and fission product mobility.

Experimental data on these issues are sparse. So, the assessment of these issues requires further study.

1.4.1.3 Challenges to Fuel Qualification

Irradiation and transient testing exist for both standard oxide and metal fuel types for burnup levels higher than the LWR fuel. The GIF SFR Advanced Fuels Project has also been studying the technical basis behind the choice and direction of research, development, and qualification of various advanced fast reactor fuels. The conclusions on current status and future challenges for qualification of each fuel type and cladding are summarized below.

Metal Fuel

Metal fuel has substantial experience demonstrated at the assembly level from the historical USA program. This experience provides the basis for utilization of various metal fuel forms in fast reactors up to approximately 10 atom-% burnup level. Capability of metal fuel have also been demonstrated up to 20 atom-% burnup in ferritic/martensitic steel cladding. The main challenge for metallic fuel qualification is consideration of this legacy data from EBR-II and FFTF irradiation tests under accepted quality assurance standards.

The USA Department of Energy's Argonne National Laboratory prepared a quality assurance program plan for the legacy data from these past metal fuel irradiation tests and submitted it to USA Nuclear Regulatory Commission (NRC) for evaluation in terms of its applicability in conformance with the current quality assurance standards. The plan focuses on identification, description, and technical evaluation of quality assurance processes under which the historic irradiation tests were conducted to facilitate the timely validation of key attributes of historic analytical and measured metallic fuel data, to the extent that such data can be used to support future commercial licensing efforts.

Following NRC review and approval, implementation of this plan is currently underway to assess the historical data integrity, completeness and availability of programmatic support information. The American Society of Mechanical Engineers (ASME) NQA-1 Standards have been adopted as the means by which the NRC's quality assurance criteria can be met. Once completed, test data and related technical information qualified under this program is expected to be used by future license applicants for new advanced reactor designs within their individual programs addressing the NRC requirements.

Republic of Korea has also carried out metal fuel qualification program with U-Zr fuel and ferritic/martensitic steel cladding, which is to be loaded to the initial core of Korean Prototype Generation IV Sodium Fast Reactor (PGSFR) design. The HT9 and FC92 cladding materials have been irradiated up to 75 dpa and U-Zr fuel has been reached to 7 atom-% burnup.

Oxide Fuel

Oxide SFR fuel is also considered to be one of highest technical readiness level with demonstrated adequate high burnup performance as driver fuel at the assembly level and transmutation fuel at the fuel pin level. This experience provides the basis for utilization of various oxide fuel forms in fast reactors up to approximately 15 atom-% burnup level. Capability of oxide fuel has also been demonstrated up to 20 atom-% burnup in both austenitic and ferritic/martensitic cladding with using wire or grid spacers. Currently, most of the available information on thermal and mechanical properties is for MOX fuel. Data on oxide fuel with minor actinides of interest is not yet satisfactory or complete. The most important gap to be filled concerns certain thermal properties such as thermal conductivity, specific heat, and the thermo-mechanical properties that may impact thermal creep. There are also some uncertainties about irradiation effects. However, the SUPERFACT 1 irradiation test performed in the

1980s with a similar composition of the GACID fuel showed generally good behavior of the fuel at medium burnup and medium linear power.

Recent fuel property studies and irradiation tests of minor-actinide-bearing oxide fuel are providing solutions to the primary issues. In the current status, no critical issue has been found for high burnup minor-actinide-bearing oxide fuel. Concerning the fabrication processes for minor-actinide-bearing oxide fuel, There are challenges concerning radioactivity protection during the fabrication process (shielding, elimination of dust, and automation) of the fuel production line as well as challenges associated with heat generation by the minor actinides. Continuous developmental effort of high burnup minor-actinide-bearing oxide fuel will prepare sets of fuel technology for its application to the future SFR driver fuels.

Nitride Fuel

The nitride fuel is currently under development in Russia in the frame of the PRORYV project. More than 1000 fuel pins of different geometry with different cladding materials have been fabricated and successfully irradiated in BOR-60 and BN-600. With regard to high burnup and transmutation fuel performance, there are no critical issues identified to-date, but these technologies are in the early stages of assessment comparing to oxide or metal fuels. The main qualification challenge for the nitride fuel is that the experience basis for normal irradiation off-normal transient response is not yet well developed. Additionally, there is limited data to support reprocessing of nitride fuel and there are manufacturing difficulties due to its high sintering temperatures.

According to early experimental result, U0.8Pu0.2N begins to vaporize beyond ~1730°C under He atmosphere, with a partial pressure much larger for Pu than for U. The preferential evaporation of metallic Pu towards U and the N_2 gas release are key issues that have to be managed for irradiation and fabrication. The out-of-pile study of mixed nitride behavior under high temperatures are currently under way in Russia.

Irradiation test have been performed for nitride fuels. Demonstration of fabrication techniques for minor-actinide-bearing fuels remains to be completed. ^{15}N enrichment processes, essential for nitride fuel developments having suitable economics must also be developed. General lack of availability of fast-flux irradiation facilities in the world is also a challenge.

Cladding

There is experience with ferritic/martensitic or austenitic steel clad and the assembly duct for both metal and oxide fuels, but these materials suffer from significant irradiation damage before reaching the irradiation levels ultimately envisaged. Advanced cladding materials are also being developed based on the experiences accumulated over half a century of cladding and core structure materials research. The candidate cladding materials are stabilized austenitic steels, ferritic/martensitic steels such as oxide dispersion strengthened (ODS) type steels, and additional advanced alloys. Many of these materials have yet to progress to the stage of industrial manufacturing. To develop licensable cladding tubes, a large amount of irradiation tests will be necessary. Furthermore, the compatibility of this new cladding material with TRU loaded fuel at a high temperature and a high burnup will have to be verified by integral tests of pins in-reactor. In case of metal fuel, the development of barrier cladding and its verification is necessary because eutectic melting temperature is decreased between transuranic metal fuel and cladding. As for the ferritic/martensitic steel duct materials, development of advanced

ferritic/martensitic steels including ODS type aims to be able to meet the dose requirements, over 200 dpa, for the Gen-IV SFR systems. General lack of availability of fast-flux irradiation facilities also hinders the progress for qualification and eventual use of advanced cladding materials.

1.4.2 Very-High-Temperature Reactor (VHTR)

The VHTR utilizes tristructural-isotropic (TRISO) coated fuel particles embedded in a graphitic matrix. TRISO particle design has evolved over the last six decades from rudimentary particles with simple, single-carbon-layer coatings to the multilayer design. An overview of the historic perspectives and current progress on coated TRISO fuel particles for high-temperature reactors can be found in (Demkowicz et. al., 2019), (M.J. Kania et. al., 2015), and (IAEA, 2010).:

1.4.2.1 Description

The two common VHTR fuel forms with multilayer TRISO particles design are shown in Fig. 1-1. The particle consists of a spherical fissile kernel (diameter typically 350 – 600 µm), surrounded by a series of coating layers including low-density pyrocarbon buffer, dense inner pyrolytic carbon (IPyC), silicon carbide, and dense outer pyrolytic carbon (OPyC).

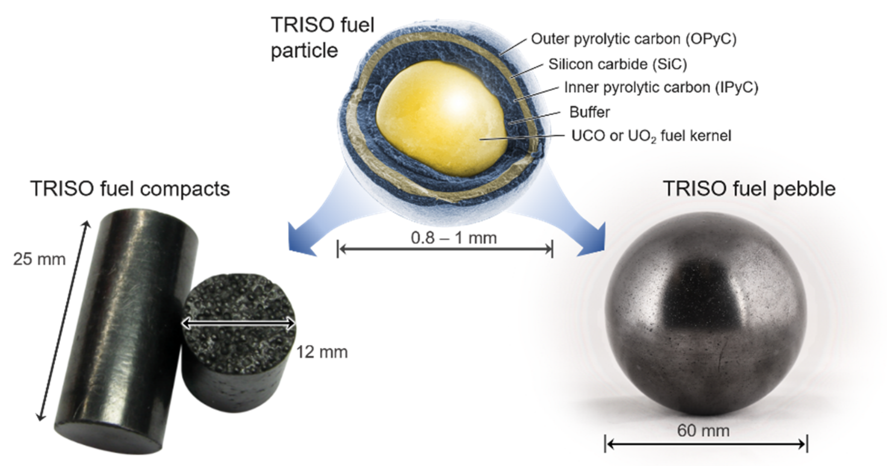

Figure 1-1. TRISO fuel particle, fuel compacts, and spherical fuel pebble

Although a wide variety of kernel compositions and geometries have been employed in the past, most modern TRISO fuel particles utilize a kernel of either uranium dioxide or UCO, a heterogeneous mixture of uranium oxide and uranium carbide. The kernel and coating layers together act to retain fission products within the particles., preventing significant release of most fission products to the gaseous coolant where they can migrate throughout the primary coolant circuit and can be released to the reactor building during various accidents. Certain fission products (e.g., strontium, cesium, silver,

and europium) have proved problematic and historically decontamination of primary circuit components has been a serious challenge.

The matrix material plays an important role in fuel performance, as it consolidates the particles into a solid fuel form for insertion into the reactor, promotes heat removal from the particles, and can provide significant retention of certain fission products once released from the particles. The matrix material consists of graphite flake mixed with phenolic resin, which is applied to the coated particles as an overcoat prior to pressing the particles into the final shape. Heat treatment of the resin eliminates volatiles, carbonizes the resin, and results in a robust final fuel form. The particles are generally formed into either cylindrical compacts (with or without a central annular hole) or into spherical "pebbles" as shown above. Cylindrical compacts are inserted into graphite blocks for use in a prismatic core reactor, and fuel pebbles are stacked together inside the core of a pebble bed reactor.

The kernels, coatings, and compacts/pebbles are subject to a wide range of property specifications to ensure acceptable irradiation performance. There are also critical specifications on the amount of loose uranium contamination in the fuel outside of the silicon carbide (SiC) layer and the fraction of "exposed kernels" (kernels surrounded by defective coatings such that they do not retain fission products). Both of these defects contribute to the release of fission gas during irradiation.

1.4.2.2 Role in Safety Case

High-temperature reactors rely on barriers within the fuel to limit radionuclide release . These barriers are responsible for the vast majority of radionuclide retention; a relatively small fraction of radionuclides is expected to escape from the fuel under any conditions, provided that a severe accident is avoided. The fuel is robust and designed to withstand relatively high temperatures, as high as ~1200 – 1300°C during normal reactor operation, with peak fuel temperatures as high as 1600°C during reactor accidents. The main sources of radionuclide release from the fuel are (1) uranium contamination outside of the SiC layer, (2) initially-defect kernels, and (3) particles in which the coating layers fail during operation. The first two of these sources are controlled by fuel quality specifications. Thus, a key metric of in-pile TRISO fuel performance is the fraction of particles experiencing coating failure.

A number of mechanisms for particle failure have been identified and experimentally observed throughout the decades of TRISO fuel development and irradiation testing. With proper particle design and operation of the fuel within an acceptable performance envelope, particle failure can be reduced to very low levels during normal operation, with fractions of around 10^{-4} or lower achievable during normal reactor operation.

In addition to release from failed particles, certain fission products can be released through intact TRISO coatings if fuel temperature is sufficiently high. The most notable example is silver, which is released from intact particles at elevated temperatures. Other fission products that can be released from intact particles at elevated temperatures include strontium and europium. Such fission product release is strongly influenced by fuel kernel temperature and is a phenomenon that needs to be considered during reactor design and regulation.

1.4.2.3 Challenges to Fuel Qualification

TRISO fuel performance is demonstrated through irradiation testing and post-irradiation high-temperature safety tests. To support the fuel qualification, in-pile tests should subject the fuel to the range of conditions expected during operation, including temperature, burnup, fast neutron fluence, and particle power. High-temperature safety tests should also be performed at temperatures representative of expected peak fuel temperatures during reactor accidents (1600°C has historically been used as a limiting value for intact circuit faults, dependent on specific design), but are commonly performed for long durations (hundreds of hours) in order to observe the effects of extended time at elevated temperatures. Higher temperature testing (up to 1800°C or more) is also needed to demonstrate performance margin, to accelerate temperature-driven fuel degradation and improve the understanding of fuel performance. The atmosphere for post-irradiation safety tests is also an important consideration. A significant amount of testing has been performed in various countries in pure helium, as this condition emulates a depressurized conduction cooldown accident in a modular high temperature gas reactor (HTGR). However, the impacts of oxidizing gasses entering the core (i.e., air or moisture) should also be included in fuel qualification efforts, as the existing international database is much more incomplete.

Fuel qualification data should include particle failure fractions (typically reported as the upper bound at 95% confidence) under the conditions described above. This requires that an adequate population of particles be tested during irradiation and during safety tests to verify the necessary failure fractions at 95% confidence. In addition to quantifying particle failure fractions under normal and accident conditions, fuel qualification should include an assessment of fission product release from particles and from the compacts or pebbles in order to understand the rate of radionuclide release to the coolant. Analysis of short-lived fission product release (e.g., I131 with a half-life of 8 days) is a particular challenge because PIE typically occurs many months following irradiation. Thus, there is a need to perform reirradiation of the fuel specimens immediately prior to safety tests to generate short-lived fission products for subsequent measurement.

TRISO fuel is currently seeing a great deal of interest for applications beyond the conventional high-temperature and very-high-temperature gas-cooled reactor designs of past decades. In particular, the fuel is targeted for use in micro-reactor designs and reactors with molten salt coolant. In addition, “advanced” TRISO fuel designs have been proposed dating to the 1970s—including replacement of the SiC layer with other refractory ceramics (e.g., ZrC) and modifications to the kernel or coating layer designs—to increase the fuel temperature or burnup limits in HTGRs. New fuel qualification data of the type considered in this section will need to be obtained for advanced coated particle designs or for conventional TRISO particles where operating conditions would lie significantly outside of the fuel performance envelope demonstrated in previous and ongoing fuel qualification efforts.

1.4.3 Gas-Cooled Fast Reactor (GFR)

The operation at a high temperature and in fast neutron flux creates great challenges for the development of GFR fuel. A GFR has never been built, but GFR projects were launched in different countries since the 1960s. Those GFR designs included probably the most diverse fuel types considered for nuclear reactors.

1.4.3.1 Description

On overview of the fuel types considered in numerous GFR designs are documented in (Stainsby et. al., 2009).

- The first GFR designs were based on the liquid-metal cooled fast reactor experience and included conventional pin-type, stainless steel cladded fuel assemblies with oxide pellets and with roughened external cladding surface to enhance heat transfer to the gas coolant.
- Coated particles were considered in some European designs with different geometrical arrangements (e.g. cylinders with perforated annuli or “stack of saucers” geometry). The proposed materials for structural elements were SiC and stainless steel.
- In the Soviet Union, chromium dispersion fuel pins were proposed with small inclusions of U metal or UO2 in a matrix of chromium for the GFRs with corrosive, dissociating N2O4 coolant.
- In Japan, coated particle fuel with nitride fuel kernels in TiN sealing layers was considered. In one assembly type, the coated particles were arranged in an annular bed. The other design featured large prismatic blocks filled with a mixture of coated particles and matrix material (TiN, SiC or ZrC). The material of the structural parts was SiC.

After 2000, new interests were expressed by several countries to develop gas-cooled fast reactor designs following the formation of GIF. Important step of these new developments was the design of a small demonstration plant which has subsequently become known as ALLEGRO (C. Poette et. al., 2010). The ALLEGRO project intended to develop and qualify the innovative refractory fuel based on two successive core configurations. At first, the standard MOX core with metallic clad would be implemented at moderate temperature in order to irradiate more innovative refractory fuel at full scale. After this preliminary phase, a full refractory core, representative of the prototypical GFR, would be implemented.

The refractory fuel candidate concepts included several design versions as documented in (C. Mitchell et. al., 2006).

- Coated particle fuels with large kernels and thin coating layers
- Dispersion fuels in which small particles of the fuel are dispersed in a ceramic matrix
- Plate type fuel elements arranged within a basket, the basket structural reference material was SiC reinforced with SiC fibers
- Conventional pellet-cladding configurations, which would then require a ceramic cladding such as SiC, and the means to join and seal the materials

The most recent activities in the development of ALLEGRO design are carried out by the V4G4 international consortium. According to the present design:

- The first core of ALLEGRO will be built with MOX or UOX fuel in 15-15Ti stainless steel cladding

- The refractory fuel for the second core of ALLEGRO reactor will be composed of UC or (U,Pu)C pellets in SiCf/SiC cladding

1.4.3.2 Role in Safety Case

The UOX/MOX fuel with stainless steel cladding for the starting core and the carbide fuel with SiCf/SiC cladding for the refractory core will be operated at high temperatures. It is expected that large amount of fission products will be released from the pellets, so the integrity of cladding has a high importance in preventing the release of radioactivity from fuel. The application of ceramic SiCf/SiC cladding may need additional design features (e.g. introduction of sandwich type structure, special closing of the cladding tubes) to guarantee the leak-tight operation of fuel rods. The current GFR designs include additional barriers (such as a guard vessel and a containment structure) beyond the primary circuit components.

1.4.3.3 Challenges to Fuel Qualification

Qualification procedures have been proposed within GIF for the start-up and refractory ALLEGRO fuel. The technology readiness level approach was applied, and the basic steps of the qualification procedure were identified. Using the currently available information, the further needs were specified, which include experimental activities, design work, development of numerical models, technology developments, establishment of fuel fabrication capabilities, irradiation in research reactors and post-irradiation examination of fuel.

Specific challenge of the GFR fuel qualification is that the candidate fuel for the refractory core cannot be tested in prototypical GFR conditions in existing reactors. A gas-cooled loop in an experimental fast reactor could be used for preliminary testing. However, for the qualification of the real fuel design, first the ALLEGRO reactor with the standard MOX core with metallic clad will need to be built in order to irradiate more innovative refractory fuel at full scale.

1.4.4 Molten Salt Reactor (MSR)

Very few countries have embarked on a molten fuel salt qualification process. The most recent effort is in the USA where the NRC is currently working to develop an efficient and appropriate process. The NRC has engaged the USA Department of Energy’s Oak Ridge National Laboratory (ORNL) to develop the technical basis for liquid fuel salt qualification. The information presented here summarizes the recent ORNL report on fuel salt qualification (ORNL, 2020) as an approach that could benefit to other countries willing to enter into such a process.

1.4.4.1 Description

At a high level, MSRs can be differentiated by the chosen fuel form. Salt-fueled MSRs contain the primary fissile/fertile material as part of the circulating molten salt, while salt-cooled MSRs utilize a solid fuel that is separate from the molten salt coolant. Since the salt-cooled MSRs are based on TRISO fuel that is covered under VHTR section, this report focuses on salt-fueled MSRs.

In MSRs, fluoride and chloride salts are preferred over other alternative salts due to their suitable neutronic properties, compatibility with structural materials, ability to dissolve fissile/fertile materials, radiation stability, and more desirable thermophysical properties. Both fast and thermal neutron spectrum salt-fueled MSR designs are possible, with fluoride salts (such as FliBe with melting and boiling points of 460 and 1430 °C, respectively) more common for thermal designs while chloride salts (such as NaCl-MgCl2 with melting and boiling points of 445 and 1465 °C, respectively) are the preferred choice for fast reactor designs due to their considerably smaller moderating impact.

The properties of fuel salt will inherently change with operation due to the fission process, or they may change inadvertently due to contamination. Properties may also be purposefully changed due to refueling and chemistry adjustment. Finally, the extent to which the fuel salt boundary materials may contaminate the fuel salt, altering its thermochemical or thermophysical properties, will be an element of fuel salt qualification.

Liquid fuel salt, which consists largely of positive and negative ions, has no long-range structure and is continuously mixed as it flows through its circuit. Consequently, representative data can be obtained from small samples. Liquid fuel salts are largely immune from radiation damage, apart from transmutation, because the kinetics and thermodynamics of the ionic compounds that comprise liquid salts ensure rapid recombination of these compounds following radiolysis. That is, while radiation may break chemical bonds in ionic liquids, those bonds would almost instantaneously reform. Measuring properties of fuel salt samples provides a near real-time, direct update of the physical and chemical status of the fuel salt within the circuit.

1.4.4.2 Role in Safety Case

In a salt-fueled MSR, the fuel salt has two functions during normal operation; it contains the fissionable nuclei that constitute the nuclear fuel and it also serves as the reactor coolant. Liquid salt fuel has three safety functions. First, it retains radionuclides; second, it is the heat transfer media for both operational and decay heat removal; and third, it must provide a net negative reactivity feedback during upset conditions.

Adequate fuel performance for liquid salt-fueled MSRs can be assured by maintaining fuel chemistry within a predetermined set of bounding values, providing reasonable assurance that the fuel safety functions are maintained. The bounding values for the salt properties are those that assure adequate safety under both normal and accident conditions. The required set of values are to be determined by performing accident progression analysis.

Liquid salt is not intended to retain all radionuclides during normal operation. The amounts and forms of radionuclide release from fuel salt will be key inputs to the overall reactor radionuclide release safety case. Fuel salt qualification requires development of a sufficient understanding of the physical and chemical behavior of the fuel salt to ensure it performs adequately under normal operation, anticipated operational occurrences, design basis accidents, design extension conditions.

1.4.4.3 Challenges to Fuel Qualification

Uncertainties need to be defined so that calculated fission product releases include the appropriate margins to ensure conservative calculation of radiological dose consequences.

A surveillance-based approach to developing an adequate understanding of the physical and chemical behavior of fuel salt has two aspects. Prior to operation, a fuel salt property database is empirically generated, providing a mapping of fuel composition and temperature to thermochemical and thermophysical properties. The database must be of adequate fidelity to enable mapping the acceptable boundaries of the fuel safety-related properties. The bounding properties database provides confidence that no safety-related fuel salt properties would be reasonably anticipated to exceed their acceptable limits as specified in the plant's technical specifications prior to their next measurements. Since the salt properties do not depend on the isotopic composition of the salt, the database can be constructed using non-radioactive or minimally radioactive isotopes (for those elements that do not have non-radioactive isotopes) of the elements comprising the fuel salts. As the fuel salt properties change over time—inherently due to the fission process and corrosion, inadvertently due to contamination, and purposefully due to refueling and chemistry adjustment—measurements will be performed periodically to confirm that the safety-related fuel salt properties remain within acceptable limits.

Retaining radionuclides within containment is a fundamental element of facility safety. Multiple layers of essentially leak-tight radionuclide barriers would need to be breached or bypassed for radionuclides to be released to the environment. In order to qualify fuel salt, adequate knowledge must be available to model the role of the fuel salt in initiating a breach or bypass of any containment layer, as well as the salt's continuing capability to retain radionuclides. Mechanistic models of the chemical and physical interactions of the fuel salt—both directly with the containment materials and indirectly with other materials within containment under accident conditions—are central to developing potential accident source terms.

Providing reasonable assurance of the continued capability to reject decay heat requires monitoring changes in the natural convection heat transfer properties of the fuel salt. The capability of a particular reactor design to adequately reject decay heat with fresh fuel salt will initially be established through thermal and hydraulic modeling and experimentation. Fuel salt viscosity, density, and heat capacity as a function of temperature are thus key measurements and elements of a fuel salt property database to ensure the continued ability to passively reject decay heat.

Fuel salt is required to provide net negative reactivity feedback with increasing temperature in the power range of operation. Reactor physics for any reactor is both computationally predicted and experimentally validated during the design process. The predictions are confirmed during start-up

testing by performing reactor physics confirmatory experiments. MSRs will follow a conventional reactor physics development and design route to ensure that the initially loaded fuel system provides a net negative reactivity feedback. Experimental validation of reactor physics simulations performed with used fuel salt provides assurance that the reactor system will continue to have net negative reactivity feedback with the increased fissile loading and change in fissile isotopic composition. Reactor feedback properties can be obtained by monitoring the change in the neutron flux due to a small, rapid reactivity stimulus. Controlled reactivity stimuli can be provided through various means without expensive design or operational changes.

Since, outside the molten salt reactor experiment (MSRE) experience dating back to 1960s, there has been no other experience with MSRs, implementation of a thorough fuel qualification process to demonstrate the various attributes enumerated above will likely remain as a major challenge for future MSR projects.

1.4.5 Lead-Cooled Fast Reactor (LFR)

LFRs share many technological aspects with SFRs, including the fuel. However, the very high boiling point of lead opens the potential for operation at higher system temperatures, if corrosion- and creep-resistant materials are suitably qualified. The lower neutron absorption of lead also tends to allow a harder neutron spectrum. Also, like all fast reactor systems, LFRs have the possibility to operate at very high burnups, provided that structural materials capable of withstanding high irradiation doses are available. Accordingly, fuel systems of LFR projects are differentiated between near- and long-term initiatives, the former ones necessarily resorting to the use of materials with high technological readiness, hence envisaging operation within burnup and temperature ranges similar to those already experienced in past SFR projects. The LFR projects to be deployed in the mid- to long-term perspective rely on the successful achievement of development programs and the qualification of higher performance fuels and materials.

1.4.5.1 Description

All LFR projects currently being developed consider pin-type fuel systems, normally with pelletized fuel or rodlets in structural cladding tubes. Almost all SFR fuel types can be used in LFRs including oxides, nitrides, carbides and metallic fuel forms. The fuel pins are usually provided with a plenum to limit the pressure inside the cladding due to fission gas release, and are arranged in bundles supported by grids (to achieve larger pitch-to-diameter ratio) needed to extract commercial power levels given the low heat capacity of lead, or possibly wire raps for low-power assemblies. Both wrapper-less designs (without an assembly duct) and those, where pins are enclosed within a steel wrapper (duct), are considered.

For near-term LFR initiatives, oxide and nitride fuels in steel claddings are considered in most designs. Either enriched uranium or mixed uranium-plutonium oxides and nitrides are selected. For cases involving oxide fuels, cladding is envisioned to be composed of advanced, titanium-stabilized austenitic steels (e.g., AIM1, D9I), such as those implemented in the latest reloads of past SFRs. However, liquid metal embrittlement of ferritic-martensitic steels have excluded these steels from being considered as a

structural material in Pb-alloy cooled reactors (IAEA 2020, p. 179). Localized corrosion in LFRs is a key issue that is not yet fully understood as demonstrated by the following examples:

- The development of a chromium rich oxide layer on the steel surface that acts as physical barrier to further oxidation in most environments, is not effective in heavy liquid metals at high temperature, and above 450~500° C severe corrosion attacks are observed in both austenitic and ferritic-martensitic steels, with the formation of thick not protective oxide layers, internal oxidation and dissolution of the steel in the coolant. In general it has been shown that above temperatures around 450~500°C, depending on the steel considered and on the working conditions, the operation in heavy liquid metal under control of the oxygen concentration is not effective in keeping a thin passivating surface layer (IAEA 2020, p. 195-197).
- Protection with surface oxide films (like other ceramic coatings) is temporary. The films are destroyed under the influence of static and dynamic stresses as a result of mechanical damage, erosion and thermal cycling due to the difference in the linear elongation coefficients of the oxide and metal during thermal cycles. Damage to the oxide film leads to the penetration of lead into steel and intense localized corrosion, the rate of which is many times higher than the rate of oxygen corrosion (IAEA 2020, p. 180).

In the case of the BREST-OD-300 reactor, nitride fuels are envisioned, capitalizing on technological developments carried out in Russia, and facilitating the implementation of an advanced pyrochemical fuel recycle. Note that this fuel option and the associated fabrication and recycle technologies are also included in the licensing dossier presently under evaluation by the Russian regulatory body. The nitride fuel system includes cladding based on silicon-doped ferritic-martensitic steel. Nitride fuel testing and qualification is currently underway with nitride fuel assemblies being irradiated both in BN-600 and BOR-60, a good example of exploitation of the expected synergies between SFR and LFR as nitrides are envisaged for use also in SFRs in Russia. However, it should be noted that tests of fuel pins in sodium reactors are insufficient for qualification and licensing of fuel for a LFR, since in these tests lack one of the main damaging factors - the negative effect of a lead coolant on corrosion and mechanical properties of claddings. Furthermore, testing of claddings by integral test of pins in reactor is necessary because early and hard contact between the fuel pellets and the cladding through splinters (crumbs) of nitride fuel in the gas gap causes large local stresses in the cladding. These stresses destroy the protective oxide film on the fuel rods surface and this leads to a significant acceleration of the cladding corrosion form the Pb or Pb-Bi coolant.

The choice of the specific combinations of candidate fuel and cladding materials is typically performed based on the extension of the knowledge base upon which the qualification and licensing process can build. Additionally, and apart from few (typically micro-reactor) designs adopting limited coolant outlet temperatures, protective cladding coatings (surface modifications) are typically considered to improve corrosion resistance in lead coolant. The most promising coatings, considered in many designs, are based on introduction of alumina through physical vapor or atomic layer deposition, although alumina-forming alloys are also considered.

For LFR projects to be deployed in the long-term, nitride fuels are often considered as a promising option. As cladding materials, several options are being investigated, including double-stabilized austenitic steels (e.g., AIM2 in France), improved ferritic-martensitic steels, oxide-dispersed steels

(ODS), alumina-forming austenitic (AFA) steels, ferritic alumina-forming (FeCrAl) steels, and even SiCf/SiC ceramics or multilayered structures.

1.4.5.2 Role in Safety Case

In accordance with the principle of Defense-in-Depth, LFRs adopt several levels of protection including successive safety barriers preventing the release of radioactive material to the environment (GIF, 2020). These include, as usual, the fuel matrix, the cladding, the boundary of the reactor primary coolant system, and the building containment system. However, thanks to the properties of lead coolant, another barrier is effectively present – i.e., the primary coolant itself - and needs to be taken into consideration in the case of LFRs. Both analytical and experimental results indicate that lead has a superior tendency to retain volatile fission products like iodine and cesium, as well as polonium, out of which only a small fraction is expected to be released to the cover gas even in accident conditions, so that the building containment system is not expected to play a fundamental role in the overall limitation of the source term to the environment.

For the fuel cladding, design limits are established based on the maximum temperature and burnup (residence time), with due attention to the accumulated corrosion-erosion and irradiation effects. This includes the data on fuel-cladding interactions, which for MOX types of fuel benefit from the extensive operating and qualification feedback conducted in the frame of SFR programs. The fuel failures experienced by pellet-clad interactions during rapid transients can be prevented by introducing a barrier layer in between and/or by employing oxide pellets with chromia-doping.

In case of cladding failure, no safety-related phenomena are expected to additionally come into play when fuel-coolant interactions take place. Recent experiments have provided evidence of the absence of exothermal chemical reactions between lead (or lead-bismuth) and both oxide and nitride fuels. The chemical stability of breached fuel may be especially important in LFR-based micro reactors for ship propulsion in which the fuel is designed to last for the entire hull lives.

Challenges for lead reactors include providing reliable shutdown in an environment where oxide CRUD can coat surfaces causing blockages or rendering them immobile. Freezing of lead is also an issue as is preventing leaks from the primary coolant circuit.

1.4.5.3 Challenges to Fuel Qualification

Some similarities of the fuel system in LFRs with that of past SFR projects provide some background for qualification; The behavior of the fuel in representative spectrum irradiation, the compatibility with the cladding materials, the resistance of the cladding to representative irradiation conditions and thermo-mechanical loads, have been studied and some results can be transferred to present LFR designs.

Novel challenges are posed instead in relation to the verification of corrosion-erosion resistance (also of known fuel systems), or even the qualification of protective coatings (whenever foreseen) (GIF, 2019). Zirconium-based barriers have been used in thermal reactors, and their qualification for LFR fuels may be explored. However, as discussed in Section 1.4.5.1, localized corrosion in LFRs is a key

issue that is not yet fully understood. The effect of the coolant on the corrosion and mechanical characteristics of the cladding under operating conditions is a key issue that needs further investigation as part of an irradiation testing program.

Other challenges for fuel qualification reside in the high level of quality required to be assured at the production line - for both fuel, cladding (and coating if present) as a whole – and are strongly conditioned by the availability of irradiation capability in the proper coolant environment. This mission is expected to be taken up by the construction of dedicated LFR demonstrators able to provide the needed conditions for fuel and cladding qualification. It is also important to note that the final fuel choice will also be strongly conditioned by the availability of reprocessing techniques able to ensure efficiency, cost and safety of the overall fabrication, re-fabrication and disposal processes.

A program for nitride fuel qualification has been recently implemented in Russia. This program is aimed to license the first BREST fuel load. The program covers both fuel rods for an LFR with a cladding made of ferritic-martensitic steel, and fuel rods for an SFR with a cladding made of austenitic steel. In 2020, at EFA-11 (Experimental Fuel Assembly number 11) tests are planned to achieve burnup of ~ 9 t.a. and a damaging dose of more than 100dpa. The EFA-11 fuel rods are the prototype of the BREST-OD-300 reactor fuel rods. From 2021 to 2027, it is planned to irradiate the IA (Irradiation Assembly) with a burnup of 12% and a damaging dose of 120 dpa. The start of fuel assembly irradiation in the BREST reactor is planned in 2027 (M.V. Skupov and L.M. Zabudko 2019).

1.4.6 Super-critical Water-Cooled Reactor (SCWR)

Within GIF, the two main conceptual SCWR designs are being pursued by Canada, the EU, and Japan: The SCWR designs with conventional reactor pressure vessel are analogous to conventional LWRs, and the SCWR designs with distributed pressure tubes (PTs) or pressure channels are analogous to conventional heavy water reactors (HWRs).

1.4.6.1 Description

The protype SCWR designs will likely rely on LWR experience with sintered UO2 or MOX fuel pellets stacked into corrosion-resistant metal alloy cladding such as Alloy 800H, 310- and 316-SS. The cladding tubes are pressurized with helium to try to minimize pellet-cladding interaction which can lead to fuel rod failure over long periods. The fuel rods are bundled in fuel assemblies with spacer grids providing desired pitch-to-diameter ratio.

The Canadian SCWR concept adopts an advanced thorium-based fuel cycle using an oxide mixture of thorium with reactor-grade plutonium, recovered from used LWR fuel. This alternative, rather than the use of low enriched uranium, is intended to extend the lifetime of natural uranium reserves. The European SCWR concept envisages using UO2 fuel with 235U-enrichment at approximately 7%, which is higher than the current Pressurized Water Reactor (PWR) levels to improve effective fuel utilization. Neither change is an inherent part of the SCWR design concept, which essentially takes a PWR and operates it at higher pressure and temperature, where fluid properties are different.

1.4.6.2 Role in Safety Case

The SCWR system is considered to be an evolutionary concept since it is based on water-cooled thermal-spectrum reactors such as PWRs, Boiling Water Reactors (BWRs), and PT-HWRs that are in widespread use today. Therefore, the role of fuel in the safety case is expected to be similar, if not identical, to the case for LWRs. Since the thorium-based fuel has a much higher melting point and thermal conductivity than uranium oxide fuel, the Canadian SCWR concept offers different safety characteristics of the fuel although the impact of mixing thorium with plutonium in oxide pellets needs further study. For the European SCWR concept, the impact of higher enrichment, leading to potentially higher burnup, will have to be studied as well. Additionally, the pressure sensitivity of the physical properties introduces particular considerations of dynamic stability and the coupling between neutronic feedback and hydraulics. This may need development of the fuel assembly configuration; balancing the need for negative thermal feedback with that of an adequate kinetic response to pressure waves. There will also be issues relating to the chemical kinetics of oxidation and deposition processes in a neutron flux.

1.4.6.3 Challenges to Fuel Qualification

The SCWR designs are based mainly on the water-cooled thermal-spectrum LWRs; therefore, the fuel qualification process and the requirements are expected to be identical to LWRs. The characteristic and performance of the thorium-based fuel and the higher enrichment UO2/MOX fuel forms will require additional irradiation and safety testing experience that can be performed in existing operating and test reactors.

2. Overview of Current Regulatory Approaches

2.1 Regulations/Laws on Nuclear Fuel Qualification

It is generally recognized that fuel qualification is a part of the overall licensing of a nuclear facility. As such, the requirements on fuel qualification are provided by top-level requirements attributed to the nuclear facility.

2.2 Regulatory Processes for Qualifying Fuel

The regulatory process for fuel qualification is generally implicit in the overall licensing of a nuclear facility. In addition, several countries have established practices for implementing new fuel designs into existing reactors that provide insight into the fuel qualification process for advanced reactors. Some of these countries use processes similar to the USA NRC topical report process. This process allows for the generic review of a safety topic and can be applied to several licensing applications. Common examples that utilize the topical report process include the analysis methodologies (e.g., loss-of-coolant accident evaluation methods, fuel performance codes) and fuel designs. The objective of the topical report process is to obtain approval on a generic safety issue that can be referenced in future licensing applications, thus supporting an efficient licensing review. The advantages of this process are that it facilitates a staged review and reduces regulatory uncertainty by obtaining approval on critical safety issues prior to the submittal of a licensing application.

In other regulatory regimes, the assessment employs a graded approach, which looks at safety considerations in a holistic way and targets particular areas for assessment in proportion to their importance to safety of a facility. Targeting can reduce the regulatory burden where safety margins are large, but cannot examine particular items in isolation from their context. Targeted approaches are typically supported by generic guidance and regulatory risk is reduced by early engagement with the vendors as part of a generic design assessment process such as that employed within the UK.

2.3 Guidance for Nuclear Fuel Qualification

Updated fuel designs that are similar to existing fuel designs are generally evaluated by considering known fuel failure and degradation mechanisms (NEA, 2012) and (USNRC, 2007). Recently, guidance has been developed for specific fuel campaigns which identifies additional new fuel degradation mechanisms (USNRC, 2020). However, regulatory guidance to address fuel qualification for advanced reactor fuel is, in general, not available. Nonetheless, there is a general consensus that fuel needs to be qualified for operation, either through experience with the same type of fuel in other reactors, or through a program of experimental testing and analysis (CNSC, 2014), (USNRC, 2017), (IAEA, 2019). It is generally recognized that fuel qualification needs to include the following:

- A defined test envelope based on expected conditions during operational states (normal operation and anticipated operational occurrences) and accident conditions (design basis accidents and design extension conditions).
- An irradiation testing program, which includes full-scale integral testing, to identify fuel failure and degradation mechanisms
- Transient testing to assess fuel performance under transient and accident conditions
- A demonstration that the controls on the manufacturing process will deliver adequate levels of reliability.

In order to further investigate these high level considerations and to examine other needs associated with fuel qualification, a systemic evaluation of requirements is performed in Section 3 in order to identify a series of objective criteria for fuel qualification.

2.4 Summary of Input to WGSAR Report

Written input to the fuel qualification was provided by WGSAR members from the UK, France, Italy, and the USA. This input addressed fuel qualification considerations in the areas of regulatory requirements, fuel safety criteria, and best practices.

There was general consensus that prescriptive regulatory limits or regulations specific to fuel qualification are not available, but fuel qualification requirements are driven by higher level safety objectives. The UK stated that they do not have prescriptive regulatory limits (limits are the burden of the licensee), but that there are principles associated with maintaining cladding integrity, coolable geometry, and limiting radiological consequences. France stated that there is no specific regulation for fuel qualification, and that the qualification of the computational codes related to fuel behaviour is required. The French Safety Authority, Autorité de sûreté nucléaire (ASN), has rules in place on the qualification of computational tools. Italy similarly did not identify specific regulatory criteria for fuel qualification and stated that safety criteria should account for margin to address uncertainties consistent with "Safety Margins Actions Plan – Final Report," NEA/CSNI/R(2007)9, 2007. The USA identified high-level regulatory criteria (Title 10 of the Code of Federal Regulations (CFR) 50.43(e) and 10 CFR 50 Appendix A, General Design Criteria) that are associated with fuel qualification and noted that specific regulatory criteria and regulatory guidance for fuel qualification are not available. However, the regulatory authority in the USA is required to develop fuel qualification guidance for advanced reactor fuel in accordance with the Nuclear Energy Innovation and Modernization Act.

There was general consensus that a fuel performance or operating envelope needs to be specified under which fuel failure and degradation mechanisms are controlled, and that identification of fuel failure and degradation mechanisms over this envelope should be supported by an irradiation testing program including post irradiation examination (PIE). Input from the UK identified the need to specify an operating envelope and to perform experiments within that envelope so that performance limits can be quantified to define the safe operating rules. France stated that the fuel safety assessment relies on the analysis of the principles, requirements, and safety criteria and methodology of accident studies. The

data requirements in France appear to be driven by the need to qualify the computational codes related to fuel behaviour. The input from Italy clarified that fuel failure mechanisms needs to be identified and evaluated. Italy identified fuel failure mechanisms for traditional light water reactor fuel (i.e., Zircalloy clad UO_2 fuel) and referenced "Nuclear Fuel Safety Criteria Technical Review," 2nd edition (2012). The input from the USA identified that a performance envelope needs to be specified to cover conditions of normal operation, including the effects of anticipated operational occurrences (AOOs), and accident conditions. Additionally, the USA stated that criteria need to be established to evaluate (1) fuel performance under conditions of normal and AOOs, (2) barrier degradation under accident conditions, and (3) assurance of safe shutdown (i.e., a safe state in accordance with the International Atomic Energy Agency (IAEA) glossary).

Additionally, some best practice considerations were discussed. The UK identified (1) the need for pilot irradiation and integral power ramp data, (2) modeling limitations associated with the reliably prediction of cladding stress, and fuel pellet fragmentation and relocation, (3) the need to consider sampling strategies for manufacturing quality control, (4) the need to set conservative operating limits, and (5) areas of uncertainty for novel fuel designs. France identified that enhanced nuclear modeling capacities and improved knowledge on fundamental physical phenomena may allow for better characterization of safety margins. The USA identified that (1) the fuel manufacturing process can have a significant impact on fuel performance and should be considered as an element of fuel qualification, (2) accelerated fuel qualification techniques are being investigated that rely on advanced modeling and simulation to reduce fuel development timelines (K.A. Terrani et. al., 2020), and (3) the use of an assessment framework in other areas of review has resulted in more efficient and comprehensive safety reviews (USNRC, 2019).

2.5 Common Positions

The following common positions were identified based on WGSAR member input on fuel qualification:

1. Fuel qualification requirements are derived from higher level nuclear power plant requirements (e.g., protection of "first barrier", accident source term, assurance of coolable geometry). However, guidance specific to the fuel qualification process is not generally available.
2. Process for qualifying fuel is generally implicit in the overall licensing of the nuclear facility.
3. An essential part of fuel qualification is to define a test envelope to cover expected operating, transient, and accident conditions to assess fuel performance and validate fuel performance codes.
4. An irradiation testing program, which includes testing of the integral fuel design, is necessary to identify fuel failure and degradation mechanisms. This testing should provide irradiation covering the exposure or burnup limits of the fuel.
5. Fuel qualification requires transient testing to assess fuel performance under transient and accident conditions.

3. Identification of Fuel Safety Criteria

This section systematically identifies fuel safety criteria. The comprehensive list of safety criteria is referred to as a fuel assessment framework and is informed by existing regulatory requirements, regulatory guidance, and experience performing safety reviews for nuclear fuel in light water and non-light water reactors (non-LWRs). The fuel assessment framework is developed using a top-down approach that starts with the high-level goal (G) that the fuel is qualified to achieve its safety functions. Consistent with the purpose of fuel qualification, which is discussed in Section 1.1, and with a regulatory focus on safety, fuel that is qualified for use means that high confidence exists that the fuel fabricated in accordance its specification will perform as described in the applicable licensing safety case. This statement is captured figuratively in Figure 3-1, which decomposes fuel qualification into two supporting goals. These goals are further decomposed into lower level supporting goals. This process is continued until objective criteria are identified which can be directly supported by evidence. The process, criteria, and associated evidence are described in the subsections that follow.

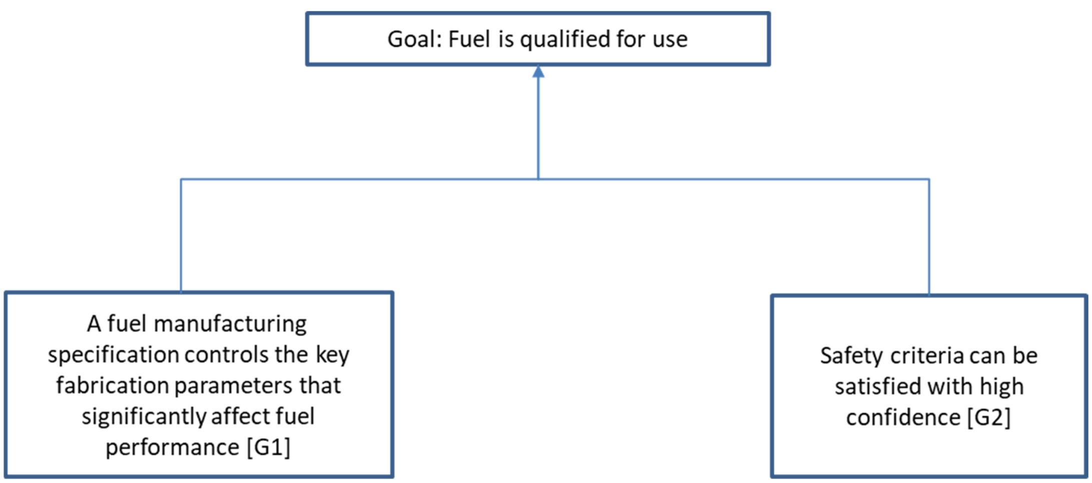

Figure 3-1. Decomposition of the main goal

3.1 G1 Fuel Manufacturing Specification

It is well known that fuel performance during normal operation and accident conditions can be highly sensitive to the fuel fabrication process. For example, failure criteria during reactivity induced accidents for LWRs with zirconium-based cladding depends upon the heat treatment of the cladding (due to the impact on microstructure) (USNRC, 2016), and on the avoidance of fuel pellet damage during manufacture. Additionally, the Electric Power Research Institute (EPRI) in the USA developed a report that identifies key manufacturing parameters for TRISO fuel that EPRI stated must be controlled in

order to ensure satisfactory performance (EPRI, 2019). It is recognized that manufacturing processes for a nuclear fuel product can evolve over the product lifecycle, and therefore, a complete manufacturing specification is not expected to be included in licensing documentation. However, sufficient information needs to be included in the licensing documentation to ensure that key parameters affecting fuel performance are controlled during the manufacturing process, and that the fuel functional requirements will be met with a high level of confidence. Accordingly, this goal is decomposed in Figure 3-2 to identify the type of information that should be included in licensing documentation. Compliance with these objectives will need to be confirmed by the results of a formal qualification of the manufacturing process, including demonstration of the adequacy of non-destructive and destructive sampling schemes.

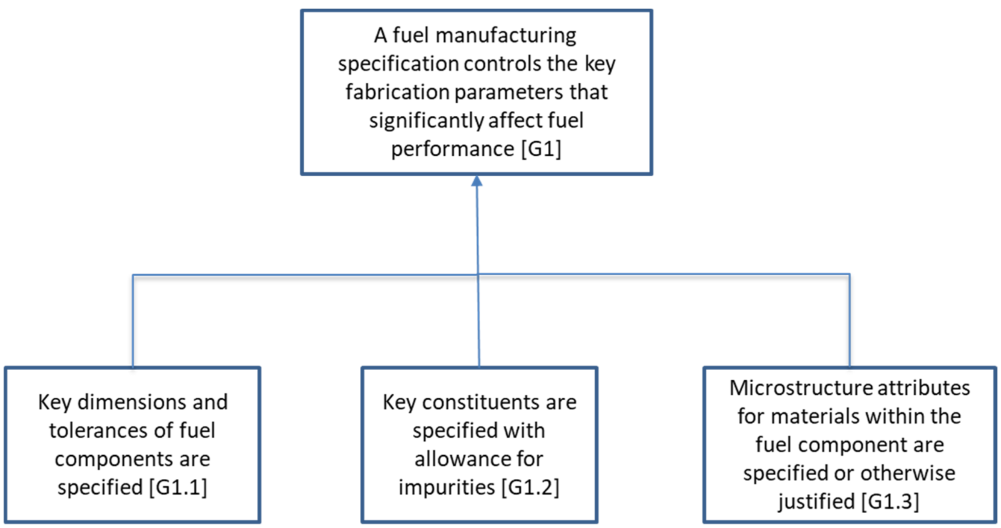

Figure 3-2. Decomposition of G1 – Fuel Manufacturing Specification

3.1.1 G1.1 Dimensions

Key dimensions and tolerances for fuel components that affect performance should be specified. Consistent with the scope of this report, as discussed in Section 1.3, these dimensions should be specific to components that impact fuel life limiting failure and degradation mechanisms that occur as a result of irradiation during reactor operation (e.g., fuel pellet and cladding dimensions). This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.1.2 G1.2 Constituents

Key constituents of fuel components (e.g., UO_2 fuel, U-Pu-10Zr fuel, cladding material, fill gas, thermal-bonding material) should be specified along with allowances for impurities, alloying elements, and dopants. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.1.3 G1.3 Microstructure

Attributes of the microstructure for the materials within fuel components should be specified or otherwise justified. It is noted that the microstructure of a material represents the desired end state of the material and this type of information may be captured in several ways. For example, specifying specific manufacturing processes (e.g., cold-working, heat treatments, deposition techniques, etc.) that are essential to create the desired end state may be specified in lieu of specifying microstructure attributes as process control parameters, but some sampling of microstructure will generally be expected to confirm the adequacy of the process control or fabrication process. Sufficient justification should be provided in licensing documentation for cases where an insensitivity to microstructure and manufacturing processes is present for a specific material. The likelihood that such an argument would be accepted would depend on the magnitude of the safety margin and the importance of the safety function under consideration. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.2 G2 Safety Criteria

An evaluation of the safety case involves an assessment against safety criteria which are associated with the protection against the release of radioactive material. In general, there are many safety criteria associated with nuclear fuel that are dependent upon the event under which the fuel is subjected. Specifically, nuclear fuel is expected to retain its integrity under conditions of normal operation, including the effects of AOOs, but some degree of fuel failure may be unavoidable for low frequency (i.e., not expected to occur during the life of the plant) design basis accident conditions and this may be acceptable provided that other barriers to release are not consequently compromised. Accordingly, this goal is decomposed in Figure 3-3 to address the varying types of safety criteria associated with the range of events for which nuclear fuel must be qualified.

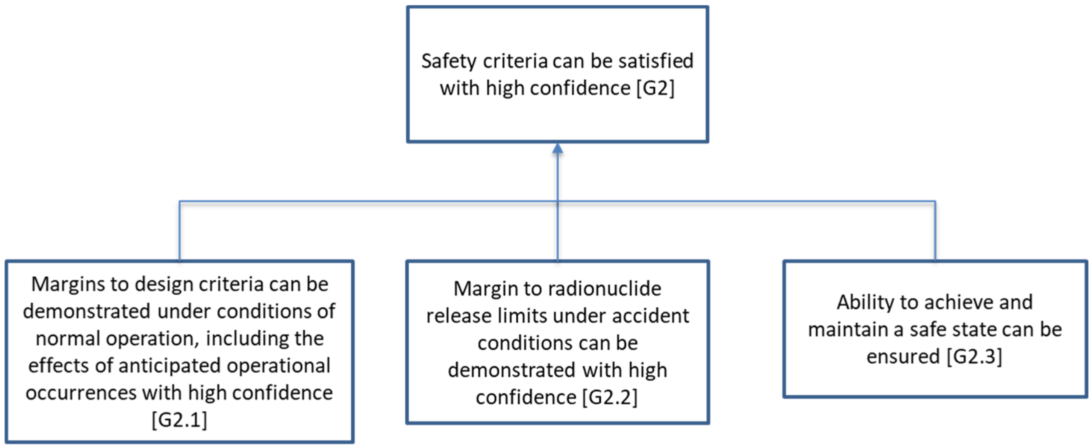

Figure 3-3. Decomposition of G2 – Safety criteria

3.2.1 G2.1 Design Criteria Under Conditions of Normal Operation, Including the Effects of AOOs

Fuel integrity is required to remain intact under conditions of normal operation, including the effects of AOOs such that failure of a fission product barrier does not occur and continued safe operation can be justified. Multiple fuel failure and degradation mechanisms may exist and limits need to be established to protect against those failure and degradation mechanisms. At the highest level, the assessment of a fuel against design limits for normal and off-normal operation requires knowledge of the conditions that the fuel is exposed to (i.e., the performance envelope) and a method to assess the fuel performance under those conditions (i.e., an evaluation model). These supporting goals are captured in Figure 3-4 and discussed in the subsections below.

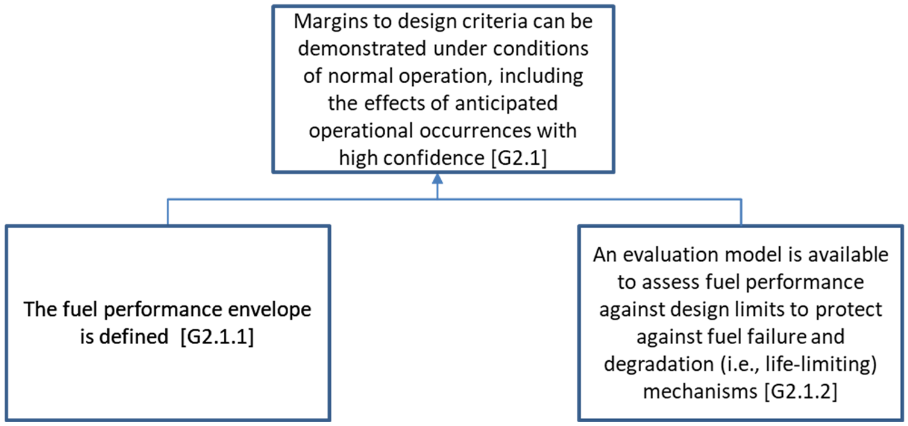

Figure 3-4. Decomposition of G2.1 – Margins to design criteria under conditions of normal operation, including the effects of AOOs

3.2.1.1 G2.1.1 Definition of Fuel Performance Envelope

The fuel performance envelope specifies the environmental conditions and radiation exposure under which the fuel is required to perform. The envelope may be specified by fuel designers and provide constraints on the design of the reactor and associated systems. Alternatively, a reactor design can be proposed that places requirements on fuel performance. G2.1 is satisfied by specifying the environmental conditions (e.g., temperatures, pressures, power), exposure, and transient conditions that the fuel is expected to encounter during normal operations and AOOs. Additionally, this goal supports G2.2 associated with the fuel contribution to source term during design basis accidents and design extension conditions and is further discussed in Section 3.2.2.1. Accordingly, this goal is fully satisfied by specifying the environmental conditions the fuel is expected to encounter during conditions of normal operation, AOOs, and design basis accident conditions to which the fuel is subject. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.2.1.2 G2.1.2 Evaluation Model

An evaluation model is available to assess fuel performance against design limits to protect against fuel failure and degradation mechanisms requires the specification of the means by which fuel is evaluated for performance, failure, and degradation. Assessment of an evaluation model is an area of review that supports several goals and requires further decomposition into several supporting goals. Therefore, a separate assessment framework for evaluation models is provided in Section 3.4 of this report. G2.1.2 is satisfied by satisfying the supporting goals in the evaluation model assessment framework in Section 3.4 of this report for fuel performance during conditions of normal operation, including AOOs.

3.2.2 G2.2 Radionuclide Release Limits

Radiological consequences under design basis accident and design extension conditions are an essential consideration regarding nuclear power plant licensing. Under postulated accident conditions some amount of fuel failure is possible and results in a contribution to the accident source term. As radionuclide inventory originates from the nuclear fuel, part of fuel qualification must involve characterizing the behaviour of the fuel under accident conditions such that the fuel contribution to accident source term can be determined in a suitably conservative manner. Accordingly, the ability to demonstrate margin to radionuclide release limits under accident conditions, as it relates to fuel qualification, is supported by four goals identified in Figure 3-5 that are related to characterizing the fuel contribution to accident source term. These goals are discussed further below.

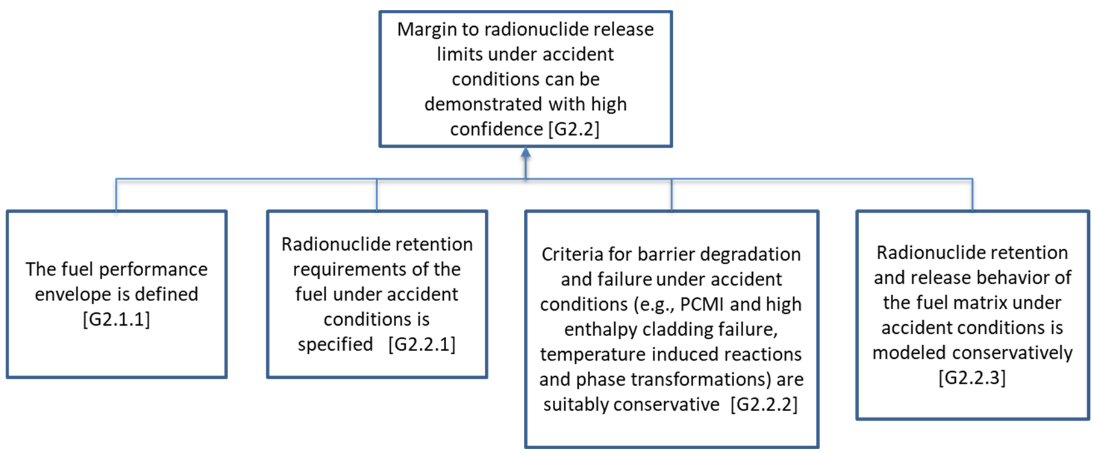

Figure 3-5. Decomposition of G2.2 – Margin to radionuclide release limits

3.2.2.1 G2.1.1 Definition of Fuel Performance Envelope

G2.1.1 is the same goal as was discussed in Section 3.2.1.1. In support of G2.2, this goal is satisfied by specifying the design basis accident and design extension conditions to which the fuel is subject. Design basis accident and design extension conditions are dependent on reactor design. However, as discussed in Section 3.2.1.1, the conditions to which the fuel is subject under design basis accidents may, to some degree, be specified independent of the reactor design resulting in constraints on the design of the reactor and associated systems. The types of design basis accident and design extension conditions that should be considered include transient overpower events (e.g., reactivity inducted accidents) and transient undercooling events (e.g. loss-of-coolant accident). This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.2.2.2 G2.2.1 Radionuclide Retention Requirements

The role that nuclear fuel plays in the safety case can vary between reactor designs and fuel types. For example, traditional light water reactor fuel that utilizes uranium dioxide pellets with zircalloy cladding is not assumed to retain cladding integrity under large break loss-of-coolant accidents. Advanced reactor designs may propose to credit retention of radionuclides within the fuel under accident conditions. To satisfy this goal, the degree to which radionuclide retention within the fuel system should be specified. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.2.2.3 G2.2.2 Criteria for Barrier Degradation

Radionuclide barrier (e.g. fuel cladding) failure and degradation mechanisms under accident conditions need to be understood when integrity of the barrier is credited (e.g., cladding integrity during reactivity induced accidents in LWRs, fission product attack of SiC layer in TRISO fuel at high temperatures). This goal is decomposed into two supporting goals in Figure 3-6.

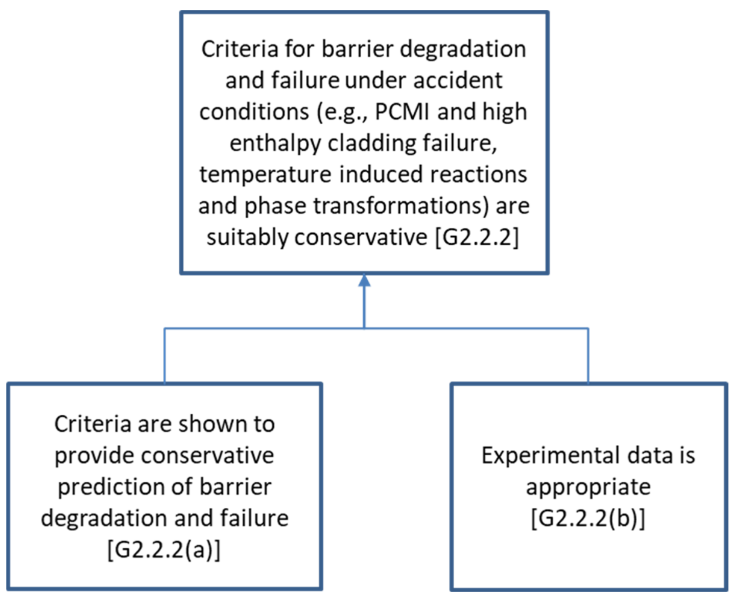

Figure 3-6. Decomposition of G2.2.2 – Criteria for barrier degradation

3.2.2.3.1 G2.2.2(a) Demonstration of Conservative Criteria

Criteria used to determine barrier degradation should be suitably conservative. These criteria are expected to be established based on transient testing and irradiated fuel samples which is further discussed under G2.2.2(b). Ideally, criteria would be established through a regression analysis using

experimental data, and then validated by assessment against a separate and independent set of data (see Section 3.4.1, ED G1 for the discussion on data independence) in order to establish a statistical confidence level (e.g., 95/95). However, this ideal scenario is not expected to be realized for prototype designs due to economic and environmental concerns associated with obtaining irradiated fuel samples and conducting transient testing in accordance with design basis accident conditions. Past experience from transient overpower testing in the USA has shown that it may be acceptable to develop realistic criteria for barrier degradation using fewer data points (USNRC, 2015) provided that the data is sufficient to demonstrate an adequate level of reliability at an appropriate statistical confidence level. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.2.2.3.2 G2.2.2(b) Experimental Data

This goal is satisfied through an evaluation against the assessment framework for experimental data provided in Section 3.4 of this report.

3.2.2.4 G2.2.3 Conservative Modeling of Radionuclide Retention and Release

Consistent with the radionuclide retention requirements specified as part of G2.2.1 and discussed in Section 3.2.2.2, radionuclide retention and release behavior of the fuel under accident conditions should be modeled conservatively. This goal is related to the barrier degradation criteria specified in G2.2.2 and discussed in Section 3.2.2.3, but is distinct in its focus on radionuclide retention within the fuel matrix.(e.g., UO_2 pellet or U-10Zr fuel ingot) or fuel particle (e.g., fuel compact for a TRISO based fuel). This goal is decomposed into two supporting goals in Figure 3-7.

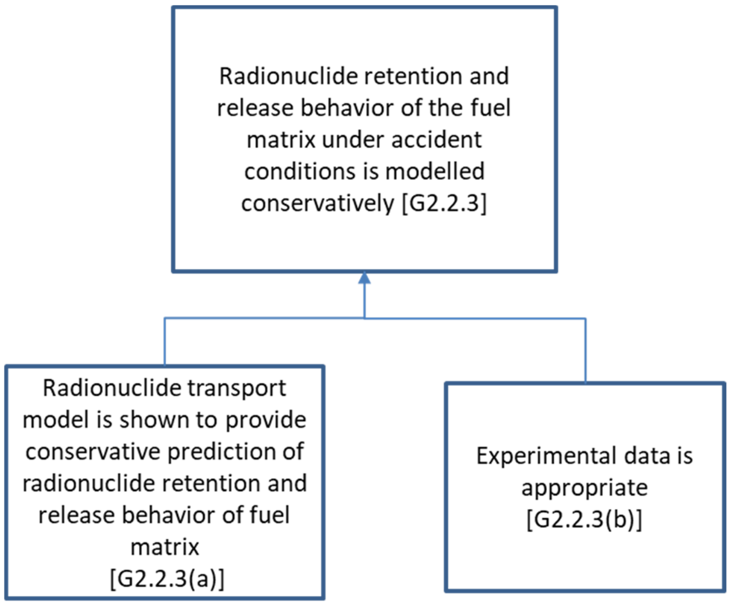

Figure 3-7. Decomposition of G2.2.3 – Radionuclide release modeling

3.2.2.4.1 G2.2.3(a) Demonstration of Conservative Transport Model

Radionuclide transport modeling from the fuel should be conservative. Similar to the scenario for barrier degradation criteria, discussed in Section 3.2.2.3.1, economic and environmental concerns may inhibit the ability to obtain significant amounts of data such that conservative or bounding estimates may be required. Additionally, experience with source term models for LWR have included some degree of expert judgement. A clarifying example of how a suitably conservative radionuclide transport model can be developed in the USA is available in regulatory guidance on accident source term (USNRC, 2000). This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.2.2.4.2 G2.2.3(b) Experimental Data

This goal is satisfied through an as evaluation against the assessment framework for experimental data provided in Section 3.4 of this report.

3.2.3 G2.3 Achieving and Maintaining a Safe State

A safe state refers to a plant state following an AOO or accident condition in which the reactor is subcritical and the fundamental safety functions (control of reactivity, cooling of radioactive material, and confinement of radioactive material) can be ensured and maintained stable for a long time The ability to achieve a safe state under any scenario needs to be ensured (IAEA, 2018). Ensuring that this safe state can be achieved requires, in part, criteria for the fuel to ensure that a coolable geometry is maintained under all scenarios and that fuel system damage is very unlikely to be so severe as to prevent control element (e.g., control rods) insertion when it is required.. These supporting goals are captured in Figure 3-8 and discussed in the subsections below.

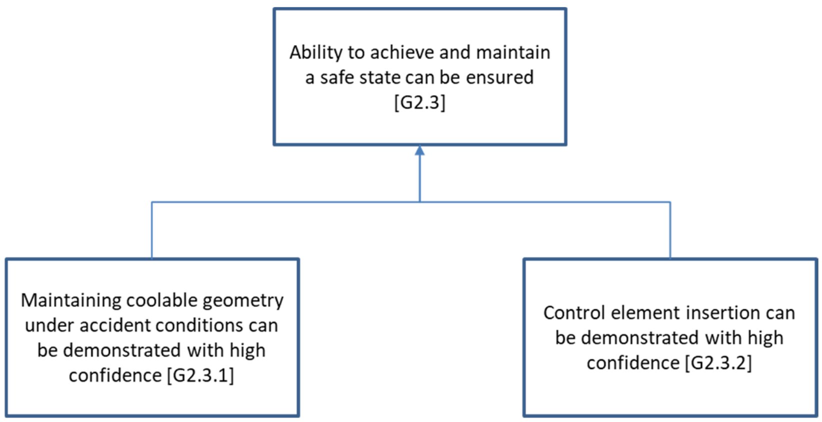

Figure 3-8. Decomposition of G2.3 – Safe state

3.2.3.1 G2.3.1 Maintaining Coolable Geometry

Maintaining coolable geometry is identified as a supporting goal to achieving and maintain a safe state. Maintaining coolable geometry is further decomposed into supporting goals in Figure 3-9. These supporting goals are discussed in the subsections below.

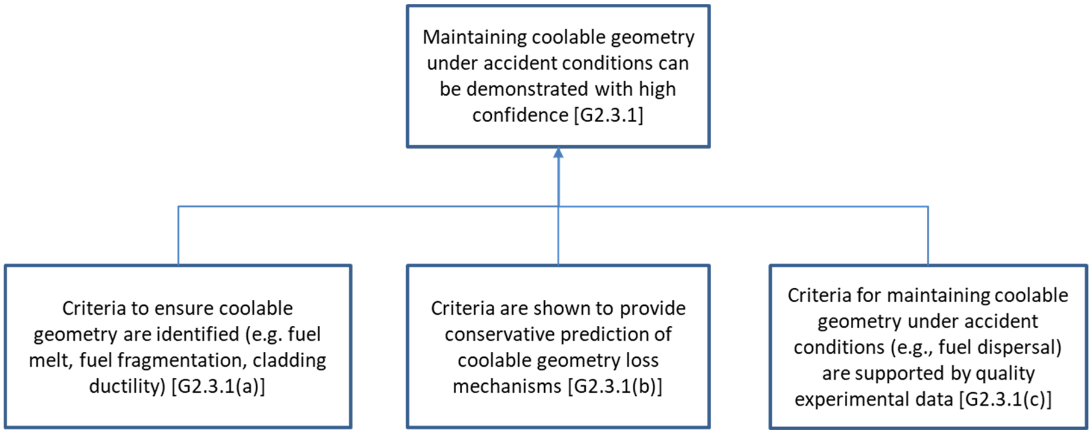

Figure 3-9. Decomposition of G2.3.1 – Coolable geometry

3.2.3.1.1 G2.3.1(a) Identification of Criteria

Criteria should be specified as far as reasonably practicable that ensure a coolable geometry is maintained. Coolable geometry criteria have historically been selected to ensure that core geometry is not significantly altered as a result of a design basis accident. Examples of coolable geometry criteria include (1) preventing centerline fuel melt and fuel fragmentation during transient overpower events, and (2) maintaining post-quench cladding ductility and long-term cladding phase stability during loss-of-coolant accidents. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.2.3.1.2 G2.3.1(b) Conservative Prediction

Criteria used to ensure coolable geometry should be conservative. Evidence needed to satisfy this goal is dependent on the associated phenomena. For example, a conservatively chosen criteria such as the onset of fuel melting should not require integral testing, but an empirically based criterion such as energy deposition for fuel dispersal or peak cladding temperature for cladding embrittlement is expected to demonstrate appropriate margin against experimental data. Historical examples of acceptable empirical criteria include the criteria developed for transient overpower (USNRC, 2015) and loss-of-coolant accidents (Hache and Chung, 2000). This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.2.3.1.3 G2.3.1(c) Experimental Data

This goal is satisfied through an evaluation against the assessment framework for experimental data provided in Section 3.4 of this report.

3.2.3.2 G2.3.2 Control Element Insertion

Control element insertion is identified as a supporting goal to achieving and maintaining a safe state. Control element insertion is further decomposed into supporting goals in Figure 3-10. These supporting goals are discussed in the subsections below.

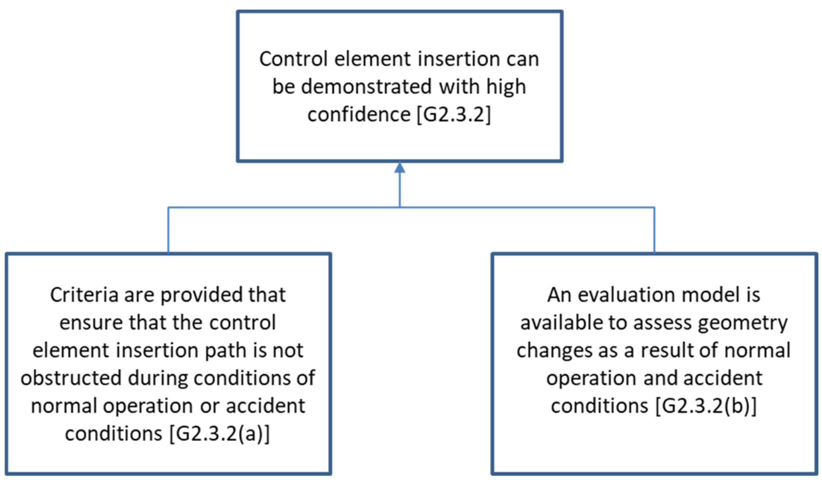

Figure 3-10. Decomposition of G2.3.2 – Control element insertion

3.2.3.2.1 G2.3.2(a) Identification of Criteria

Criteria should be specified to ensure as far as reasonably practical that the control element insertion path is not obstructed during the conditions of normal operation or accident conditions. These criteria should consider (1) loads from internal events and external events (e.g., seismic) that can distort the insertion path, (2) the potential to build up deposits (e.g., oxides) that can potentially obstruct insertion, (3) fretting induced wear that can potentially obstruct insertion, and (4) migrating bodies or foreign material that can obstruct insertion. Examples of these criteria for traditional LWR are the stress limit imposed on the control rod guide tubes sufficient to inhibit distortion of the insertion path, and control

rod repositioning occurring at sufficient frequency to avoid excessive fretting wear. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.2.3.2.2 G2.3.2(b) Evaluation Model

This goal is satisfied through an evaluation against the assessment framework for evaluation models provided in Section 3.3 of this report.

3.3 Assessment Framework for Evaluation Models (EM)

The term "evaluation model" is used in this report in a generic sense. Typically, an evaluation model is an analytical tool or computer code. However, use of a sophisticated tool, such as a computer code, may not be necessary. For example, a simple mathematical expression or set of data can be used as an evaluation model provided sufficient evidence exists to support its use. The assessment framework developed here is expected to be applicable generically. Initial design calculations should be revised if design parameters change (e.g., as-built flow rates result in higher than anticipated flow-induced vibration).

The assessment framework developed here supports G2.1.2 and G2.3.2(b), which are associated with evaluating design limits under conditions of normal operation, including the effects of AOOs, and control rod insertion criteria, respectively. It is noted that there is conceptual overlap between the assessment framework for evaluation models presented here and the goals established to determine criteria for barrier degradation, radionuclide retention and release, and coolable geometry which are presented in Sections 3.2.2.3, 3.2.2.4, and 3.2.3.1, respectively. The goals to support criteria for barrier degradation, radionuclide retention and release, and coolable geometry are distinct in that they are associated with accident conditions and have historically involved destructive testing using irradiated nuclear fuel under accident conditions. Accordingly, goals presented in Sections 3.2.2.3, 3.2.2,4, and 3.2.3.1 were developed separate from the evaluation model assessment framework described here.

The top-level goal of an acceptable evaluation model is supported by the goals of (1) having adequate modeling capabilities, and (2) assessment against experimental and plant data. This decomposition is shown in Figure 3-11 and discussed in the following subsections.

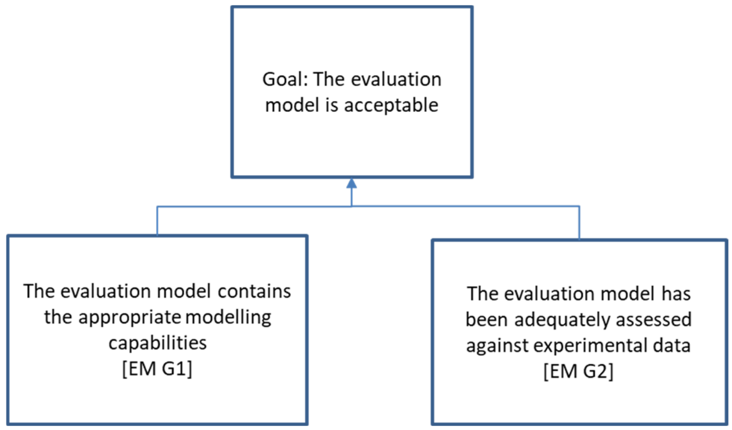

Figure 3-11. Decomposition of the main goal for evaluation model assessment

3.3.1 EM G1 Evaluation Model Capabilities

The evaluation model capabilities goal is decomposed into three supporting goals in Figure 3-12. In the USA this decomposition is informed by the Predictive Capability Maturity Model (PCMM) framework which identifies "Representation and Geometric Fidelity" and "Physics and Material Model Fidelity" as assessment elements (SAND, 2007). Other countries have their own guidance, but this document is reference for illustration purposes. Additional elements of PCMM framework are also considered in the evaluation model assessment framework. Specifically, "Model Validation" and "Uncertainty Quantification and Sensitivity Analysis" are addresses under the code assessment goal EM G2 and further discussed in Section 3.3.2. The remaining elements of the PCMM framework "Code Verification" and "Solution Verification" are expected to be addressed as part of a quality assurance program applicable to the design, analysis, and fabrication of a nuclear power facility. The goals supporting EM G1 are discussed in the following subsections.

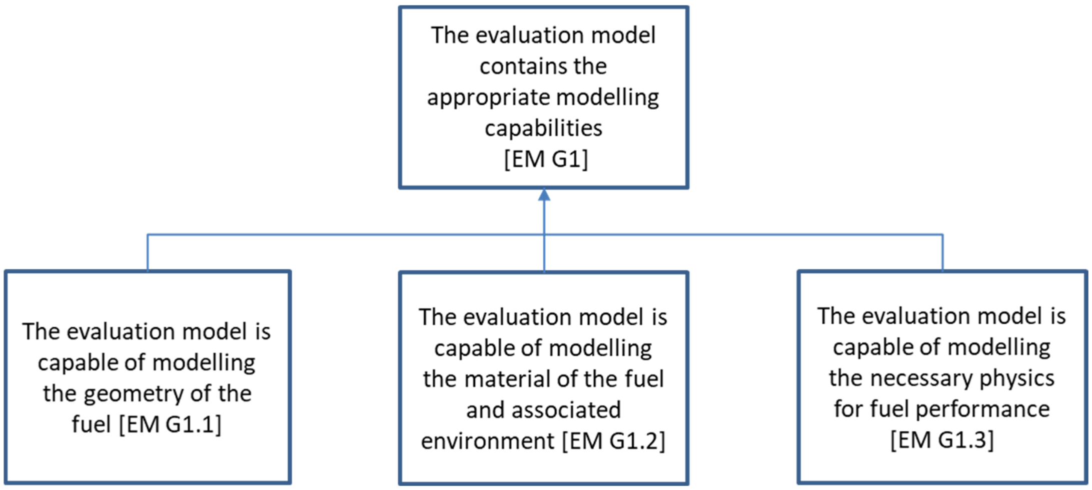

Figure 3-12. Decomposition of the EM G1 - Modeling Capabilities

3.3.1.1 EM G1.1 Geometry Modeling

The evaluation models should to be capable of modeling the geometry of the fuel system. Guidance of the levels of maturity to assess the geometry is provided in Table 3 of the PCMM, which includes the consideration of peer review (SAND, 2007). It is recognized that some fuel designs may require simplifying assumptions to address geometric modeling difficulties. For example, TRISO based particulate fuel involves coupled phenomena occurring at different geometric scales (e.g., micro-scale within the TRISO particle, meso-scale within the fuel compact, and macro-scale within the reactor core). Geometric modeling for such particulate fuel is expected to involve simplifications and assumptions that may not be required for a fuel design with less heterogeneity. Additionally, the evaluation model should have the ability to capture geometric changes associated with irradiation and exposure to the in-reactor environment (e.g. fuel swelling, cladding creep, oxide layer growth). Irrespective of imposed simplifications, appropriate justification should be provided for the geometric modeling scheme, and validation of the integrated evaluation model is accomplished through the assessment process under EM G2. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.3.1.2 EM G1.2 Material Modeling

The evaluation model should be capable of modeling material properties of the fuel system and its surrounding environment. This also includes changes in material properties due to irradiation and exposure to the in-reactor environment (e.g., thermal-conductivity degradation in nuclear fuel, changes to melting temperature, eutectic formation, changes to Young’s modulus). Guidance of the levels of maturity to assess the material modeling is provided in Table 3 of the PCMM from the USA, which

includes considerations for model calibration against test data and peer review (SAND, 2007). Justification should be provided for the material modeling scheme, and validation of the integrated evaluation model is accomplished through the assessment process under EM G2. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.3.1.3 EM G1.3 Physics Modeling

The evaluation model should be capable of modeling the physical and chemical processes that impact fuel performance. This goal requires knowledge of failure mechanisms, including changes due to irradiation and exposure to the in-reactor environment for the specified fuel. The evaluation model is expected to have sufficient physics models to address known failure mechanisms type (e.g., cladding oxidation and hydrogen pickup, fuel rod internal pressure, cladding strain). Guidance on the levels of maturity to assess the physics modeling is provided in Table 3 of the PCMM from the USA, which includes considerations for model calibration against test data and peer review (SAND, 2007). Justification should be provided for the physics models incorporated into the evaluation model, and validation of the integrated evaluation model is accomplished through the assessment process under EM G2. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.3.2 EM G2 Evaluation Model Assessment

Evaluation model assessment is an essential process that provides the confidence in the application of the evaluation model. To ensure that the evaluation model prediction is suitably conservative, any bias or uncertainty in the evaluation model prediction should be adequately quantified such that design and safety analyses can account for this uncertainty and bias. The assessment process in general relies on comparing evaluation model predictions against experimental data. This process is illustrated in Figure 3-13 which decomposes evaluation model assessment into two supporting goals, which are discussed in the following subsections.

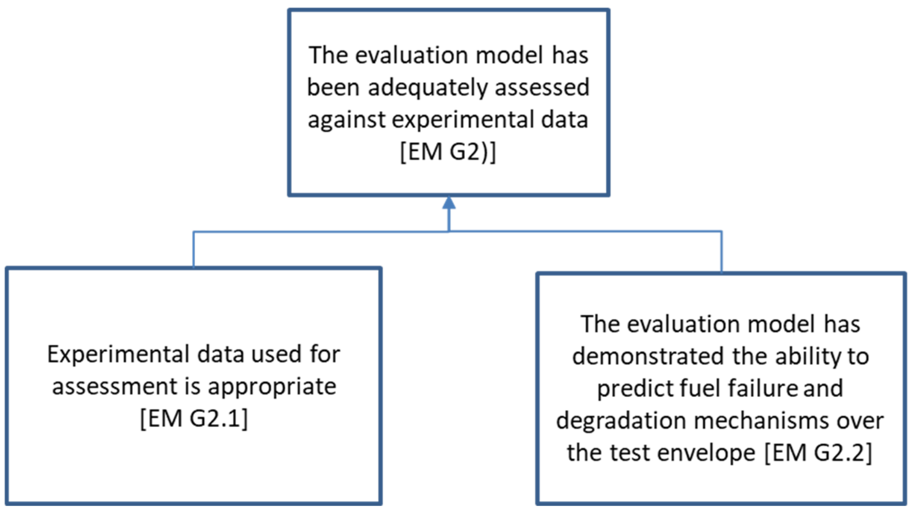

Figure 3-13. Decomposition of the EM G2 – Assessment against data

3.3.2.1 EM G2.1 Experimental Data

This goal is satisfied is satisfied through an as evaluation against the assessment framework for experimental data provided in Section 3.4 of this report.

3.3.2.2 EM G2.2 Demonstrated Prediction Ability over Test Envelope

Satisfying EM G2.2 involves comparing evaluation model predictions against experimental data. This comparison should establish evaluation model uncertainties and biases and identify limitations in the

evaluation model applicability. EM G2.2 is satisfied by addressing the four supporting goals shown in Figure 3-14, and discussed in the subsections below.

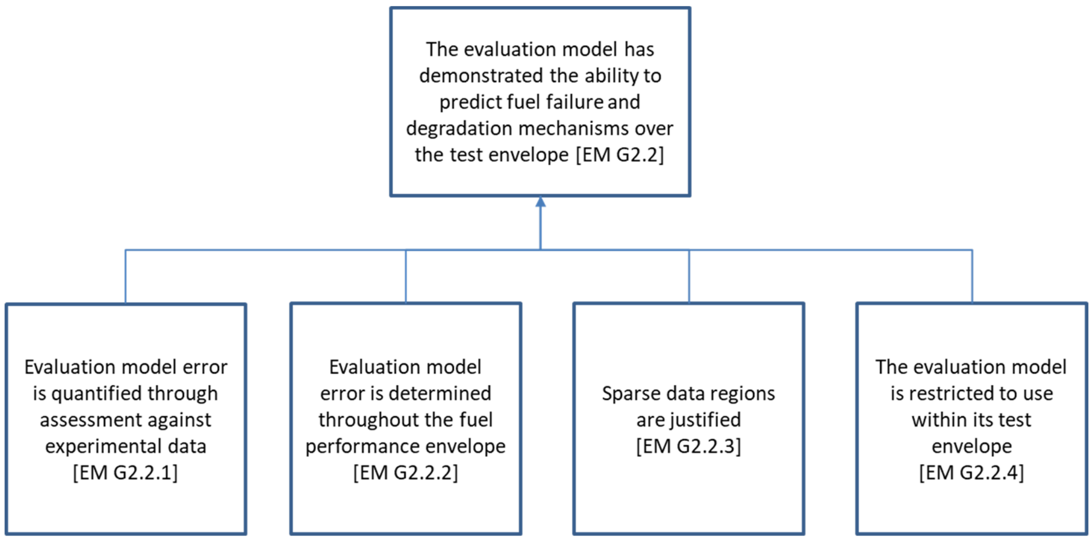

Figure 3-14. Decomposition of the EM G2.2 – Demonstrated ability

3.3.2.2.1 EM G2.2.1 Quantification of Error

Evaluation model uncertainties and biases for figures of merit need to be sufficiently understood in order to establish confidence in the evaluation model. It is expected that evaluation model predictions for assessment cases are compared against assessment data and the differences in measured-to-predicted values quantified in order to determine prediction biases and uncertainties. If sufficient data exists, then statistical confidence levels could be placed on the uncertainties of the evaluation model predictions. However, a more bounding or conservative approach can be taken (e.g., applying a bias or penalty to the model predictions, showing that evaluation model is inherently conservative). This goal is satisfied by a statement on the evaluation model biases and uncertainties along with justification through a quantification of predicted-to-measured values for assessment cases This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.3.2.2.2 EM G2.2.2 Span of Validation Data

Assessment data should to be distributed throughout the fuel performance envelope. The performance envelope, discussed in Sections 3.2.1.1 and 3.2.2.1, should be used to specify the test envelope. Accordingly, assessment data should be available to assess the evaluation model over the entire span of the performance envelope. However, it is recognized that data may not be available in all regions of

the fuel performance window. In such cases, it may be sufficient to provide justification that data in a specific region of the performance envelope is not required (e.g., limiting phenomena are known to not be present below a specified burnup). This goal is satisfied by demonstrating that assessment data is available over the entire performance envelope, and any gaps in assessment data are sufficiently justified. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.3.2.2.3 EM G2.2.3 Sparse Data Regions

Assessment data should be appropriately distributed throughout the fuel performance envelope. As discussed in Section 3.3.2.2.2, it may be acceptable to have regions in the performance envelope where the evaluation model is not directly supported by assessment data from integral experiments. However, in regions where assessment data is needed to validate the evaluation model, a sufficient number of data points should be available to assess the evaluation model. It is reasonable to expect data density to be greater near conditions of normal operation as fuel designers may require additional data in order to satisfy fuel reliability targets. However, any sparse data regions (i.e., regions of low data density) in the fuel performance envelope need to be adequately justified.

The expectation is that where data is sparse, the designer should exercise the precautionary principle. This can involve bounding physical arguments on the scope for error in physical processes or could be based on statistical treatment of uncertainty; either in determining the confidence level to assign to a hypothesis tested by a set of measurements, or on the confidence interval to apply to derived limits. It is good practice to add an appropriate contingency in the case of novel design limits.

This goal is satisfied by justifying the data density throughout the fuel performance window. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.3.2.2.4 EM G2.2.4 Restriction to Test Envelope

The evaluation model should only be applied within the region where it has been adequately assessed. Accordingly, some type of control should be in place in order to ensure that the evaluation model is not used outside its assessment base (e.g., code error or warning messages, administrative controls). This goal is satisfied by specifying the method used to ensure that application of the evaluation model is restricted to within its assessment base (i.e., the test envelope). This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.4 Assessment Framework for Experimental Data (ED)

An assessment of experimental data is the biggest area of review for fuel qualification. The assessment framework developed here supports all goals requiring evaluations against assessment data. Due to the several types of experiments that are expected as part of a fuel qualification program (e.g., steady-state irradiation of integral test specimens, transient ramp testing, design basis accident testing), the levels of evidence expected to support a goal associated with experimental data can vary. Such variance in the levels of evidence is discussed as applicable in the development of this assessment framework. The top goal for assessment data is decomposed in Figure 3-15 and discussed in the following subsections.

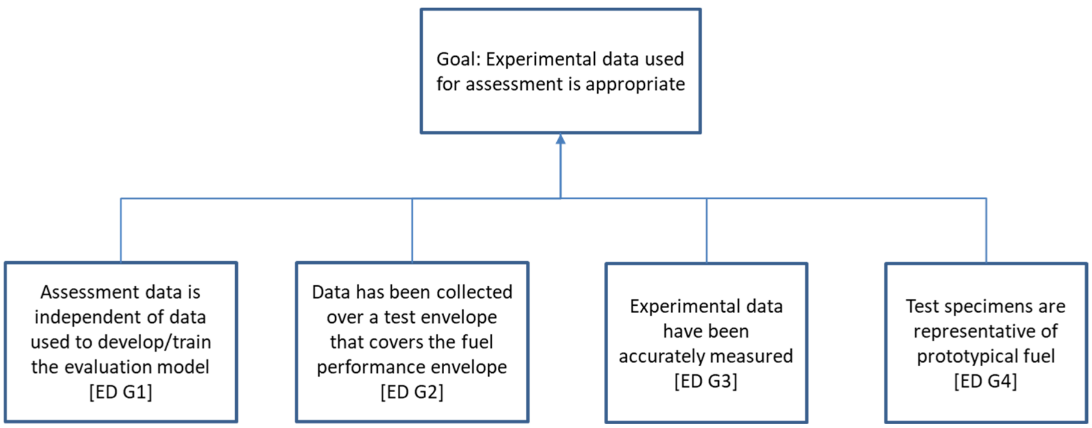

Figure 3-15. Decomposition of the main goal for data assessment

3.4.1 ED G1 Independence of Validation Data

Assessment data are the experimentally measured values that are used to quantify the evaluation model's error. Ideally, assessment data should be independent from any data used in the development (i.e, training) of the evaluation model. Although it may seem that use of the training data would be appropriate, the evaluation model has already been "tuned" to that data. Thus, quantifying the error of the training data would provide an estimate of "how well the model can predict data that were used in the generation of the model." This is different from "how well the model can predict data that were not used to generate the model." Because substantially more data points appear in the application domain (an infinite number) than were used to generate the model and because these points are the ones of most interest in future uses of the model, the focus should be on generating an estimate of the error over those points which were not used to generate the model. Thus, experimental data that have not been used to train the model should be held in reserve and used only to validate the model because the model's behavior using these data are indicative of the type of predictions that will be made in its future uses.

In some instances, the validation data and the training data are one and the same. There are methods in machine learning that can be applied to determine whether the selection of the training data affects the

resulting uncertainty, such as random subsamples and k-folds. In each of these methods, the data are randomly separated into subsets of training and validation data. The training data are used to develop the coefficients of the model, and the validation data are used to determine the overall uncertainty of the model. Then, the process is repeated with a different randomly selected data set assigned to training and the remaining data assigned to validation. Processes like these can provide reasonable estimates of the impact of using the same training data as validation data.

This goal is satisfied by demonstrating that the data used in the evaluation model assessment has sufficient independence. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.4.2 ED G2 Test Envelope

Data should be collected over a test envelope that spans the performance envelope. The performance envelope should address conditions of normal operation, AOOs, transient, and accident conditions. The types of tests that should be considered in the development of the test envelope include (1) steady-state integrated testing of the fuel system in a prototypical environment, (2) high power and undercooling tests to address: AOOs, off-normal conditions and to assess design margin, (3) power ramp testing to assess fuel performance during anticipated power changes, and (4) design basis accident tests to establish margin to fuel breach and contribution to source term under accident conditions. Design basis accident scenarios of typical interest include overpower events (e.g., reactivity insertion accidents) and undercooling events (e.g., loss-of-coolant accidents).

Many of the tests necessary to qualify a fuel for use require the use of irradiated test specimens. However, a situation is often encountered where test specimens are not available at the desired burnups. To address such situations is may be possible to propose the use of lead test specimens in order to extend the burnup limits of a fuel type. Under such lead test specimen programs, provisions should be specified to ensure safe operation of the fuel design during operation. Provisions such as ensuring that the lead test specimens are located in non-limiting regions of the reactor core and ensuring that sufficient monitoring is in place to detect potential failures should be considered if a lead test specimen program is being proposed.

This goal is satisfied by demonstrating that the test envelope addresses the necessary performance envelope for the fuel design. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.4.3 ED G3 Data Measurement

An understanding of data accuracy, associated with the data measurement and processing, is essential to establish overall confidence in the data used to develop and assess evaluation models. This goal is decomposed in Figure 3-16, and discussed in the subsections below.

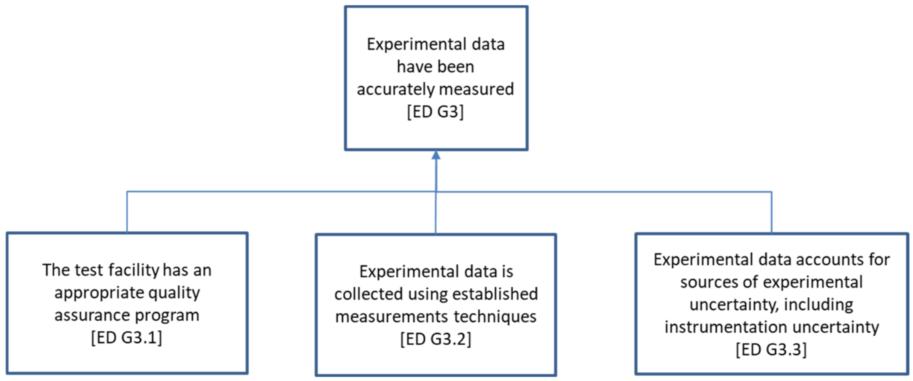

Figure 3-16. Decomposition of ED G3 – Data measurement

3.4.3.1 ED G3.1 Test Facility Quality Assurance

Experimental data should be collected under an appropriate quality assurance program. Standards are available to address quality assurance for test facilities such as the ASME NQA-1. Additionally, provisions may be applied to existing data in order to make it compliant with quality assurance requirements (ANL, 2019). This goal is satisfied by demonstrating that data collection was performed under an appropriate quality assurance program or providing an alternative justification for use of existing data. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.4.3.2 ED G3.2 Measurement Techniques

Data should be collected and processed using established or otherwise proven techniques. Experimental data may reflect direct measurements of physical parameters, but the more general case involves some degree of information processing. Use of novel and first-of-a-kind techniques should provide adequate justification for its use. This goal is satisfied by specifying the measurement and processing techniques and providing justification for the use of any novel or first-of-a-kind techniques. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.4.3.3 ED G3.3 Experimental Uncertainties

An error analysis of the experiment should be performed to assess sources of bias and uncertainty. Measurement uncertainty should be quantified when possible, and a discussion on the overall impact on assessment data should be provided. This goal is satisfied by providing an experimental error analysis. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.4.4 ED G4 Test Specimens

The test specimens used in the experiment should be representative of the prototypical fuel. This goal is decomposed in Figure 3-17 and discussed in the subsections that follow.

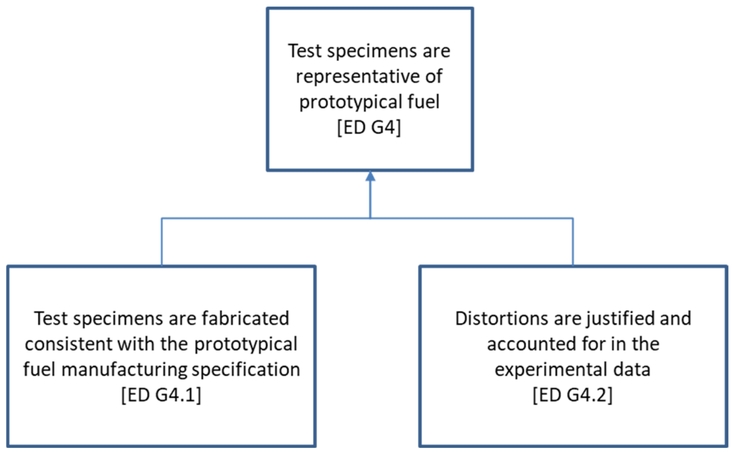

Figure 3-17. Decomposition of ED G4 – Test specimens

3.4.4.1 ED G4.1 Manufacturing of Test Specimens

Test specimens should be fabricated consistent with the manufacturing specification. This goal is associated closely with G1 Fuel Manufacturing Specification discussed in Section 3.1, which highlighted that fuel performance during normal operation and accident conditions can be highly sensitive to the fuel fabrication process. Alternatively, it may be possible to provide justification that differences in fuel fabrication between the prototype and test specimens are acceptable. Such justifications are expected to be addressed on a case-by-case basis. This goal is satisfied by demonstrating that test specimens are fabricated consistent with the fuel manufacturing specification. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

3.4.4.2 ED G4.2 Evaluation of Test Distortions

An evaluation of test distortions should be conducted. Test distortions are in reference to differences between test specimen and prototype. These differences may be associated with fabrication techniques, dimension, composition, and environment. An example of a test distortion that is expected is the geometry distortion typically associated with transient testing in a test reactor as test reactors are typically too small to accommodate full size prototypical fuel. This goal is satisfied by an analysis of the test distortions and justification for any identified distortions. This goal is recognized as an objective criterion that can be directly supported by evidence and is not decomposed any further.

4. Conclusions and Recommendations

A systematic evaluation of the requirements for qualifying nuclear fuel has been performed and a list of criteria has been identified to support a determination that nuclear fuel is qualified for use. The evaluation and justification are provided in Section 3 of this report. The tables below provide a concise list of all the criteria. The criteria highlighted in gray are identified as objective criteria for which direct evidence is needed to determine that the criteria are met. Higher level criteria in white are satisfied by satisfying all the lower level supporting criteria.

List of Goals in Fuel Qualification Assessment Framework

GOAL				Fuel is qualified for use
G1				Fuel is manufactured in accordance with a specification
	G1.1			Key dimensions and tolerance of fuel components are specified
	G1.2			Key constituents are specified with allowance for impurities
	G1.3			Microstructure attributes for materials within fuel component are specified for otherwise justified
G2				Margin to safety limits can be demonstrated with high confidence
	G2.1			Margin to design criteria under conditions of normal operation, including the effects of AOOs
		G2.1.1		Fuel performance envelope is defined
		G2.1.2		Evaluation model (go to EM Assessment Framework)
	G2.2			Margin to radionuclide release limits for accident conditions
		G2.1.1		Fuel performance envelope is defined
		G2.2.1		Radionuclide retention requirements are specified
		G2.2.2		Criteria for barrier degradation and failure
			(a)	Conservative criteria
			(b)	Experimental data is appropriate (go to ED Assessment Framework)
		G2.2.3		Radionuclide retention and release from fuel matrix
			(a)	Conservative model
			(b)	Experimental data is appropriate (go to ED Assessment Framework)
	G2.3			Ability to achieve a safe state can be ensured
		G2.3.1		Criteria specified for ensuring coolable geometry
			(a)	Criteria to ensure coolable geometry are specified
			(b)	Criteria are shown to provide conservative prediction of coolable geometry loss
			(c)	Criteria are supported by experimental data (go to ED Assessment Framework)
		G2.3.2		Control element insertion can be demonstrated with high confidence
			(a)	Criteria provided to ensure control element insertion path is not obstructed
			(b)	Evaluation model (go to EM Assessment Framework)

List of Goals in Evaluation Model Assessment Framework

GOAL			Evaluation model is acceptable for use
EM G1			Evaluation model contains the appropriate modeling capabilities
	EM G1.1		Geometry
	EM G1.2		Materials
	EM G1.3		Physics
EM G2			Evaluation model has been adequately assessment against experimental data
	EM G2.1		The data used for assessment is appropriate (go to ED Assessment Framework)
	EM G2.2		The evaluation model has demonstrated the ability to predict fuel failure and degradation mechanism over the test envelope
		EM G2.2.1	Evaluation model error is quantified through assessment against experimental data
		EM G2.2.2	Evaluation model error is determined through the fuel performance envelope
		EM G2.2.3	Sparse data regions are justified
		EM G2.2.4	Evaluation model is restricted to use within its test envelope

List of Goals in Experimental Data Assessment Framework

GOAL		Experimental data used for assessment is appropriate
ED G1		Assessment data is independent of data used to develop/train the evaluation model
ED G2		Data has been collected over a test envelope that covers the fuel performance envelope
ED G3		Experimental data have been accurately measured
	ED G3.1	The test facility has an appropriate quality assurance program
	ED G3.2	Experimental data is collected using established measurement techniques
	ED G3.3	Experimental data accounts for sources of experimental uncertainty
ED G4		Test specimens are representative of prototypical fuel
	ED G4.1	Test specimens are fabricated consistent with the prototypical fuel manufacturing specification
	ED G4.2	Distortions are justified and accounted for in the experimental data

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