Microreactor Transportability Challenges – 21072

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ABSTRACT
Microreactors are very small nuclear reactors with a power output of 20 MW of electrical power or less and are designed to be factory-built, modular in nature, and highly portable. Potential modes of transport for microreactors include highway, rail, barge/ship, and air.

In order to assess the transportability of current microreactor designs, previous U.S. Army and U.S. Air Force nuclear reactor transportability studies were examined. The flight testing of Aircraft Shield Test Reactor in a modified B-36H bomber provided useful insights into potential weight and shielding issues associated with the air transport of microreactors. Transportability studies conducted for the ML-1 nuclear power plant provided useful insights into shipping a microreactor by air, truck, and rail, including weights, dimensions, tie-downs, loading procedures, and the shock and vibration associated with the railroad environment.

Detailed schedules of U.S. Department of Transportation (DOT) and U.S. Nuclear Regulatory Commission (NRC) regulations that are relevant to the shipment of microreactors were also developed. The schedules are patterned after similar schedules contained in International Atomic Energy Agency (IAEA) Specific Safety Guide No. 33, Schedules of Provisions of the IAEA Regulations for the Safe Transport of Radioactive Material. These schedules concentrated on the transport of fissile material packages and Type B packages, which would be most applicable to the transport of the microreactor and its fuel before and after irradiation, respectively. Schedules were also developed for the air shipment of fissile material and Type B packages by military aircraft and for air shipment of Type C packages based on IAEA regulations.

Based on the previous transportability studies and the DOT, NRC, U.S. Air Force, and IAEA regulatory schedules, potential issues associated with the transport of microreactors were identified. Potential areas of issue include the microreactor fuel type, fissile material and Type B packaging requirements, the microreactor shielding and weights, specific transportation mode (highway, rail, barge/ship, and air), and technical packaging evaluations (structural, thermal, containment, shielding, and criticality). Issues associated with defense-in-depth were also evaluated.

One option for addressing most of the issues associated with transporting a microreactor is to ship the microreactor without unirradiated or irradiated fuel thus avoiding the need to develop new transportation packages designed for the specific fuel type of the microreactor.

For transport by air, shipping the unirradiated or irradiated fuel separately from the microreactor might simplify the design and certification of the fissile material package for the fuel because the design would not have to consider the design of the microreactor, but the fissile material packages would still have to meet the enhanced testing standards for air transport of fissile material identified in the regulations.

A second option for transporting a microreactor is to ship the microreactor together with its contents, i.e., unirradiated or irradiated fuel. This option is likely to be more challenging than shipping the microreactor and its fuel separately because the microreactor with its contents would have to be certified as a fissile material package or as a Type B package.

As an alternative to demonstrating compliance with the deterministic requirements contained in DOT and NRC regulations, probabilistic risk assessment (PRA) is one method that could be used to demonstrate the equivalent safety of transporting a microreactor together with its contents. PRA techniques have been applied to the transportation of spent nuclear fuel, most notably in NUREG/CR-4829, NUREG/CR-6672, and NUREG-2125.

A PRA for transporting radioactive materials would be based on accident event trees that represent a set of possible transportation accidents. Each potential mode of transport (highway, rail, barge/ship, and air) would require their own event trees.

In a transportation PRA, knowledge of potential radionuclide source terms would be required. This would include radionuclide inventories and the response of the fuel and microreactor to various classes of accidents, including collision-only accidents, fire-only accidents, and accidents that involve a collision and a fire. Determining the response of fuel and microreactor to these accidents would involve detailed structural, thermal, shielding, and criticality analyses.

In order to establish safety equivalency using a transportation PRA, safety goals would need to be established. A safety goal could be based on the quantitative safety goals for nuclear power plants. This approach would require detailed consequence assessments. Alternatively, a safety goal could be based on demonstrating that the overall level of safety in transport for the shipment is at least equivalent to that which would be provided if all the applicable requirements had been met. In the context of a transportation PRA, the resulting safety goal could be “the probability of an accident that could result in a release greater than allowed by transportation regulations is very small.” A quantitative assessment of what constitutes “very small” in the context of a transportation PRA has not been conducted, but previous transportation risk assessments (e.g., NUREG-2125 or NUREG/CR-6672) could provide insights.

INTRODUCTION

Microreactors are typically defined as reactors that generate up to 20 MW of electrical power. These reactors are designed to be factory-built, modular in nature, and ideal for providing a test bed for component and subsystem development and demonstration. They are also anticipated to lend themselves well to mobile or transportable applications.

Current microreactor designs are typically fueled with high-assay low-enriched uranium (HALEU), which has enrichments that range from 5% to 20%. One of the most common fuels being discussed for microreactors is tristructural isotropic (TRISO) fuel. For example, the microreactors being developed for the U.S. Department of Defense Strategic Capabilities Office Project Pele are fueled with TRISO. Figure 1 shows TRISO fuel particles, fuel compacts, prismatic graphite blocks, and spherical fuel pebbles.

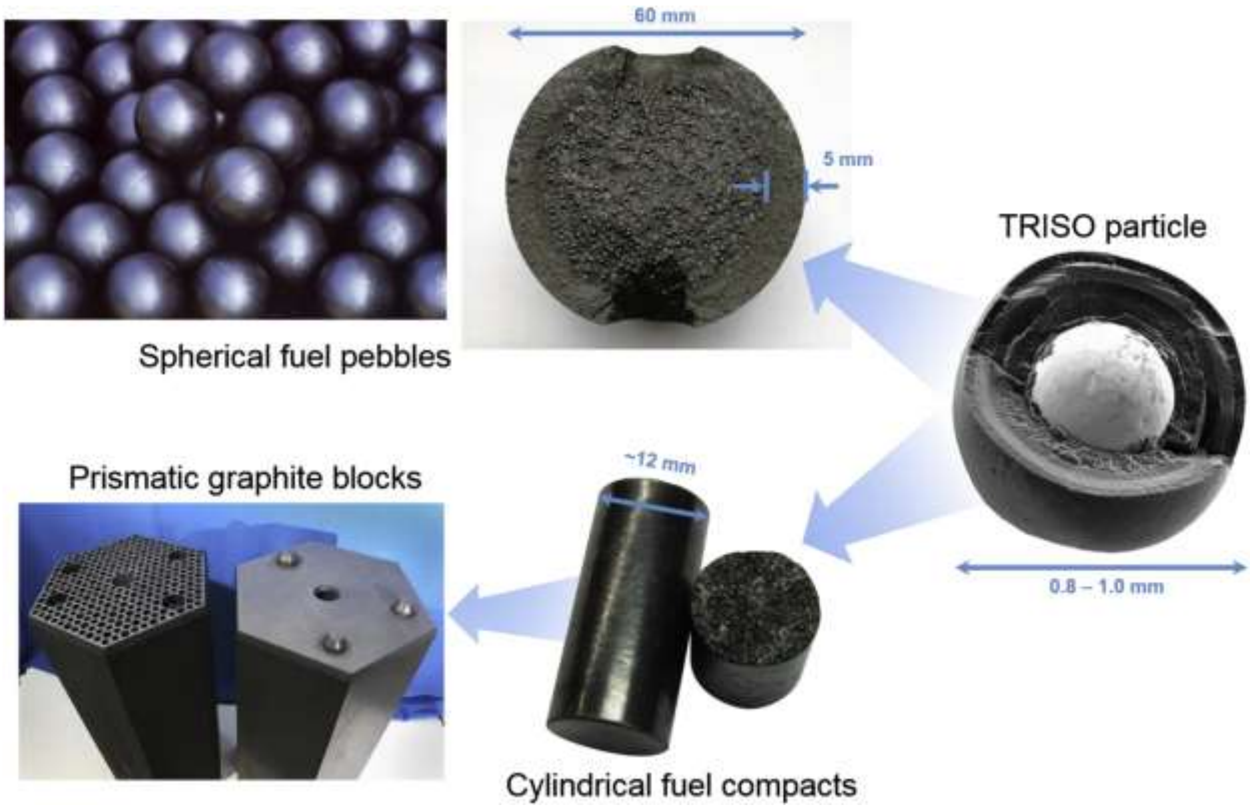

Fig. 1 TRISO Particles, Fuel Compacts, Prismatic Graphite Blocks, and Spherical Fuel Pebbles [1]

PREVIOUS ARMY AND AIR FORCE MICROREACTORS

The U.S. Army and the U.S. Air Force designed and deployed small nuclear reactors during the 1950s through the 1970s. Examples of small nuclear reactors deployed by the U.S. Army are listed in Table I. In addition to the reactors listed in Table I, the U.S. Air Force developed and tested the Aircraft Shield Test Reactor (ASTR).

Table I. Small Nuclear Reactors Deployed by the U.S. Army

Reactor	Power (MW thermal)	Period of Operations
Gas-Cooled Reactor Experiment (GCRE)	2.2	1959-1962
Mobile Low Power Plant (ML-1)	3.3	1961-1965
Mobile High Power Plant (MH-1A)	45	1967-1977
Portable Medium Power Plant (PM-1)	9.37	1962-1968
Portable Medium Power Plant (PM-2A)	10	1960-1964
Portable Medium Power Plant (PM-3A)	9.51	1962-1972
Stationary Medium Power Plant (SM-1)	10	1957-1975
Stationary Medium Power Plant (SM-1A)	20.2	1962-1972
Stationary Low Power Plant (SL-1)	2.2	1958-1961

Aircraft Shield Test Reactor

The Aircraft Nuclear Propulsion (ANP) Program was a joint program between the U.S. Air Force and the U.S. Atomic Energy Commission (AEC). The ANP Program had its origins in the Nuclear Energy for Propulsion of Aircraft (NEPA) project which began in 1946. The ANP Program was established in 1951 and was terminated in 1961.

As part of the ANP Program, the ASTR was developed to serve as a radiation source for supplying shielding information not obtainable at ground facilities. This 1 MW thermal reactor consisted of 34 plate-type fuel elements containing 4.8 kg of highly enriched uranium and used demineralized water as the moderator, reflector, and coolant. The ASTR and its shielding weighed 16 metric tons. The ASTR was flight tested in a modified Convair B-36H that was redesignated the XB-36H and then the NB-36H and called the Nuclear Test Aircraft or NTA. The NTA also contained a shielded crew compartment.

Criticality during flight first occurred on September 5, 1955. The NTA containing the ASTR made 47 test flights and 215 hours of flight time (during 89 hours of which the reactor was operated) between July 1955 and March 1957. The ASTR was later upgraded to 3 MW thermal and 10 MW thermal and used exclusively for ground testing.

ML-1 Nuclear Power Plant

The Army Nuclear Power Program (ANPP) was a joint program between the U.S. Department of Defense and the AEC. The ANPP was initiated in 1954; during its lifetime, it designed, constructed, operated, and deactivated nine nuclear power plants. By 1977, due to changing military requirements and funding limitations, major program activities had ceased when the last ANPP facility was deactivated.

The ML-1 nuclear power plant was a 3.3 MW thermal gas-cooled reactor that consisted of 61 pin-type fuel elements containing 49 kg of highly enriched uranium and used water as a moderator and nitrogen as a coolant. The ML-1 operated at the National Reactor Testing Station (later Idaho National Laboratory) from 1961-1965. In addition to operating the ML-1, tests were performed to determine the transportability of the ML-1 nuclear power plant [2].

The ML-1 plant consisted of the nuclear reactor package, the power conversion package, the control cab, and the auxiliary package. The nuclear reactor package was 2.8 x 2.7 x 2.4 meters and weighed 14 metric tons. The power conversion package was 4.3 x 2.9 x 2.4 meters and also weighed 14 metric tons.

To assess transportability, full-scale mockups of the reactor and power conversion packages were fabricated that duplicated the overall dimensions, the weight, and center-of-gravity of the ML-1 packages. Mockups were not constructed of the control cab or of the auxiliary package, because the weights and sizes of these units permitted handling by conventional techniques. The mockups duplicated the skid-mounting concept and the shock mounts, consisting of silicone rubber cores bonded to aluminum supports, planned for the ML-1. The shock mounts, in conjunction with nylon rope tie-downs, were designed to attenuate shock loads encountered during transport and to reduce these loads to values less than the maximum allowable. Figure 2 shows the full-scale mockup of the truck-mounted ML-1 nuclear power plant.

Fig 2. Full-Scale Mockup of the ML-1 Nuclear Power Plant

The primary means of transporting the ML-1 was to be the U.S. Army M-172, or M-172-A-1, low-bed semitrailer. It was further specified that the control cab be transported by an M-35 2.3-metric ton cargo truck; and that auxiliary equipment be transported either on the M-35 or on M-55 4.5-metric ton trucks. Because mobility and rapid deployment were among the prime requisites of the ML-1, it was also capable of being transported by aircraft. U.S. Air Force C-124, C-130, and C-133 aircraft were the specified carriers. In the case of the C-124 and the C-130, three separate air lifts were required to transport the ML-1. (One aircraft carried the reactor; a second, the power conversion equipment; and a third, the seven-man crew, the control cab, and the power plant auxiliaries.) The C-133 had the capacity to transport the entire power plant, including its crew, control cab, and auxiliaries; in one flight. A third transport method specified for the ML-1 was by standard railroad flatcar. When positioned on a flatcar, the ML-1 was designed to meet the clearance requirements of United States and European main lines.

Trailer transport tests were performed with the mockup packages near the Aerojet General Corporation plant at San Ramon, California, in September 1959. The U.S. Army supplied the M-172 trailer, M-52 tractor, and a 3-person crew for the tests.

The purposes of the tests were to determine that: 1) the ML-1 packages could be loaded by standard techniques, and 2) the experimental shock mounts were satisfactory both during loading and while in transit.

The tests, augmented by time and motion studies, also provided data for the development of loading procedures.

The tests showed that the ML-1 power plant could be loaded in the field by conventional methods and transported by the M-172 trailer. The ML-1 could also be satisfactorily transported cross country by the M-172 trailer and M-52 tractor. When the reactor and power conversion packages were coupled and loaded on the M-172 tractor, the resulting tandem axle load exceeded the maximum permitted on most U.S. highways. The tests also showed that overseas military height restrictions could be met with the ML-1 loaded on the M-172 trailer. The height restriction was exceeded with the load on an M-172-A-1 trailer, but the rear wheels could be partially deflated, temporarily, to permit clearing overhead obstructions.

Aircraft loading tests using the C-124, C-130, and C-133 aircraft were conducted during the period September 15-23, 1959. The results of the tests showed that the ML-1 package was able to be loaded aboard the C-124, C-130, and C-133 aircraft with equipment normally available in the field. In the case of the C-133, side clearances were less than the minimum specified for personnel passage. Also, the tie-down system was adequate to protect the ML-1 during aircraft emergency landings.

The railroad tests were conducted on the San Ramon spur line of the Southern Pacific Railroad Company, during the last two weeks of October 1959. The tests were performed for two purposes: 1) to determine the effectiveness of the tie-down system for shock reduction under severe railroad switching and humping operations; and 2) to simulate an aircraft emergency landing and establish the effectiveness of the tie-downs during such a landing.

The railroad tests showed that when loaded and tied down on railroad flatcars, the ML-1 met all clearance limitations specified for both United States and European main lines. The tie-downs and shock mounts were found to isolate all shocks sufficiently to eliminate any danger of damage to the ML-1 during railroad switching and humping operations.

MICROREACTOR TRANSPORT OPTIONS

There are two options for transporting a microreactor, separately from its unirradiated or irradiated fuel or together with its unirradiated or irradiated fuel. Shipping unirradiated or irradiated fuel separately from the microreactor would require a fissile material package for the unirradiated fuel and a Type B package for the irradiated fuel. Depending on the source of the HALEU fuel, shipping unirradiated fuel might also require a Type B package. The requirements for fissile material and Type B packages are contained in 10 CFR Part 71 [3].

For shipping a microreactor together with its unirradiated or irradiated fuel contents, the same fissile material and Type B packaging requirements would apply. However, meeting these requirements will be more challenging because the microreactor must serve two purposes, i.e., it must function as an operating reactor and also as a transportation package. If the microreactor cannot meet the fissile material or Type B packaging requirements, several options are available:

- The microreactor transportation package could be approved using alternate environmental and test conditions [see 10 CFR 71.41(c)]. In addition, compensatory measures would be necessary to ensure an equivalent level of safety. This regulatory option has been used in the past. For example, Revision 19 of the certificate of compliance for the 10-160B radioactive waste transportation cask (Docket No. 71-9204) was approved based on alternate environmental and test conditions and compensatory measures. For this option, a certificate of compliance would be issued by the U.S. Nuclear Regulatory Commission (NRC).
- The microreactor transportation package could be approved using a special package authorization [see 10 CFR 71.41(d)]. Special package authorizations have been used for the transport of large components such as contaminated steam generators from nuclear power plant sites but only in

limited circumstances and only for one-time shipments. For this option, a certificate of compliance would not be issued by the NRC.

- The microreactor transportation package could be approved using an exemption (see 10 CFR 71.12). This option would require NRC to prepare an environmental assessment for the transport of the microreactor and would require the shipper to obtain a special permit from the U.S. Department of Transportation (DOT). This regulatory option has been used in the past, such as for the transport of the Trojan Nuclear Power Plant reactor pressure vessel package for disposal. A certificate of compliance would not be issued for the microreactor transportation package and the exemption would not be transferrable to another entity.

TRANSPORTATION REGULATIONS RELEVANT TO MICROREACTOR TRANSPORT OPTIONS

Maheras and Adkins [4] presents the transportation regulations that are relevant to microreactor transport options in the form of schedules. These schedules contain specific regulatory requirements for the transport of fissile material packages and Type B packages. Schedules for common provisions of transportation regulations, radioactive material package design and testing, military air shipment of fissile material and Type B packages, and Type C fissile material packages are also presented in Maheras and Adkins [4].

The schedules are patterned after similar schedules presented in Cook et al. [5] and IAEA [6] and presented in a two-column format. The right column is the regulatory requirement and the left column is the citation of the regulation in which the requirement is found. In some cases, the regulatory requirement may be paraphrased from the actual regulations for simplicity and conciseness. In general, the schedules are organized into sections on materials, packaging/package, radiation, contamination, decontamination, mixed content, loading and segregation, marking and labeling, placarding, transport documents, storage and dispatch, carriage, and other provisions.

ISSUES ASSOCIATED WITH MICROREACTOR TRANSPORT

Microreactor Fuel Type

As mentioned previously, one of the fuel types being discussed for microreactors is TRISO fuel with enrichments approaching 20%. There have been transportation packages certified by the NRC for both unirradiated and irradiated TRISO fuel. Examples include the TN-FSV (Docket No. 71-9523) and the FSV-3 (Docket No. 71-6347). In addition, the Versa-Pac (Docket No. 71-9342) is certified for the transport of TRISO fuel compacts. Based on this experience, shipping TRISO fuel separately from its microreactor appears feasible. However, shipping a microreactor with its TRISO fuel would be much more challenging because the microreactor and its unirradiated or irradiated contents would need to meet the requirements contained in 10 CFR Part 71. If these requirements cannot be met, it may be possible for the microreactor transportation package to be approved using alternate environmental and test conditions and associated compensatory measures [see 10 CFR 71.41(c)], a special package authorization [see 10 CFR 71.41(d)], or an exemption (10 CFR 71.12).

Microreactor Shielding and Weights

The principal sources of radiation during the operation of a microreactor are:

- Fast neutrons arising from fission in the core.
- Thermal neutrons from fast flux removal in the moderator and neutron shield.
- Gamma radiation resulting from prompt fission in the core.
- Gamma radiation from fission products developed during operation.
- Gamma radiation from neutron capture in the core, shield, and structural members.

During the transport of a microreactor, the gamma radiation from fission products and neutron capture are likely to be the principal sources of radiation.

In past microreactor designs such as the ML-1 reactor or the ASTR, several types of shielding have been discussed:

- Shutdown shielding. The shielding provided to reduce the radiation levels from the reactor following shutdown from extended operation at full power to the values contained in the specifications for the reactor.
- Operational shielding. The shielding provided in addition to the shutdown shield to minimize, as possible within the weight and size limits contained in the reactor specifications, the radiation levels from the reactor during operation.
- Preferential shielding. The practice of providing more shielding in one direction when compared to another direction.
- Divided shielding. Most often used for an airborne reactor. As discussed in Westfall [7], there are essentially two types of shields that may be used in an aircraft, the unit shield and the divided shield. The unit shield concentrates all the attenuating material at the reactor. At the shield's outer surface, the radiation has been reduced to established permissible levels. This type of shield provides the crew freedom of movement, reduces radiation damage to materials to a minimum, and considerably eases aircraft maintenance. The price for it is an extremely large, concentrated weight and a huge, ungainly size. The divided concept places a portion of the shield at the reactor, and the remainder at the crew compartment. This involves the definition of two permissible dose rates, one for the crew and the other for equipment between the crew compartment and reactor shields. The saving in weight and size effected by the more efficient use of materials in this type of shield is most attractive. In comparison, attendant disadvantages, such as cramped crew quarters, high radiation damage to materials outside the reactor shield and crew shield, and difficult remote-handling maintenance, assume less importance.
- Expedient shielding. Expedient shielding consists of shielding materials commonly available in the field such as water, dry sand, wet sand, limestone, soil, and wood.

It is important to recognize that shielding will be a major contributor to the size and weight of a microreactor [8]. In addition, there is an inherent trade-off among the shielding associated with a microreactor, the weight of a microreactor, and radiation dose rates. For this reason, it is important to establish radiation dose rate and weight performance specifications that a microreactor must meet in order to be transportable. This may involve time constrained scenarios and time unconstrained scenarios.

Transportation Mode Issues

Highway. The typical tractor-trailer weighs about 16 metric tons. State permits are required if the gross vehicle weight exceeds 36 metric tons, which limits the weight of the microreactor transportation package to about 20 metric tons without obtaining a state permit. A state permit would also be required if the microreactor transportation package was overdimension.

It is likely that a microreactor with its irradiated fuel contents would be a highway route controlled quantity truck shipment and would require a hazardous materials safety permit [49 CFR 385.403(a)] [9], which would require a written route plan and a pre-trip Commercial Vehicle Safety Alliance (CVSA) Level VI inspection. This requirement is not unique to shipping a microreactor and its irradiated fuel and should not be an issue.

Likewise, the requirements in 49 CFR 397, Subpart D [10] for the routing of Class 7 (radioactive) materials, including the requirements for motor carriers and drivers (49 CFR 397.101) and requirements for state routing designations (49 CFR 397.103), are not unique to microreactors and should not be an issue.

Rail. It is likely that the railroad's expectation will be that shipments of irradiated fuel be conducted using railcars that comply with Association of American Railroads (AAR) Standard S-2043 [11]. AAR Standard S-2043 railcars have been developed and are currently under development; however, the minimum test load for these railcars is likely to greatly exceed the weight of a microreactor and its components. This issue would need to be discussed with the AAR Equipment Engineering Committee to determine if additional testing would be required to ship a microreactor and its components in compliance with AAR Standard S-2043 or if additional ballast weights could be added to the railcar to satisfy the requirements of AAR Standard S-2043.

Air. 10 CFR 71.55(f)(1)(iv) requires enhanced puncture, thermal, and drop tests, e.g., a fire of one hour duration, double the duration of that required for a fissile material package not shipped by air, and an impact onto an unyielding surface with a velocity of 90 m/s. In addition, 10 CFR 71.74 contains additional tests for the air transport of plutonium, including an impact onto an essentially unyielding surface with a velocity of 129 m/s and a terminal free-fall velocity test. Because of these enhanced tests, if a microreactor containing its unirradiated or irradiated fuel were to be shipped by air, it could be a significant challenge to meet the regulatory requirements in 10 CFR 71.55(f) or 10 CFR 71.74.

If a microreactor and its irradiated fuel were shipped by air internationally, it would likely need to meet IAEA requirements for a Type C package because the irradiated fuel would contain an activity greater than 3000 A_2.

Issues Associated with Technical Packaging Evaluations

Structural. For shipping a microreactor with its unirradiated or irradiated contents, microreactor designers should consider the need to demonstrate compliance with the requirements for the normal conditions of transport (NCT) and hypothetical accident conditions (HAC) in 10 CFR Part 71 in their microreactor designs.

Defining the microreactor with its unirradiated or irradiated contents as the package, approval standards in 10 CFR 71.41 state that, to demonstrate compliance, the effects of tests specified in 10 CFR 71.71 and 10 CFR 71.73 must be evaluated. Breaking the component safety groups into containment, criticality, and other safety components, they can be further defined as follows.

Containment components are defined as all the components required for retaining the radioactive contents. The function of all the containment vessel and closure components, which could include or be defined by the reactor vessel depending on the microreactor packaging definition, is to maintain the containment boundary so that all NCT and HAC containment requirements are met. Included in this component safety group are any closure lids, seals, port components, and bolts. Type B, Category I packaging design criteria dictates that the structural evaluations and stress limits need to comply with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Division 1, Subsection NB per NUREG/CR-3854 [12]. Stress limits differ between NCT and HAC such that meeting NCT structural performance requirements tends to be more challenging than to meet HAC structural performance requirements.

Analytical methods should be used to demonstrate that the containment vessel meets the vessel design criteria presented in Subsection NB, Article NB-3000 as amended by Regulatory Guide 7.6 [13]. Also, in the case of closure bolts, due to their importance, applicants/developers are directed to use acceptance criteria provided in NUREG/CR-6007 [14].

Limits for release of contents are specified in 10 CFR 71.51 for both NCT and HAC evaluations. Demonstration of compliance with the specified limits must be in accordance with the methods laid out in ANSI N14.5 [15]. Additionally, in accordance with 10 CFR 71.61, a Type B package containing more than 10^5 A_2 must be designed so that its undamaged containment system can withstand an external water pressure of 2 MPa for a period of not less than 1 hour without collapse, buckling, or in-leakage of water.

Criticality components are defined as all components required in controlling nuclear criticality during transport of fissile material in the package. Included in this component safety group are neutron absorbers and related structures required to retain the relative position of the fissile materials and/or neutron absorbers. Type B, Category I packaging design criteria dictates that the structural evaluations and stress limits need to comply with ASME BPVC, Section III, Division 1, Subsection NG per NUREG/CR-3854 [12].

Other safety components are defined as all other packaging safety-related components. This includes but is not limited to both gamma and neutron shielding components; secondary seals, bolts, and closures; impact limiters; and lifting lugs and tie-down devices. Type B, Category I packaging design criteria dictates that the structural evaluations and stress limits need to comply with ASME BPVC, Section VIII, Division 1, or Section III, Subsection NF per NUREG/CR-3854 [12].

ASME BPVC, Section III, Division 1, Subsection NB contains the requirements for materials, design, fabrication, examination, and testing of Class 1 reactor components to form acceptable design criteria for shipping containment vessels. A critical consideration to consider is that the choice of materials is limited in number and temperature use limits are 371 °C for carbon and low alloy (ferritic) steels and 427 °C for austenitic and high alloy steels. As such, if the reactor vessel is defined so as to provide reactor containment during operation as well as content containment during transport, a code case will likely need to be developed and leveraged to accommodate the higher temperatures customary or normally expected with the microreactor demonstration definitions.

Thermal. Similar to the structural discipline, the thermal requirements in 10 CFR Part 71 and 49 CFR Part 173 [16] must be met before a package can be certified for the transportation of radioactive materials. Packaging definitions used to transport radioactive material must be capable of withstanding intense thermal environments while preventing the release of contents and maintaining shielding and nuclear subcriticality. Achieving this capability requires the use of construction materials that enable the package to withstand serious thermal insult. Accordingly, all safety significant components of a packaging definition comprised of a microreactor with unirradiated or irradiated contents must be demonstrated to survive a broad temperature range and significant temperature differential exposures.

Additionally, a microreactor with irradiated contents or associated spent fuel, possibly anticipated to be shipped separately, must employ design strategies that account for contents that generate a relatively large amount of heat as the issue of thermal protection becomes much more complicated. In this case, insulating material must work effectively to reduce the heat added to the package during an upset or accident condition while also allowing internally generated heat to escape under regular operating conditions. These requirements are in conflict with one another and must be carefully balanced during the packaging design process. Also, when there is a significant heat source within the package, such as the microreactor with irradiated contents, an effective thermal insulation may cause the interior portions of the package to overheat under HAC or even possibly NCT conditions. This could then lead to failures of safety significant items. This must also be evaluated and demonstrated to not be the case.

Following NCT, a package may appear visually fully functional, but it must be shown that there is no substantial reduction in the effectiveness of the package. The failure of safety significant containment items as it relates to thermal insult is most likely during or after an accident scenario involving a fire.

Therefore, it is important that the packaging definition be designed such that all containment items withstand the highest expected temperature under HAC after the entire structural impact loading sequence. After the HAC structural and thermal loading sequences of the packaging definition, a maximum leak rate of an A_2 per week (or 10 A_2 of Kr-85 per week) is permissible under HAC as specified in 10 CFR 71.51(a)(2).

Loss of radiation shielding from the packaging definition, due to either melting of gamma shielding and burning or pyrolysis of the neutron shielding material, can result in severe circumstances. A typical design evaluation approach is to not take credit for any material that may be consumed or lost during a thermal (or any other type) accident. This may prove to be especially challenging when attempting to demonstrate that the public, worker, and environment are being protected.

Generally, criticality control is provided by geometric spacing and strategic poisoning. As such, these items are always considered to be safety significant and must be demonstrated to remain intact during NCT as well as HAC evaluation scenarios. Additionally, it is possible to create a criticality event by simply flooding the contents of a package. This must also be evaluated per 10 CFR Part 71 and the potential must be shown to not exist.

Containment. The design of a packaging definition typically starts with the containment system. The containment system is defined as an assemblage of all the components required to retain the radioactive contents. In general, this includes items such as the containment vessel, possible seals and port components, and closure bolts. The containment boundary is an assemblage of all the components required to retain the contents and is in direct communication with the internal cavity of the containment vessel. The structural design criteria for certain package component safety groups are based on the ASME BPVC.

As previously discussed, in the development of a compliant packaging design definition, the following four key features should be incorporated:

- Containment system
- Thermal management system or convention
- Shielding
- Criticality controls.

A containment system is always required to provide strict retention of the radionuclide inventory. An alternate containment boundary definition may very well need to be exercised or defined in the instance of a reactor containment vessel which then becomes the content containment during transport. If TRISO is used for fueling and relied upon as partial definition of containment due to its built-in radionuclide retention boundaries, codified regulatory requirements will still not be met. As such, some sort of defense-in-depth approach will need to be applied to obtain a certificate of compliance.

The use of impact limiting feature(s) or energy-consuming devices is typically incorporated to protect vital containment boundary components (i.e., end closures and ports) in the design of a package against severe NCT loads (i.e., free drops) and HAC loads (i.e., free drop, crush, puncture, and fire). Adequate protection and satisfactory performance of the containment vessel is typically demonstrated by subjecting the packaging definition to NCT and HAC regulatory performance tests to verify the containment vessel maintains containment and structural integrity.

Containment design requirements for packaging definition designs that are used to transport radioactive materials must be developed to ensure that any release of radioisotopes during postulated NCT or HAC events falls within the specified regulatory limits.

Primary regulatory documents that contain the general requirements for containment are 49 CFR Part 173 and 10 CFR Part 71 and condition specific requirements are listed in 10 CFR 71.51 and 10 CFR 71.71 for NCT and in 10 CFR 71.51 and 10 CFR 71.73 for HAC. Containment requirements are specified in 10 CFR 71.43, General Standards for All Packages. The phrase "no loss or dispersal of radioactive contents" is clarified in 10 CFR 71.51, Additional Requirements for Type B Packages.

Although shipping packages are designed to contain radioactivity and to maintain their structural integrity under the most severe reasonably anticipated conditions, leak testing is necessary to ensure that the packages are manufactured and assembled correctly and that no unacceptable leak paths have developed from subsequent use [17]. The limits for maximum permissible rate of release of contents is specified in 10 CFR 71.51 for both NCT and HAC and are 10^{-6} A_2 per hour and one A_2 in one week, or 10 A_2 Kr-85 in one week, respectively. Additionally, NCT testing is to be conducted at the most unfavorable conditions of external temperature (between -29 °C and +38 °C) and pressure (between 25 kPa absolute and 140 kPa absolute).

Demonstration of compliance with specified limits should be in accordance with the methods outlined in ANSI N14.5-2014 [15] as it provides the bridge from the regulatory release rate requirement to a measurable allowable leakage rate. ANSI N14.5-2014 also provides a list and descriptions of several accepted leakage rate test methodologies. More often than not, designers/developers will alternatively select a leaktight definition (ANSI N14.5-2014 defines a leakage rate of 10^{-7} ref-cm^3/s or less as leaktight) as a less-complex means of determination of acceptance. If this is done, then establishment of a content-dependent leakage rate determination is not required.

Demonstration and successful determination of an adequate containment system is typically expected and readily achievable for microreactor associated spent fuel anticipated to be shipped separately with a purpose-built spent fuel shipping package. However, shipping a microreactor with irradiated contents is more challenging because the reactor vessel has to serve as a containment vessel for reactor operation, as well as meeting the previously stated requirements for transportation, after being potentially partially disassembled for transportation purposes. A microreactor with irradiated contents will likely not meet a leaktight definition. This is especially true when considering the significant structural and thermal loading evaluations prescribed by the regulations prior to demonstrating the efficacy of the containment system.

Leakage rate testing is required prior to first use, periodically (annually for standard definition Type B packaging) during package life, and following maintenance or repair activities. Leakage rate tests may also be required during design and associated verification testing, fabrication, and preshipment. National Nuclear Security Administration (NNSA) SG-100 [18] describes and gives excellent examples of leak testing methods and supported details to consider while applying ANSI N14.5-2014.

Finally, the use of pressure vents and valves is not encouraged on packages. Pressure vents and valves must comply with 10 CFR 71.43(e) if deemed necessary for use, and the concept of a fail-safe mode must also be demonstrated.

Shielding. Design of packaging radiation shielding is concerned with establishing that the regulatory radiation dose rate limits on the exterior of the packaging definition are not exceeded. The same calculations that produce radiation source term evaluations for shielding analyses are also typically used to determine the heat sources used in the thermal analyses. In addition to shielding, other aspects of the package that are included are subcriticality, structural integrity, containment (for both the contents escaping from the package or outside material entering), thermal management, and the thermal conditions presented by an external heat source. Shielding performance requirements are listed in 10 CFR 71.47 and 10 CFR 71.51.

10 CFR 71.47 specifies dose rate limits for the microreactor packaging definition anticipated to be used for the purposes of transport. A package containing radioactive material must be designed and prepared for shipment such that under NCT, the radiation dose rate does not exceed 2 mSv/h at any point on the external surface of the package, as specified by 10 CFR 71.47, 49 CFR 173.441, and IAEA Regulations for the Safe Transport of Radioactive Material (SSR-6) [19]. The maximum dose rate at one meter from any external surface position of the package under NCT must also not exceed 0.1 mSv/h.

The NCT dose rate limits apply to a shipping package without regard to the method of shipment. If the package is shipped as exclusive use, which will likely be the case in this instance, the limits can be relaxed to take into account the material and geometric shielding properties of the conveyance vehicle (see 10 CFR 71.4 for the definition of “exclusive use”). A maximum package external dose of 10 mSv/h for NCT is allowed in a closed vehicle if the 2 mSv/h limit is met on the external surface of the vehicle. The details of exclusive use limits are given in 10 CFR 71.47(a) and (b).

The maximum dose rate at one meter from any external surface position of the package under HAC must not exceed 10 mSv/h, as specified in 10 CFR 71.51(a)(2). For HAC, the dose rate of the external package surface is assumed to be that defined by the configuration conservatively established by its post-accident geometric state.

Criticality. Subcriticality design and performance requirements are described in 10 CFR 71.55 through 10 CFR 71.61. NNSA SG-100 [18] gives excellent recommendations regarding establishing a criticality safety basis for package licensing.

All performance standards and regulatory requirements for U.S. certification of fissile material packages are prepared by the NRC and provided in 10 CFR Part 71. The NRC and the DOT work together to ensure that the DOT regulations for transport of hazardous material (49 CFR Part 173 and 10 CFR Part 71) are consistent. The performance standards and requirements for certification of packages containing fissile material are only included in 10 CFR Part 71. For international shipments, the IAEA sets forth performance standards and requirements in IAEA SSR-6 [19]. Minor differences exist between the performance standards of IAEA SSR-6 and the 10 CFR Part 71 regulations, but the fissile material requirements are essentially the same.

For packages that transport fissile material, such as a microreactor with its unirradiated or irradiated contents, adequate protection is provided by using a design and safety-assessment philosophy that effectively eliminates the possibility of a criticality event occurring under any and all credible non-operating as well as transport scenarios. As such, the microreactor system (defining the package) must always maintain subcriticality in all conceivable transport configurations and during all transportation modes.

A detailed consideration of the many parameters that interact to influence the neutron behavior is needed to provide an adequate safety basis for the package design definition. Principal parameters that affect criticality safety are: 1) type, mass, and form of the fissile material; 2) moderator-to-fissile material ratio (degree of moderation); 3) amount and distribution of absorber materials; 4) package geometry—internal and external; and 5) reflector effectiveness.

The conditions prescribed by the regulations require computational evaluations that incorporate statistical techniques to model neutron transport and predict k_{eff} for the system. Calculational biases and uncertainties are also part of this determination. The margin of subcriticality allowed for a package configuration must include the effect of these biases and uncertainties, together with design uncertainties, and an additional subcritical margin that would provide subcriticality for all non-operating credible scenarios even in the absence of all uncertainties.

Restriction of fissile mass or using a favorable geometry to provide enhanced neutron leakage from the package definition as a means of controlling the neutron balance are not feasible options for a microreactor with unirradiated or irradiated contents. Instead, strategic incorporation of neutron poison materials and moderators are the primary means of controlling neutron balance. Neutron poisons added to the package definition require special attention because their presence must be certain under all conditions and because their incorporation may change the mechanical and/or thermal properties of host materials. Additionally, the geometry of heterogeneous fissile material (e.g., TRISO fuel compacts), the design and placement of absorber materials, and the separation between fissile materials are all important to the criticality evaluation.

Whatever the control mechanism, an adequate margin of subcriticality must be demonstrated for both the single package in isolation and for arrays of packages, despite the fact that arrays of packages would be unlikely. Undamaged (normal transport conditions - NCT) and damaged (subsequent to accident conditions - HAC) packages must be considered using the credible fissile material configuration and the moderator and reflector conditions that provide the maximum effective neutron multiplication factor, k_{eff}.

Criticality safety evaluations must be performed in establishing the safety basis of the packaging definition to demonstrate that the package will remain in a subcritical condition under NCT (see 10 CFR 71.71) and HAC (see 10 CFR 71.73). Domestic and international regulations require that a single, water-flooded package be adequately subcritical both in the undamaged or damaged condition. The specific domestic requirements that must be met for a single package certified to carry fissile material are described in 10 CFR 71.55.

The undamaged package is considered to be the physical condition of the package under the NCT; the damaged package refers to the physical condition of the package following its exposure to the tests for the HAC. All internal voided volumes of the package, including the containment vessel, must be assumed to be filled with optimum water moderation for this evaluation. The fissile material contents used in the evaluation must also be in the most reactive credible condition consistent with NCT and HAC. All forms of hydrogenous moderation are intended for consideration in determining the optimum moderation (i.e., moderation conditions for highest k_{eff} value).

Regulations state that the criticality safety evaluation must include an analysis to determine whether the portion of the package defined as the containment system, if closely reflected by water, would have a greater reactivity (higher k_{eff}) than the packaging in combination with water. If the package and containment system are not the same, then two analyses must be done for the package in its undamaged condition—one with a water-flooded and water-reflected containment system separate from the package and one with a water-flooded and water-reflected package. The results from these two system analyses should be compared to select the one with the highest k_{eff} as the limiting case for the certification process. In short, the packaging design definition must evaluate all possible moderation and reflection configurations as part of establishing their safety basis.

A requirement included in the domestic 10 CFR 71.55(f) and international regulations applies to packages containing fissile material that will be shipped by air. This requirement is provided to preclude a rapid approach to criticality that may arise from potential geometrical changes in a single package (thus, water in-leakage does not need to be considered) subsequent to the tests (developed to be consistent with HAC that might arise from air transport) prescribed in 10 CFR 71.55(f).

Quality Assurance. There are many critical aspects to consider when establishing a quality assurance (QA) program geared to lead the development of a microreactor that can be shipped with its unirradiated or irradiated contents.

Quality control throughout the design process is as important to the successful certification of a package as the package's ability to successfully complete the regulatory testing and verification of safety basis (simulated or physical). The U.S. Department of Energy (DOE) as well as NRC certifying bodies require applicants for packaging certification and shipment to adhere to 10 CFR 71, Subpart H requirements. QA programs based on implementation of a full-featured NQA-1 program [20] are acceptable as well.

A description requirement to partially address 10 CFR 71.31, is to identify established codes and standards proposed for use in the package design, materials of fabrication, fabrication, assembly, testing, maintenance, and use. In the absence of codes, standards, and applicable code cases, the basis and rationale used to formulate the QA program needs to be described and justified to the regulatory faction.

It is strongly recommended that the QA program ensure that activities important to safety and applicable to the design, purchase, fabrication, and testing of packaging be described by written procedures and instructions that will be approved by a certifying agency official and be in effect prior to engaging in these activities. A Quality Assurance Plan (QAP) developed for the program should address all 18 elements required and identify the procedures that will be used to achieve the applicable development quality requirements for the packaging definition. The QAP should be developed, submitted to the certifying regulatory body for approval, and available for direct application prior to the initiation of the design and development effort.

QA specialists should have experience with packaging regulations, DOE orders, and national and international standards relating to quality assurance in addition to 10 CFR 71, Subpart H requirements. The applicant's QA program should detail their approach to the control of purchased items and services in order to fulfill the requirements of 10 CFR 71.115, Control of Purchased Material, Equipment, and Services and 10 CFR 71.109, Procurement Document Control. Vendors should be carefully selected based on their capability to comply with applicable sections of 10 CFR 71, Subpart H, their facility and QA program, and their previous records and performance. Also, all activities related to fabrication should be documented in a SARP and conducted under a certifying regulatory body's approved QA program.

Regulatory Guide 7.10 [21] was developed to provide individuals subject to the QA requirements of 10 CFR 71, Subpart H, with guidance on developing QA programs for implementation with respect to the transport of radioactive materials in Type B and fissile material packages. Regulatory Guide 7.10 establishes a graded quality safety category system that delineates packaging definition items between: 1) critical to safe operation, 2) has a major influence on safety, 3) only has a minor influence on safety. NNSA SG-100 [18] gives examples of this delineation process.

Radiation shielding evaluation aspects, all physical testing initiatives, leakage rate testing, and instrument calibration also need to be performed in accordance with a written and accepted QA program and QA plan that conform with the applicable requirements of 10 CFR 71, Subpart H, and other relevant codes and standards. Additionally, measures must be established to ensure that test results are documented, evaluated, and maintained as QA records to meet the requirements of 10 CFR 71.123.

Issues Associated with Defense-in-Depth

Defense-in-depth approaches have primarily been used in the certification of SNF transportation casks to address uncertainties in the performance of high burnup spent nuclear fuel (SNF) cladding during NCT or HAC. For example, for the transport of high burnup SNF that has been in storage for longer than 20 years, NUREG-2224 [22] discusses the option of performing supplemental safety analyses assuming fuel reconfiguration scenarios. In addition, NUREG-2216 [23] discusses fuel reconfiguration-based analyses as defense in depth strategies for high burnup SNF.

These fuel reconfiguration scenarios would involve performing thermal, shielding, and criticality evaluations that include the fuel reconfiguration scenarios contained in NUREG/CR-7203 [24]. Burnup credit and water in-leakage could also potentially be used as defense-in-depth strategies.
For microreactor fissile material and Type B transportation packages, defense-in-depth strategies could also be used during the transportation package certification process. However, the defense-in-depth strategies used in the certification of light-water-reactor SNF transportation casks are based on light-water-reactor fuel, not on a microreactor or its associated fuel, and would need modification to adequately assess microreactor fuels.

OPTIONS FOR ADDRESSING ISSUES

Transport Microreactor and Fuel Separately

One option for addressing most of the issues associated with transporting a microreactor is to ship the microreactor without unirradiated or irradiated fuel. New transportation packages may have to be designed depending on the fuel type of the microreactor, but for TRISO fuel, these types of packages have been designed and certified by the NRC or DOE and there is no reason to believe that it would not be feasible to design and certify new transportation packages for TRISO fuel.

The ability to ship an irradiated reactor vessel either intact or segmented has been demonstrated numerous times in the commercial nuclear power industry. Shipping the microreactor without irradiated fuel would be similar to shipping an irradiated reactor vessel and there is no reason to believe that it would not be feasible to ship a microreactor without irradiated fuel either intact or segmented.

For transport by air, shipping the unirradiated or irradiated fuel separately from the microreactor might simplify the design and certification of the fissile material package for the fuel because the design would not have to consider the design of the microreactor, but the fissile material packages would still have to meet the enhanced testing standards contained in 10 CFR Part 71 for transport by air.

Transport Microreactor and Fuel Together

A second option for transporting a microreactor is to ship the microreactor together with its contents, i.e., unirradiated or irradiated fuel. This option is likely to be much more challenging than shipping the microreactor and its fuel separately because it may not be feasible from a weight and size perspective to deploy a microreactor that was also designed and certified as a fissile material package or as a Type B package. It should be noted that NRC has approved special package authorizations for Type B packages with a containment that is not leak tight but that demonstrated compliance with 10 CFR Part 71 leakage rates for NCT in 10 CFR 71.51(a)(1) and with HAC in 10 CFR 71.51(a)(2) [25].

NRC regulations [10 CFR 71.41 (c)] allow for environmental and test conditions different from those specified for NCT (10 CFR 71.71) and HAC (10 CFR 71.73) to be approved by the NRC if the compensatory measures proposed by the shipper are demonstrated to be adequate to provide equivalent safety of the shipment.

Probabilistic risk assessment (PRA) is one method that could be used to demonstrate the equivalent safety of transporting a microreactor together with its contents. For nuclear reactors, PRA has been conducted since the 1970s (e.g., see WASH-1400 [26], NUREG-1150 [27], and NUREG-1935 [28]). PRA has also been used to assess a dry cask storage system at a nuclear power plant (see NUREG-1864 [29]). PRA techniques have also been applied to the transportation of SNF, most notably in NUREG/CR-4829 [30], NUREG/CR-6672 [31], and NUREG-2125 [32]. Additional guidance is provided in IAEA [33].

A PRA for transporting radioactive materials would be based on accident event trees that represent a set of possible transportation accidents.

These event trees include accidents involving collisions and accidents that do not involve collisions. Collision accidents include accidents with non-fixed object (trains, trucks, other vehicles, etc.) and fixed objects (bridges, buildings, walls, etc.). Non-collision accidents include fires and explosions, jackknifes, rollovers, etc. Event trees are typically constructed using transportation accident data and geographic information system (GIS) data. As such, event trees can be made country-specific and also modified to include additional branches or exclude branches that are not applicable or of no interest.

At this time, microreactor designs are not finalized. Therefore, it is not known what specific NRC regulatory requirements will not be met. If a microreactor design does meet some of the NRC regulatory requirements, these areas could be excluded from the transportation PRA and a limited scope transportation PRA could be performed for those areas of the regulations that the microreactor design does not meet.

In order to establish safety equivalency using a transportation PRA, a safety goal will need to be established. For nuclear power plants, there are two quantitative safety goals (51 FR 30028) [34]:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

It should be noted that in transportation accident consequence assessments, the thresholds for prompt radiation-related fatalities are rarely if ever approached. For either of the safety goals associated with nuclear power plants, a consequence assessment would have to be performed. Potentially relevant consequence assessments include Reiss [35] and the consequence assessment contained in NUREG-2125 [32].

CONCLUSIONS

The ability to transport a microreactor and its unirradiated or irradiated fuel contents in compliance with transportation regulations is a critical component of the successful deployment of microreactors in the U.S. and overseas. At the same time, it may not be feasible for a microreactor to meet the codified regulatory requirements that a fissile material or Type B or C package must meet.

For this reason, future microreactor transportation work should be concentrated in the area of building the framework for a transportation PRA that would provide the basis for approving the transport of a microreactor and its contents by transportation regulators. The transportation PRA would build on the microreactor’s operating history and experience, benchmarked and validated analytical reactor predictive codes with data collected during operation/demonstration, and would leverage developed code cases. During the development of the transportation PRA, significant dialogue with the NRC would be required, with the goal of moving beyond one-at-a-time approvals to a process that is similar to the traditional package certification process. The ultimate goal of the transportation PRA would be the international transport of the microreactor which would require approval of a foreign competent authority.

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