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NPP Container Concept Evaluation

Final Report

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Executive Summary

TBD

1. Introduction

NPP's primary question for SNL is on the technical feasibility of the NPP container concept from the perspective of being able to meet technical and regulatory criteria for use in the storage, transport, and disposal of radioactive material. For the purposes of this discussion, SNL is to assume the container can be manufactured as described in the associated patents, and is not expected to evaluate the marketability of the product (e.g., cost competitiveness, market competition, etc.). Rather, SNL brings considerable experience and expertise in design, testing, and certification of radioactive material packaging, which can be useful in determining the viability of the NPP container concept in this highly regulated environment.

SAND2010-6561P, "NPP Container Concept Evaluation Interim Draft Report," was issued in October 2010, and provided a summary of the 10 CFR 71 Type B packagings currently certified by the U.S. Nuclear Regulatory Commission (NRC) for storage and transport of radioactive material and the associated regulatory criteria. The report includes a brief discussion of the NPP container concept and possible applications for that concept. Additional discussion between SNL and NPP since the release of that report has resulted in a better understanding by SNL of the NPP container concept and the nature of the information NPP is interested in obtaining from SNL.

Originally, at NPP's request, SNL's focus had been on evaluating the concept for use with commercial spent nuclear fuel, which was agreed during the PATRAM 2010 meeting in London in early October 2010 as not likely to be viable and which would involve significant capital investment, a long time frame and considerable technical risk. During the October discussions, it was decided that SNL should instead focus on storage and transport research reactor spent fuel. This material would still require a Type B package for transport, but involves much smaller inventories having less severe thermal and radioactive characteristics, and is an area less dominated by regulatory and major industrial interests. Consideration of the use of the NPP container for storage, transport, and disposal of other nuclear waste was identified as a secondary focus at the October meeting.

During the November discussions, several technical issues and uncertainties were identified related to the viability of the current NPP container concept for achieving certification as a Type B package for research reactor spent fuel. Design modifications were identified that could improve the potential viability, but entail an unknown degree of technical risk and will require extensive analysis and testing to reduce uncertainty. NPP's partnership with Trelleborg for the application of materials having unique thermal and impact absorption properties was viewed with great enthusiasm by SNL, and appears to have considerable promise in addition to potential applications for the NPP container.

From a more pragmatic perspective, the November discussions resulted in a decision to focus first on the use of the NPP concept for a Type A container for transport, storage, and disposal of low-level radioactive waste. This option was considered the most feasible option which presents the least technical risk, and is discussed first. Following that discussion, considerations associated with further developing the NPP container concept as a Type A container for storage and transport of spent research reactor fuel or other higher activity materials is presented.

2. Type A Waste Transport and Disposal Container Concept

This chapter begins with an overview of the low-level radioactive waste classification system used in the U.S. The waste classification system is based on the concentration (activity per unit volume) of radionuclides in the waste and the half-lives of those radionuclides, and is used to define several levels of protective measures established to ensure long-term radiological safety. In contrast to the concentration-based waste classification scheme, transportation of low-level radioactive waste is classified by total activity limits permitted in packagings designed to survive increasingly more severe accident conditions. Low-level radioactive waste transportation limits are described in Section 2. For the NPP container concept evaluation, the focus is first on a Type A transportation package for low-level waste transport and disposal, and later on a Type B transportation package for higher activity materials including waste and research reactor spent fuel. Section 3 next describes Type A radioactive materials packagings, including the requirements relevant to transportation as well as those relevant to disposal. In Section 4, two sets of calculations are described which were performed to evaluate the applicability of the NPP container concept as a Type A package for transport and disposal of low-level radioactive waste. Finally, Section 5 summarizes conclusions of the evaluation of the NPP container for this application.

2.1 LOW-LEVEL RADIOACTIVE WASTE CLASSIFICATION

In the U.S., near-surface land disposal of low-level waste is regulated by 10 CFR 61. Such wastes are classified as Class A, Class B, and Class C, depending on the concentrations of both long-lived and short-lived radionuclides in the waste material. Internationally, the IAEA ("Classification of Radioactive Waste," General Safety Guide No. GSG-1, 2009) classifies similar wastes as Low Level Waste (LLW) and Intermediate Level Waste (ILW). The concentration limits for waste classification were based on an evaluation of the performance of a disposal facility for protecting the general population from radioactive releases, protecting future inadvertent intruders, protecting individuals during disposal operations, and ensuring stability of the site after closure. Institutional controls are required for up to 100 years, a period during which Class A and Class B will decay to levels that present an acceptable hazard to a post-closure intruder. Class B and Class C waste should be designed to be stable over a period of 300 years. Waste that will not decay to an acceptable level within 100 years is designated as Class C, and is disposed of at greater depth or with the addition of intruder barriers; Class C waste decays to a level after 500 years that does not pose an unacceptable risk to an intruder or public health and safety. Low-level waste that exceeds Class C concentrations is referred to as "Greater-Than-Class C," and is not generally acceptable for near-surface disposal. Classification concentration limits for U.S. Low-Level Waste are presented in Table 1 and 2 of 10 CFR 61.55, and are shown in the following tables. Also shown in the tables are the A_1 and A_2 activity limits for Type A transportation packagings from Appendix A to 10 CFR 71, and primary decay modes for these radionuclides.

Table 1 (Long-lived nuclides)

Nuclide	Class A limit (μCi/cm^3)*	Class C limit (μCi/cm^3)	A_1 (Ci) (special form)	A_2 (Ci) (normal form)	Decay Mode
C-14	0.8	8	1100	81	β
C-14 (activated metal)	8	80	1100	81	β

Nuclide	Class A limit (μCi/cm^3)*	Class C limit (μCi/cm^3)	A_1 (Ci) (special form)	A_2 (Ci) (normal form)	Decay Mode
Ni-59 (activated metal)	22	220	unlimited	unlimited	EC/X-ray
Nb-94 (activated metal)	0.02	0.2	19	19	β/γ
Tc-99	0.3	3	1100	24	β
I-129	0.008	0.08	unlimited	unlimited	β
	nCi/g	nCi/g			
α emitter w/τ > 5 yr	10	100	varies	varies	α
Pu-241**	350	3,500	1100	1.6	β
Cm-242	2,000	20,000	270	0.027	α

** 1 μCi/cm^3 = 1 Ci/m^3*
*** Pu-241 decays to Am-241, which decays with α and γ emissions*
Note: No Class B limits for long-lived nuclides
Note: Greater than A1/A2 quantities require Type B transport package

Table 2 (Short-lived nuclides)

Nuclide	Class A limit (μCi/cm^3)	Class B limit (μCi/cm^3)	Class C limit (μCi/cm^3)	A_1 (Ci) (special form)	A_2 (Ci) (normal form)	Decay Mode
Total all nuclides w/τ < 5yr	700	--*	--	varies	varies	varies
H-3	40	--	--	1100	1100	β
Co-60	700	--	--	11	11	γ
Ni-63	3.5	70	700	1100	810	β
Ni-63 (activated metal)	35	700	7,000	1100	810	β
Sr-90	0.04	150	7,000	8.1	8.1	β
Cs-137	1	44	4,600	54	16	γ

** implies no Class B and C limits for these nuclides. Other considerations, such as external dose for transportation or internal heat generation will limit these nuclides. Class B unless other nuclides independently determine a higher classification.*

According to the NRC, approximately 2 million cubic feet and 780 thousand curies of low-level radioactive waste were disposed of in the U.S. in 2008. (http://www.nrc.gov/waste/llw-disposal/licensing/statistics.html) An October 2009 report by the European Commission estimates that 2.5 million packages containing radioactive materials are shipped annually across the European Union, although most contain relatively small quantities of radioactive materials for medical uses. ("Preliminary Report on Supply of Radioisotopes for Medical Use and Current Developments in Nuclear Medicine," European Commission, October 2009) It should be noted that the bulk of the volume of these materials involves very low levels of radioactivity, and can be transported and disposed of in standard industrial packaging, unlike the more highly regulated Type A and Type B packages required for higher activity wastes.

The DOE Manifest Information Management System (MIMS) shows the following breakdown by volume and activity of the waste disposed in 2009 for each of the three classes of waste. (http://mims.apps.em.doe.gov/)

Classification	Class A	Class B	Class C	Total
Activity (Ci)	6,523.42	3,016.50	1,140.36	10,071.90
Volume (ft^3)	1,792,697.89	3,024.00	1,336.20	1,797,128.45

In this example year, approximately 99.8% of the volume consisted of Class A waste, which accounted for about 65% of the total activity. The remaining 0.2% of the volume (~4400 ft^3) included about 24% of the total activity as Class B waste, and the remaining roughly 12% as Class C waste.

2.2 LOW-LEVEL RADIOACTIVE WASTE TRANSPORTATION LIMITS

Regulations for transport packagings are promulgated by the Department of Transportation (DOT) in 49 CFR 173 and the Nuclear Regulatory Commission (NRC) in 10 CFR 71. Refer to Appendix A for the text of the more important of these requirements.

There are basically four kinds of radioactive material transport packaging types defined in 49 CFR 173:

- Excepted - extremely low level radioactivity; very low hazard if accidental release, e.g., smoke detector, ; 49CFR173.421
- Industrial - low activity, e.g., waste materials; no identifiable release during normal transport and handling; 49CFR173.411
- Type A - small quantities with higher radioactivity concentrations; typically constructed of steel, wood, fiberboard with an inner vessel surrounded by packaging material; integrity and shielding maintained under normal transport conditions; not designed to withstand accident forces, but consequence of release insignificant due to limited quantities;49CFR173.412
- Type B - used to transport highest level of radioactivity; design to survive Type A test and severe worst-case accident; Type B required for life endangering quantities; 49CFR173.411, 49CFR173.413, and 10CFR71

Appendix A to 10 CFR 71 provides tables of the A_1 and A_2 activity limits for Type A transportation packagings. A_1 values are based on the activity yielding an external gamma dose rate of 0.1 Sv/h (10 rem/h) at 1 m and apply to packaging of "special form" material. Special form material is defined in 49 CFR 173.403 as either an indispersible solid or sealed capsule. A_2 values are based on dispersal of the material from a breached package accounting for multiple internal and external exposure pathways, and apply to packaging of "normal form" material. These limits on the allowable activity that can be shipped in a Type A package were established to ensure radiological safety during normal transport conditions, but that would not produce significant radiological consequences if an accidental release occurred. Note that this transportation constraint limits the total activity for a Type A shipment of waste material to a disposal facility.

2.3 TYPE A RADIOACTIVE MATERIAL PACKAGINGS

The key Type A package requirements in 10 CFR 71.71 are for normal conditions of transport and include the following sequential tests:

- Water spray (5 cm/hr for 1 hr)
- Free drop (1.2 m for packages < 5000 kg)
- Stacking (5 x the package weight for 24 hours for packages < 5000 kg)
- Penetration (6 kg steel cylinder 3.2 cm in diameter from 1 m)

It does not seem unreasonable to expect that the NPP container concept could satisfy these test conditions.

Although Type A packages include those made of cardboard (prohibited for use in disposal by 10 CFR 61.56(a)(1)), wood boxes, or steel drums, the discussion of "Concepts" in the waste disposal requirement 10 CFR 61.7(b)(2), includes the statement *"…To the extent that it is practicable, Class B and C waste forms or containers should be designed to be stable, i.e., maintain gross physical properties and identity, over 300 years…",* and in 10 CFR 20 Appendix G, "Requirements for Transfers of Low-Level Radioactive Waste Intended for Disposal at Licensed Land Disposal Facilities and Manifests," the NRC defines the "High Integrity Container" (HIC) as *"…a container commonly designed to meet the structural stability requirements of § 61.56 of this chapter, and to meet Department of Transportation requirements for a Type A package."* (See Appendix A for the text of 10 CFR 61.56 regarding waste characteristics; DOT Type A transport packaging requirements are found in 10 CFR 71, in particular 10 CFR 71.71.) The requirements of 10 CFR 61.56 are not specifically associated with the packaging, but rather concern the characteristics of the waste itself, such as minimizing void space and the presence of liquids or reactive materials, and *"…placing the waste in a disposal container or structure that provides stability after disposal…"*.

In the U.S., existing design of containers approved as High Integrity Containers for Class B and C wastes and used at various low-level disposal sites in the U.S. have been constructed of reinforced concrete, corrosion-resistant metal alloys, polymer-coated metal, and high-density polyethylene (HDPE). One must determine how best to demonstrate stability of the package over 300 years, which may need to include consideration of high-density polymer creep, radiolytic or other environmental degradation effects, etc. The NRC Technical Position on Waste Form (Revision 1, January 1991), which is included in Appendix A, provides guidance on acceptable methods for demonstrating compliance with the waste form structural stability requirements of 10 CFR 61, and in Section C.4, describes design considerations relevant to high integrity containers.

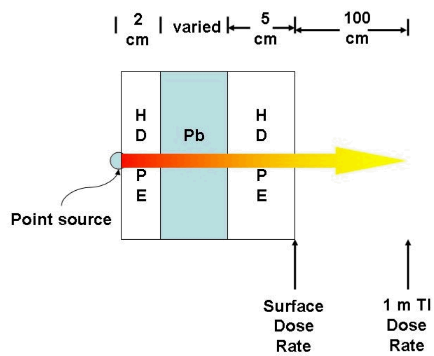
Development of the NPP container concept for use as a HIC for disposal of higher activity low level radioactive waste is being evaluated as a viable application with the least technical risk of applications which have been considered to date.

2.4 WASTE CALCULATIONS

In addition to the above discussion of the types of testing that would be required to certify the NPP container for this application, two types of calculations were performed to evaluate the types and concentrations of low-level radioactive wastes that might be used with the NPP container. First was a gamma radiation calculation of the thickness of the lead shielding with the nominal NPP container dimensions required to safely transport the maximum activity allowed for a Type A package for the major gamma emitting radionuclides Co-60, Cs-137, and Ir-192. Second was a calculation for several radionuclides of the concentrations of low-level waste that could be shipped as a Type A package.

A. Lead Shielding Thickness Calculation for Maximum Content for a Type A Package

A simple, conservative model was developed for calculating dose rates with MicroShield 8.03 for gamma emitting radionuclides for the NPP container geometry. A diagram of the model is shown below.



The calculation assumed a point source at the inner HDPE surface of the container, a lead shielding layer of variable thickness, and an outer 5 cm HDPE layer. No credit for shielding by the HDPE is assumed. Dose rates were calculated both with and without buildup at the package surface and at 1 m from the package surface. 10 CFR 71.47 requires for a Type A package that:

1. the surface dose rate is < 200 mrem/h (2 mSv/h), and
2. has a Transport Index < 10.

The Transport Index (T.I.) is defined as the maximum dose rate in mrem/h at 1 m.

Calculations were run for three gamma emitters, Co-60, Cs-137, and Ir-192 using the conservative model for both normal form and special form material. The activities used correspond to the 10 CFR 71 Appendix A values of A_1 for special form and A_2 values for normal form material. (A_1 values are based on the activity yielding an external gamma dose rate of 0.1 Sv/h (10 rem/h) at 1 m; A_2 values are based on dispersal of the material from a breached package accounting for multiple internal and external exposure pathways.)

Results for the conservative calculation indicate that for these source activities, the thickness of the lead shielding assuming surface buildup required is driven by the 200 mrem/h surface dose limit. These values are highlighted in the following table, which also shows the thicknesses of lead shielding to meet the surface dose criterion of 200 mrem/hr without buildup, and the Transport Index criterion of 10 mrem/h at 1 m without and with buildup.

Nuclide	Form	Activity (Ci)	Pb Shielding Thickness (cm)			
			Surface Dose (w/o buildup)	Surface Dose (w/buildup)	T.I. (w/o buildup)	T.I. (w/buildup)
Co-60	normal	11	12	13.7	10.7	12.6
Co-60	special	11	12	13.7	10.7	12.6

Cs-137	normal	16	4.8	5.3	3.5	4.0
Cs-137	special	54	4.8	5.4	3.6	4.1
Ir-192	normal	16	4.9	5.4	3.8	4.2
Ir-192	special	27	5.3	5.8	4.2	4.7

For Co-60, 13.7 cm of lead shielding for the nominal NPP container having a 19 cm inner radius and height of 120 cm corresponds to more than 3000 kg not including the lid or base. For Cs-137 and Ir-192, the 5.4 cm and 5.8 cm of shielding correspond to approximately 1000 and 1100 kg, respectively. Note that if the activities shown in the table above were contained in waste fully occupying the 1.36×10^5 cm^3 inner volume of the nominal NPP container, the Co-60 and Ir-192 would be classified as Class A low-level waste, and the Cs-137 as Class C.

Conclusion: This calculation suggests that approximately 5-6 cm of lead shielding used with the nominal Type A NPP container could be used to ship Type A quantities of Cs-137 and Ir-192 gamma emitting low-level waste. Considerably more shielding would be needed for shipping Type A quantities of Co-60 is probably not suitable for use with the NPP container.

B. Allowable Low-Level Waste Concentrations for a Type A Shipment

A second calculation was performed to determine if an NPP container could be filled with Class C waste and shipped in a Type A container.

NPP container inner volume = $\pi r^2 h = \pi(19\text{ cm})^2(120\text{ cm}) = 1.36 \times 10^5\text{ cm}^3$

Co-60:

Class A upper limit = 700 μCi/cm^3 = 7×10^{-4} Ci/cm^3;

$(7 \times 10^{-4}\text{ Ci/cm}^3)(1.36\times10^5\text{ cm}^3) = 95$ Ci

Filling the NPP container with Co-60 at the upper Class A limit exceeds the A_1 and A_2 values of 11 Ci for either normal or special form material, and only (11/95=) 12% of the volume can be occupied with waste of this concentration.

Cs-137:

Class C upper limit = 4600 μCi/cm^3 = 4.6×10^{-3} Ci/cm^3;

$(4.6 \times 10^{-3}\text{ Ci/cm}^3)(1.36\times10^5\text{ cm}^3) = 626$ Ci

Class B upper limit = 44 μCi/cm^3 = 4.4×10^{-5} Ci/cm^3;

$(4.4 \times 10^{-5}\text{ Ci/cm}^3)(1.36\times10^5\text{ cm}^3) = 5.98$ Ci

The values of A_1 and A_2 for Cs-137 are 54 Ci and 16 Ci, respectively. One may therefore ship Cs-137 Class A and B waste in an NPP container, but will be limited in the amount (or concentration) of Class C waste. That is, at the upper Class C limit, exceeding (16/626=) 2.5% of the volume with normal form material, or (27/626=) 4.3% of the volume with special form material will not qualify for Type A shipment.

Ir-192:

Ir-192 is not specifically called out in 10 CFR 61, but is used regularly for industrial applications. Its 74 day half-life implies the same 700 μCi/cm^3 Class A upper limit as Co-60 (Table 2 of 10 CFR

61.55 for nuclides with half-lives <5yr). Filling the NPP container with waste at the upper Class A limit implies a maximum of 95 Ci of Ir-192 in the container.

Filling the NPP container with Ir-192 at the upper Class A limit exceeds the A_1 and A_2 values of 27 and 16 Ci, respectively. For normal form material only (16/95=) 17% of the volume can be occupied with waste of this concentration, and for special form only (27/95=) 28.4% of the volume.

<u>Calculation for other long-lived nuclides:</u>

Nuclide	Class C limit (μCi/cm^3)	Limit (Ci)*	Comment
C-14	8	81	No gamma shielding needed (1.1 Ci in NPP container at Class C limit).
C-14 (activated metal)	80	1100	No gamma shielding needed (10.9 Ci of activated metal in NPP container at Class C limit).
Ni-59 (activated metal)	220	unlimited	~0.01 cm Pb needed to meet Surface Dose limit w/waste at Class C limit (29.9 Ci in NPP container)
Nb-94 (activated metal)	0.2	19	~3 cm Pb needed to meet Surface Dose limit w/waste at Class C limit (0.027 Ci in NPP container)
Tc-99	3	24	No gamma shielding needed (0.41 Ci in NPP container at Class C limit).
I-129	0.08	unlimited	~0.001 cm Pb needed to meet Surface Dose limit to meet Surface Dose limit w/waste at Class C limit (0.011 Ci in NPP container)
Pu-241	3500 nCi/g	1.6	TBD – requires assumed waste density
Cm-242	20000 nCi/g	0.027	TBD – requires assumed waste density

** Limit based on A_2 value for normal form material, except for activated metal which is considered special form and uses the A_1 limit.*

<u>Calculation for other short-lived nuclides:</u>

Nuclide	Class C limit (μCi/cm^3)	Limit (Ci)*	Comment
H-3	No Class C limit; Class A limit 40	1100	No gamma shielding required. The A_2 limit of 1100 Ci implies for the 1.36×10^5 cm^3 inner volume of the NPP container a maximum waste concentration of 8.06×10^3 μCi/cm^3.
Ni-63	700	810	No gamma shielding required. For Ni-63, Filling the NPP container at the Class C limit of 700 μCi/cm^3 implies 95.2 Ci of Ni-63. Ni-63 in normal form can be shipped in NPP container at Class C levels.

Nuclide	Class C limit (μCi/cm^3)	Limit (Ci)*	Comment
Ni-63 (activated metal)	7000	1100 (**A_1**)	No gamma shielding required. For Ni-63, Filling the NPP container at the Class C limit of 7000 μCi/cm^3 implies 952 Ci of Ni-63 as special form activated metal, less than the 1100 Ci A_1 limit. Ni-63 in special form can be shipped in NPP container at Class C levels.
Sr-90	7000	8.1	No gamma shielding required. Filling the NPP container with Sr-90 at the Class C limit, however, implies 952 Ci, considerably above A_2 limit; 8.1/952 = ~1% of the volume filled with Class C Sr-90 waste would exceed A_2.

** Limit based on A_2 value for normal form material, except for activated metal which is considered special form and uses the A_1 limit.*

Conclusion: For most of the other nuclides specified in 10 CFR 61 for classification of low-level waste, no or minimal gamma shielding is required. For the waste volume of the nominal NPP container, waste at the Class C limit can be shipped as a Type A package.

C. Overall Conclusion from Waste Calculations

Inclusion of the lead gamma shielding in the NPP concept indicates its potential usefulness for shipments of maximum Type A quantities of the important gamma emitting radionuclides Cs-137 and Ir-192. The much greater amount of shielding needed for the maximum Type A quantity of Co-60, however, implies the NPP container concept may not be suitable for Co-60 shipments.

With limited or no gamma shielding, the NPP container as a Type A package could be used to transport low-level radioactive waste for most other radionuclides up to the Class C limit without exceeding activity limits imposed by transportation regulations.

2.5 TYPE A TRANSPORT AND DISPOSAL EVALUATION CONCLUSION

Developing the NPP container as a Type A package does have the advantage of being subject to less rigorous testing requirements than is associated with the Type B packages used for larger quantity or higher activity radioactive materials. It seems reasonable to assume the nominal NPP container of lead shielding sandwiched between layers of HDPE can satisfy the 10 CFR 71.71 testing requirements (water spray, free drop, stacking, and penetration) for normal transport conditions. With sufficiently thick lead shielding, the container should also be suitable for the transport and disposal of Cs-137 and Ir-192 at their maximum permissible Type A transport activities. This transportation limit allows the transportation and disposal of Class A and B concentrations of Cs-137 waste, but limits the concentration or quantity of Class C Cs-137 waste that can be transported to a disposal facility. Type A transport limits for Ir-192 limit it to a fraction of the Class A waste limits for disposal. Not yet considered is calculation of waste quantities and concentrations if the container were used solely for disposal, in which case the waste classification concentrations and quantities may be increased and other considerations, such as disposal worker dose or internal heat generation issues may be important. With little or no lead shielding, most other radionuclides could be transported and disposed of at Class C levels without exceeding transportation limits. Additional research on waste generation, i.e., quantity, form and content of waste intended for use with this container, could be useful in establishing market potential. For example, Co-60 and Ir-192 waste materials may in general exceed Type A limits and require shipment in Type B packagings. Assuming the stability characteristics of the package can be established for the 300-year regulatory lifetime, it should be possible to certify the container as a High Integrity Container for disposal of low-level radioactive waste, although activity limits for transportation in Type A packagings constrain the quantity and concentration of the waste.

3. Research Reactor Spent Fuel Storage and Transport Container Concept

In the earlier report, SAND2010-6561P, only six NRC Certificates of Compliance were associated with research reactor fuel; of these, only two, USA/9228/B(U)F-96 (GE Hitachi Nuclear Energy Americas, LLC) and USA/9341/B(U)F-96 (AREVA Federal Services LLC), are certified for transporting spent fuel for these reactors. An overview of the research reactor inventory is provided in Appendix B, which shows the wide variety of reactor and fuel types used in the 249 operational research reactors. Research reactors use much less fuel than commercial power reactors, and the spent fuel has far less fission product build-up and much lower thermal power output.

The application of the NPP container concept to this material inventory was suggested because the radioactivity content and thermal output of spent fuel from these reactors is much lower than that of commercial power reactors, and because the small inventory and wide variation in fuel characteristics has not been subjected to the regulatory and industry interest associated with commercial power reactors. Research reactors are typically operated by small governmental or research institutes who may have an interest in dry storage or transport of spent fuel to regional storage facilities, but are unlikely to have resources available to develop and certify packaging for that application.

3.1 SPENT NUCLEAR FUEL STORAGE REQUIREMENTS

Spent Fuel Storage Casks are certified under 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-related Greater Than Class C Waste." In particular, the requirements of §72.236, "Specific requirements for spent fuel storage cask approval and fabrication," are relevant to this evaluation and are provided in Appendix A. In evaluating the NPP container concept against the storage requirements of 10 CFR 72, a strong recommendation was made that the conceptual package consisting of a layer of lead shielding sandwiched between two layers of high-density polyethylene (HDPE) should be encased with a steel lining for both the inner volume surface and the external surface. This additional liner material was viewed as helpful in demonstrating satisfaction of several of the storage requirements, including redundant sealing, maintenance, facilitating decontamination, simplifying inspection criteria, aiding heat removal, ensuring compatibility with loading and unloading facilities, and supporting confinement testing. Storage requirements include demonstrating reasonable maintenance of containment under normal, off-normal, and credible accident conditions. Some of the more important 10 CFR 72 requirements include:

- Maintaining subcriticality under credible conditions
- Sufficient radiation shielding and confinement
- Redundant sealing of confinement
- Adequate passive heat removal
- Minimum storage and maintenance for 20 years
- Compatibility with wet or dry spent fuel loading and unloading facilities
- Facilitate decontamination

- Permit inspection for confinement effectiveness
- Demonstrate reasonable confinement for normal, off-normal, and credible accidents

3.2 RESEARCH REACTOR SPENT FUEL STORAGE EVALUATION CONCLUSION

With the addition of a steel liner to protect both the inner volume and the external surface of the NPP HDPE/Pb container, it should be possible to satisfactorily address the storage requirements of 10 CFR 72. Note that additional research is needed on the inventory and characteristics of research reactor spent fuel, including the geometry, radiological activity, and decay heat, to support development of a marketable product.

3.3 TYPE B SPENT NUCLEAR FUEL TRANSPORTATION REQUIREMENTS

Transportation of these materials requires a Type B package, which must be subjected to the extensive testing requirements of 10 CFR 71 described in Appendix A. In addition to the tests required for Type A packaging, described above, Type B packages are subjected to a number of additional requirements including impact, fire, and water immersion tests conducted in sequential order and in the most damaging configuration. These tests include:

- Impact (9m drop onto an unyielding target, and 1 m drop onto 15 cm steel par on an unyielding target)
- Fire (fully engulfing 30 min fire with an 800ºC minimum average temperature)
- Water immersion (15 m, and 0.9 m if fissile, and 200 m for larger quantities of radioactive material)

To satisfy these requirements, a steel liner is also recommended as being able to contribute to the mechanical, heat transfer, and structural integrity of the package. For example, a steel outer layer can provide much protection to the outermost HDPE layer during the 1-meter puncture test and would add structural rigidity to the body of the package to minimize overall deformations. A steel outer layer would also provide some thermal protection to the HDPE and the lead when the package is exposed to the regulatory fire. This is, nevertheless, an area of concern, as phase change and deformation of the HDPE and the lead due to the exposure of the package to the 800ºC regulatory fire are likely. Additionally, a steel "casing" can make the design of the closure system easier and more resistant to leakage during and after imposing the mechanical and thermal loads required by the regulations.

The Trelleborg rubber material discussed at the November meeting is also viewed as a promising candidate for future development of the NPP container concept as a Type B package, both for its thermal properties as well as its mechanical properties for serving as an impact limiter. Most impact limiters currently contain redwood because of its unique mechanical and thermal properties. Recently, however, due to the limited availability of redwood, techniques using foam and aluminum honeycomb are being developed as an alternative; the Trelleborg material is of interest as a potentially even better alternative.

In certifying spent fuel transport casks the cask designs require impact limiters to provide protection in both hypothetical accident conditions and in normal conditions of transport. The certification process requires protection to both thermal and structural scenarios. Historically, redwood was used as an acceptable impact limiter material. Redwood provides both the necessary structural and thermal protection. However, over the past 30 years an acceptable quantity of this material has become increasingly more difficult to secure. Over the past 5 years Sandia has been involved in programs to design spent fuel transportation cask impact limiters that utilize alternate materials to provide structural and thermal properties that are acceptable for certification. These materials have included aluminum honeycomb, perforated aluminum sheets and molded foam. Other DOE-funded organizations have also pursued evaluating alternate materials. These programs have resulted in limited success and a 'perfect' material has yet to be identified that can meet both the thermal and structural requirements.

Sandia staff believes that the Trelleborg material properties shows great promise in possibly meeting both the thermal and structural conditions necessary in the design of a spent fuel transport cask impact limiter. Sandia would be very interested in having an opportunity to perform small scale engineering tests on this material to better determine its structural and thermal properties at the Sandia Actuator Test Facility and the Thermal Test Complex to assist in determining if this material would be acceptable for further evaluation. Given its unique properties, there may be other applications of this material and additional dialogue within Sandia organizations and within Lockheed Martin Corporation could possibly highlight other uses for this material.

Considerable research, development, testing and analysis will be needed to determine if the current NPP container design concept can be adapted to validate the ability of this packaging to satisfy the stringent testing and certification requirements.

3.4 Research Reactor Spent Fuel Type B Transport Evaluation Conclusion

The current design concepts for the NPP container do not appear sufficiently mature to recommend pursuing development of a Type B transportation package for research reactor spent fuel. A considerable period of development and expenditure of resources with a significant degree of technical risk will be needed to determine if this application would be successful.

4. Observations and Conclusions

The NPP container concept is clearly at an early conceptual stage without any mature designs or much supporting technical documentation to fully establish its viability for the radioactive material packaging market. The basic concept of a package incorporating lead (Pb) gamma shielding between layers of high-density polyethylene (HDPE) is not particularly unique, and examples have been identified where these materials are used for other available radioactive material packagings. Any uniqueness of the basic concept is attributed to the patented production process, which is also at a very early developmental stage.

NPP has proposed ideas for utilizing certain Trelleborg rubber products having unique properties for providing thermal and mechanical protection for the basic container. These Trelleborg materials do appear to have significant potential for application to radioactive material packaging, but require further evaluation and testing to fully establish their characteristics and applicability.

The NPP container concept was evaluated for several different potential applications, including the original application for storage and transport of commercial spent nuclear fuel, storage and transport of research reactor spent nuclear fuel, and most recently, as a Type A package for storage, transport, and disposal of low-level radioactive waste. The following table summarizes the evaluation by Sandia of the viability of the NPP container concept for these applications.

Evaluation Summary

Application	Storage	Transport	Disposal
Low-level Radioactive Waste	Not specifically evaluated, but reasonably viable as a Type A package.	Reasonably viable as a Type A package.	Reasonably viable.
Research Reactor Spent Nuclear Fuel	Reasonably viable w/addition of steel liner.	Potentially viable w/Type B development.	Likely precluded by regulatory uncertainty and lack of disposal facilities.
Commercial Spent Nuclear Fuel	Not considered viable at this time.	Not considered viable at this time.	Not viable at this time. Likely precluded by regulatory uncertainty and lack of disposal facilities.

Observations made and discussions held during the 16th International Symposium on the Packaging and Transport of Radioactive Materials (PATRAM) in October 2010, along with information compiled in September 2010 for the NPP Container Concept Evaluation Interim Draft Report, resulted in a conclusion that the NPP container concept would not be a viable application for the commercial spent nuclear fuel market segment due to significant technical, regulatory and marketplace issues.

At the PATRAM meeting, application of the NPP concept to the storage and transport of research reactor spent fuel was identified as the second candidate for evaluation. For a number of reasons, it was recommended that the package must have a steel liner inside and out for this application. With that constraint, application of the NPP container concept for storage appears reasonably viable. Transportation, however, would require satisfying Type B packaging requirements, which would entail considerable technical risk and technical development, and is not considered particularly viable at this point in time. A development process that incorporates the Trelleborg products with the steel-lined container, however, could potentially address these requirements at some lower degree of technical risk and could also lead to 'spin off' applications for other packaging applications including commercial spent fuel transport.

Discussions held in early November at Sandia with NPP led to the identification of a third evaluation candidate – use of the NPP concept as a Type A transport package for the storage, transport, and disposal of low-level radioactive waste. Evaluation of this application led to a conclusion that development of the NPP container as a Type A package for storage and transport of much of this waste is a reasonably viable option, although minimal gamma shielding would be required in general. Provided the long-term (300 year) stability of the container can be established, application of the NPP container concept as a high-integrity container for disposal of low-level radioactive wastes should also be reasonably viable.

Appendix A. Regulatory Requirements

1. Low-Level Waste Stability Requirements

10 CFR 61.7(b)(2)
"…To the extent that it is practicable, Class B and C waste forms or containers should be designed to be stable, i.e., maintain gross physical properties and identity, over 300 years…."

10 CFR 61.56 Waste characteristics.

(a) The following requirements are minimum requirements for all classes of waste and are intended to facilitate handling at the disposal site and provide protection of health and safety of personnel at the disposal site.

(1) Waste must not be packaged for disposal in cardboard or fiberboard boxes.

(2) Liquid waste must be solidified or packaged in sufficient absorbent material to absorb twice the volume of the liquid.

(3) Solid waste containing liquid shall contain as little free standing and noncorrosive liquid as is reasonably achievable, but in no case shall the liquid exceed 1% of the volume.

(4) Waste must not be readily capable of detonation or of explosive decomposition or reaction at normal pressures and temperatures, or of explosive reaction with water.

(5) Waste must not contain, or be capable of generating, quantities of toxic gases, vapors, or fumes harmful to persons transporting, handling, or disposing of the waste. This does not apply to radioactive gaseous waste packaged in accordance with paragraph (a)(7) of this section.

(6) Waste must not be pyrophoric. Pyrophoric materials contained in waste shall be treated, prepared, and packaged to be nonflammable.

(7) Waste in a gaseous form must be packaged at a pressure that does not exceed 1.5 atmospheres at 20^{o}C. Total activity must not exceed 100 curies per container.

(8) Waste containing hazardous, biological, pathogenic, or infectious material must be treated to reduce to the maximum extent practicable the potential hazard from the non-radiological materials.

(b) The requirements in this section are intended to provide stability of the waste. Stability is intended to ensure that the waste does not structurally degrade and affect overall stability of the site through slumping, collapse, or other failure of the disposal unit and thereby lead to water infiltration. Stability is also a factor in limiting exposure to an inadvertent intruder, since it provides a recognizable and nondispersible waste.

(1) Waste must have structural stability. A structurally stable waste form will generally maintain its physical dimensions and its form, under the expected disposal conditions such as weight of overburden and compaction equipment, the presence of moisture, and microbial activity, and internal factors such as radiation effects and chemical changes. Structural stability can be provided by the waste form itself, processing the waste to a stable form, or placing the waste in a disposal container or structure that provides stability after disposal.

(2) Notwithstanding the provisions in § 61.56(a) (2) and (3), liquid wastes, or wastes containing liquid, must be converted into a form that contains as little free standing and noncorrosive liquid as is reasonably achievable, but in no case shall the liquid exceed 1% of the volume of the waste when the waste is in a disposal container designed to ensure stability, or 0.5% of the volume of the waste for waste processed to a stable form.

(3) Void spaces within the waste and between the waste and its package must be reduced to the extent practicable.

2. High Integrity Container Waste Form Technical Position

Reference: Technical Position on Waste Form, U.S. NRC, January 1991, Revision 1.

Section C.4. High Integrity Containers

a. The maximum allowable free liquid in a high integrity container should be less than one percent of the waste volume as measured using the method described in ANS 55.1. A process control program should be developed and qualified to ensure that the free liquid requirements in 10 CFR Part 61 will be met upon delivery of the wet solid material to the disposal facility. This process control program qualification should consider the effects of transportation on the amount of drainable liquid which might be present.

b. High integrity containers should have as a design goal a minimum lifetime of 300 years. The high integrity container should be designed to maintain its structural integrity over this period.

c. The high integrity container design should consider the corrosive and chemical effects of both the waste contents and the disposal environment. Corrosion and chemical tests should be performed to confirm the suitability of the proposed container materials to meet the design lifetime goal.

d. The high integrity container should be designed to have sufficient mechanical strength to withstand horizontal and vertical loads on the container equivalent to the depth of proposed burial assuming a cover material density of 120 lb/ft^3. The high integrity container should also be designed to withstand the routine loads and effects from the waste contents, waste preparation, transportation, handling, and disposal site operations, such as trench compaction procedures. This mechanical design strength should be justified by conservative design analyses.

e. For polymeric material, design mechanical strengths should be conservatively extrapolated from creep test data. It should be demonstrated for high integrity containers fabricated from polymeric materials that the containers will not undergo tertiary creep, creep buckling, or ductile-to-brittle failure over the design life of the containers.

f. The design should consider the thermal loads from processing, storage, transportation and burial. Proposed container materials should be tested in accordance with ASTM B553 in the manner described in Section C2(b) of this technical position. No significant changes in material design properties should result from this thermal cycling.

g. The high integrity container design should consider the radiation stability of the proposed container materials as well as the radiation degradation effects of the wastes. Radiation degradation testing should be performed on proposed container materials using a gamma irradiator or equivalent. No significant changes in material design properties should result following exposure to a total accumulated dose of 10 E+8 Rads. If it is proposed to design the high integrity container to greater accumulated doses, testing should be performed to confirm the adequacy of the proposed materials. Test specimens should be prepared using the proposed fabrication techniques.

   High integrity container designs using polymeric materials should also consider the effects of ultra-violet radiation. Testing should be performed on proposed materials to show that no significant changes in material design properties occur following expected ultra-violet radiation exposure.

h. The high integrity container design should consider the biodegradation properties of the proposed materials and any biodegradation of wastes and disposal media. Biodegradation testing should be performed on proposed container materials in accordance with ASTM G21 and ASTM G22. No indication of culture growth should be visible. The extraction procedure described in Section C2(d) of this technical position may be performed where indications of visible culture growth can be attributable to contamination, additives, or biodegradable components on the specimen surface that do not affect the overall integrity of the substrate. It is also acceptable to determine biodegradation rates using the Bartha-Pramer Method described in Section C2(d). The rate of biodegradation should produce less than a 10 percent loss of the total carbon in the container material after 300 years. Test specimens should be prepared using the proposed material fabrication techniques.

i. The high integrity container should be capable of meeting the requirements for a Type A package as specified in 49 CFR 173.411 and 173.412. Conditions that may be encountered during transport or movement are to be addressed by meeting the requirements of 10 CFR 71.71.j. The high integrity container and the associated lifting devices should be designed to withstand the forces applied during lifting operations. As a minimum, the container should be designed to withstand a 3g vertical lifting load.

k. The high integrity container should be designed to avoid the collection or retention of water on its top surfaces in order to minimize accumulation of trench liquids which could result in corrosive or degrading chemical effects.

l. High integrity container closures should be designed to provide a positive seal for the design lifetime of the container. The closure should also be designed to allow inspections of the contents to be conducted without damaging the integrity of the container. Passive vent systems should be designed to minimize the entry of moisture and the passage of waste materials from the container.

m. Prototype testing should be performed on high integrity container designs to demonstrate the container's ability to withstand the proposed conditions of waste preparation, handling, transportation and disposal.

n. High integrity containers should be designed, fabricated, and used in accordance with a quality assurance program. The quality assurance program should address the following topics concerning the high integrity container: fabrication, testing, inspection, preparation for use, filling, storage, handling, transportation, and disposal. The quality assurance program should also address how wastes which are detrimental to high integrity container materials will be precluded from being placed into the container. Special emphasis should be placed on fabrication process control for those high integrity containers which utilize fabrication techniques such as polymer molding processes.

Section C2(b)

Waste specimens should be resistant to thermal degradation. The heating and cooling chambers used for the thermal degradation testing should conform to the description given in ASTM 553, Section 3 (Ref. 7). Samples suitable for performing compressive strength tests in accordance with ASTM C39 or ASTM D1074 should be used. Samples should be placed in the test chamber and a series of 30 thermal cycles carried out in accordance with Section 5.4.1 through 5.4.4 of ASTM 553. The high temperature limit should be 60ºC and the low temperature limit -40ºC. Following testing the waste specimens should have the maximum practical compressive strengths; (a minimum compressive strength of 60 psi as tested using ASTM D1074 is acceptable for bituminized waste forms--for cement-stabilized wastes see Section I.C of Appendix A).

Section C2(d).

Specimens for each proposed waste stream formulation should be tested for resistance to biodegradation in accordance with both ASTM G21 and ASTM 622 (Refs. 8 & 9, respectively). No indication of culture growth should be visible. Specimens should be suitable for compression testing in accordance with ASTH C39 or ASTM 1074, as applicable. Following the biodegradation testing specimens should have the maximum practical compressive strengths a minimum compressive strength of 60 psi as tested using ASTH D1074 is acceptable for bituminized waste forms--see Section II.E of Appendix A for guidance on biodegradation testing of cement-stabilized wastes).

For polymeric or bitumen products, some visible culture growth from contamination, additives, or biodegradable components on the specimen surface that does not relate to overall substrate integrity may be present. For these cases, additional testing should be performed. If culture growth is observed upon completion of the biodegradation test for polymeric or bitumen products, the test specimens should be removed from the culture and washed free of all culture and growth with water, with only light scrubbing. An organic solvent compatible with the substrate may be used to extract surface contaminants. The specimen should be air dried at room temperature and the test repeated. Specimens should have observed culture growths rated no greater than 1 in the repeated ASTM G21 test. The specimens should have no observed growth in the repeated ASTM G22 test. Compression testing should be performed in accordance with ASTM C39 or ASTM D1074, as applicable, following the repeated G21 and G22 tests. The minimum acceptable compressive strength for bituminized waste forms is 60 psi. Maximum practical compressive strengths should be established for other media.

If growth is observed following the extraction procedure, longer term testing of at least six months should be performed to determine biodegradation rates. The Bartha-Pramer Method (R. Bartha and D. Pramer, "Features of a Flask and Method for Measuring the Persistence and Biological Effects of Pesticides in Soils," Soil Science 100 (1), pp. 68-70, 1965.) is acceptable for this testing. Soils used should be representative of those at burial grounds. Biodegradation extrapolated for full-size waste forms to 300 years should produce less than a 10 percent loss of the total carbon in the waste form.

3. Type A Packaging Test Requirements

Both Type A and Type B packages are subjected to testing defined for normal conditions of transport under §71.71, including, in sequence, the following tests:

- Water spray (5 cm/hr for 1 hr)
- Free drop (1.2 m for packages < 5000 kg)
- Stacking (5 x the package weight for 24 hours for packages < 5000 kg)
- Penetration (6 kg steel cylinder 3.2 cm in diameter from 1 m)

4. Spent Nuclear Fuel Storage Requirements

Spent Fuel Storage Casks are certified under 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-related Greater Than Class C Waste." In particular, §72.236, "Specific requirements for spent fuel storage cask approval and fabrication," provides requirements for storage casks. The requirements of §72.236 are given below.

The certificate holder and applicant for a Certificate of Compliance shall ensure that the requirements of this section are met.

(a) Specifications must be provided for the spent fuel to be stored in the spent fuel storage cask, such as, but not limited to, type of spent fuel (*i.e.,* BWR, PWR, both), maximum allowable enrichment of the fuel prior to any irradiation, burn-up (*i.e.,* megawatt-days/MTU), minimum acceptable cooling time of the spent fuel prior to storage in the spent fuel storage cask, maximum heat designed to be dissipated, maximum spent fuel loading limit, condition of the spent fuel (*i.e.,* intact assembly or consolidated fuel rods), the inerting atmosphere requirements.

(b) Design bases and design criteria must be provided for structures, systems, and components important to safety.

(c) The spent fuel storage cask must be designed and fabricated so that the spent fuel is maintained in a subcritical condition under credible conditions.

(d) Radiation shielding and confinement features must be provided sufficient to meet the requirements in §§72.104 and 72.106.

(e) The spent fuel storage cask must be designed to provide redundant sealing of confinement systems.

(f) The spent fuel storage cask must be designed to provide adequate heat removal capacity without active cooling systems.

(g) The spent fuel storage cask must be designed to store the spent fuel safely for a minimum of 20 years and permit maintenance as required.

(h) The spent fuel storage cask must be compatible with wet or dry spent fuel loading and unloading facilities.

(i) The spent fuel storage cask must be designed to facilitate decontamination to the extent practicable.

(j) The spent fuel storage cask must be inspected to ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce its confinement effectiveness.

(k) The spent fuel storage cask must be conspicuously and durably marked with—

(1) A model number;

(2) A unique identification number; and

(3) An empty weight.

(l) The spent fuel storage cask and its systems important to safety must be evaluated, by appropriate tests or by other means acceptable to the NRC, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

(m) To the extent practicable in the design of spent fuel storage casks, consideration should be given to compatibility with removal of the stored spent fuel from a reactor site, transportation, and ultimate disposition by the Department of Energy.

(n) Safeguards Information shall be protected against unauthorized disclosure in accordance with the requirements of §73.21 and the requirements of §73.22 or §73.23 of this chapter, as applicable.

5. Type B Packaging Test Requirements

Both Type A and Type B packages are subjected to testing defined for normal conditions of transport under §71.71, including, in sequence, the following tests:

- Water spray (5 cm/hr for 1 hr)
- Free drop (1.2 m for packages < 5000 kg)
- Stacking (5 x the package weight for 24 hours for packages < 5000 kg)
- Penetration (6 kg steel cylinder 3.2 cm in diameter from 1 m)

Type B packages are subject to additional tests specified in §71.73 for hypothetical accident conditions, again conducted in sequence and in the most damaging configuration, which include the following:

- Impact (9m drop onto an unyielding target, and 1 m drop onto 15 cm steel par on an unyielding target)
- Fire (fully engulfing 30 min fire with an 800°C minimum average temperature)
- Water immersion (15 m, and 0.9 m if fissile, and 200 m for larger quantities of radioactive material)

General: The general standards from §71.43 include:

Minimum Dimension	(a)	The smallest overall dimension of a package may not be less than 10 cm (4 in).
Tamper Indication	(b)	The outside of a package must incorporate a feature, such as a seal, that is not readily breakable and that, while intact, would be evidence that the package has not been opened by unauthorized persons.
Pressure Seal	(c)	Each package must include a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package.
Material Properties	(d)	A package must be made of materials and construction that assure that there will be no significant chemical, galvanic, or other reaction among the packaging components, among package contents, or between the packaging components and the package contents, including possible reaction resulting from inleakage of water, to the maximum credible extent. Account must be taken of the behavior of materials under irradiation.
Unauthorized operation	(e)	A package valve or other device, the failure of which would allow radioactive contents to escape, must be protected against unauthorized operation and, except for a pressure relief device, must be provided with an enclosure to retain any leakage.
Normal Transport	(f)	A package must be designed, constructed, and prepared for shipment so that under the tests specified in §71.71 ("Normal conditions of transport") there would be no loss or dispersal of radioactive contents, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging.
Surface Temperature	(g)	A package must be designed, constructed, and prepared for transport so that in still air at 38 °C (100 °F) and in the shade, no accessible surface of a package would have a temperature exceeding 50 °C (122 °F) in a nonexclusive use shipment, or 85 °C (185 °F) in an exclusive use shipment.
Venting	(h)	A package may not incorporate a feature intended to allow continuous venting during transport.

<u>Lifting and Tie-Down:</u> Lifting and tie-down standards in §71.45 include:

(a) Lifting — Any lifting attachment that is a structural part of a package must be designed with a minimum safety factor of three against yielding when used to lift the package in the intended manner, and it must be designed so that failure of any lifting device under excessive load would not impair the ability of the package to meet other requirements of this subpart. Any other structural part of the package that could be used to lift the package must be capable of being rendered inoperable for lifting the package during transport, or must be designed with strength equivalent to that required for lifting attachments.

(b) Tie-down devices:

(1) If there is a system of tie-down devices that is a structural part of the package, the system must be capable of withstanding, without generating stress in any material of the package in excess of its yield strength, a static force applied to the center of gravity of the package having a vertical component of 2 times the weight of the package with its contents, a horizontal component along the direction in which the vehicle travels of 10 times the weight of the package with its contents, and a horizontal component in the transverse direction of 5 times the weight of the package with its contents.

(2) Any other structural part of the package that could be used to tie down the package must be capable of being rendered inoperable for tying down the package during transport, or must be designed with strength equivalent to that required for tie-down devices.

(3) Each tie-down device that is a structural part of a package must be designed so that failure of the device under excessive load would not impair the ability of the package to meet other requirements of this part.

<u>External Radiation:</u> External radiation standards in §71.47 include:

(a) Except as provided in paragraph (b) of this section, each package of radioactive materials offered for transportation must be designed and prepared for shipment so that under conditions normally incident to transportation the radiation level does not exceed 2 mSv/h (200 mrem/h) at any point on the external surface of the package, and the transport index does not exceed 10.

(b) A package that exceeds the radiation level limits specified in paragraph (a) of this section must be transported by exclusive use shipment only, and the radiation levels for such shipment must not exceed the following during transportation:

(1) 2 mSv/h (200 mrem/h) on the external surface of the package, unless the following conditions are met, in which case the limit is 10 mSv/h (1000 mrem/h):

(i) The shipment is made in a closed transport vehicle;

(ii) The package is secured within the vehicle so that its position remains fixed during transportation; and

(iii) There are no loading or unloading operations between the beginning and end of the transportation;

(2) 2 mSv/h (200 mrem/h) at any point on the outer surface of the vehicle, including the top and underside of the vehicle; or in the case of a flat-bed style vehicle, at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the load or enclosure, if used, and on the lower external surface of the vehicle; and

(3) 0.1 mSv/h (10 mrem/h) at any point 2 meters (80 in) from the outer lateral surfaces of the vehicle (excluding the top and underside of the vehicle); or in the case of a flat-bed style vehicle, at any point 2 meters (6.6 feet) from the vertical planes projected by the outer edges of the vehicle (excluding the top and underside of the vehicle); and

(4) 0.02 mSv/h (2 mrem/h) in any normally occupied space, except that this provision does not apply to private carriers, if exposed personnel under their control wear radiation dosimetry devices in conformance with 10 CFR 20.1502.

(c) For shipments made under the provisions of paragraph (b) of this section, the shipper shall provide specific written instructions to the carrier for maintenance of the exclusive use shipment controls. The instructions must be included with the shipping paper information.

(d) The written instructions required for exclusive use shipments must be sufficient so that, when followed, they will cause the carrier to avoid actions that will unnecessarily delay delivery or unnecessarily result in increased radiation levels or radiation exposures to transport workers or members of the general public.

<u>Additional Type B Requirements:</u> The additional requirements for Type B packages specified in §71.51 include:

(a) A Type B package, in addition to satisfying the requirements of §§71.41 through 71.47, must be designed, constructed, and prepared for shipment so that under the tests specified in:

(1) Section 71.71 ("Normal conditions of transport"), there would be no loss or dispersal of radioactive contents—as demonstrated to a sensitivity of $10^{-6}A_2$ per hour, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging; and

(2) Section 71.73 ("Hypothetical accident conditions"), there would be no escape of krypton-85 exceeding 10 A_2 in 1 week, no escape of other radioactive material exceeding a total amount A_2 in 1 week, and no external radiation dose rate exceeding 10 mSv/h (1 rem/h) at 1 m (40 in) from the external surface of the package.

(b) Where mixtures of different radionuclides are present, the provisions of appendix A, paragraph IV of this part shall apply, except that for Krypton-85, an effective A_2 value equal to 10 A_2 may be used.

(c) Compliance with the permitted activity release limits of paragraph (a) of this section may not depend on filters or on a mechanical cooling system.

(d) For packages which contain radioactive contents with activity greater than 105 A_2, the requirements of §71.61 must be met.

<u>Normal Transport:</u> The general standards from §71.43(f) point to §71.71 "Normal conditions of transport."

(a) *Evaluation.* Evaluation of each package design under normal conditions of transport must include a determination of the effect on that design of the conditions and tests specified in this section. Separate specimens may be used for the free drop test, the compression test, and the penetration test, if each specimen is subjected to the water spray test before being subjected to any of the other tests.

(b) *Initial conditions.* With respect to the initial conditions for the tests in this section, the demonstration of compliance with the requirements of this part must be based on the ambient temperature preceding and following the tests remaining constant at that value between −29 °C (−20 °F) and +38 °C (+100 °F) which is most unfavorable for the feature under consideration. The initial internal pressure within the containment system must be considered to be the maximum normal operating pressure, unless a lower internal pressure consistent with the ambient temperature considered to precede and follow the tests is more unfavorable.

(c) *Conditions and tests —*

(1) *Heat.* An ambient temperature of 38 °C (100 °F) in still air, and insolation according to the following table:

Insolation Data

Form and location of surface	Total insolation for a 12-hour period (g cal/cm^2)
Flat surfaces transported horizontally:	
Base	None
Other surfaces	800
Flat surfaces not transported horizontally	200
Curved surfaces	400

(2) *Cold.* An ambient temperature of −40 °C (−40 °F) in still air and shade.

(3) *Reduced external pressure.* An external pressure of 25 kPa (3.5 lbf/in^2) absolute.

(4) *Increased external pressure.* An external pressure of 140 kPa (20 lbf/in^2) absolute.

(5) *Vibration.* Vibration normally incident to transport.

(6) *Water spray.* A water spray that simulates exposure to rainfall of approximately 5 cm/h (2 in/h) for at least 1 hour.

(7) *Free drop.* Between 1.5 and 2.5 hours after the conclusion of the water spray test, a free drop through the distance specified below onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected.

Criteria for Free Drop Test (Weight/Distance)

Package weight		Free drop distance	
Kilograms	**(Pounds)**	**Meters**	**(Feet)**
Less than 5,000	(Less than 11,000)	1.2	(4)
5,000 to 10,000	(11,000 to 22,000)	0.9	(3)
10,000 to 15,000	(22,000 to 33,100)	0.6	(2)
More than 15,000	(More than 33,100)	0.3	(1)

(8) *Corner drop.* A free drop onto each corner of the package in succession, or in the case of a cylindrical package onto each quarter of each rim, from a height of 0.3 m (1 ft) onto a flat, essentially unyielding, horizontal surface. This test applies only to fiberboard, wood, or fissile material rectangular packages not exceeding 50 kg (110 lbs) and fiberboard, wood, or fissile material cylindrical packages not exceeding 100 kg (220 lbs).

(9) *Compression.* For packages weighing up to 5000 kg (11,000 lbs), the package must be subjected, for a period of 24 hours, to a compressive load applied uniformly to the top and bottom of the package in the position in which the package would normally be transported. The compressive load must be the greater of the following:

(i) The equivalent of 5 times the weight of the package; or

(ii) The equivalent of 13 kPa (2 lbf/in^2) multiplied by the vertically projected area of the package.

(10)Penetration. Impact of the hemispherical end of a vertical steel cylinder of 3.2 cm (1.25 in) diameter and 6 kg (13 lbs) mass, dropped from a height of 1 m (40 in) onto the exposed surface of the package that is expected to be most vulnerable to puncture. The long axis of the cylinder must be perpendicular to the package surface.

Additional general requirements for fissile material packages are specified in §71.55 and §71.59. §71.55 contains the following:

(a) A package used for the shipment of fissile material must be designed and constructed in accordance with §§71.41 through 71.47. When required by the total amount of radioactive material, a package used for the shipment of fissile material must also be designed and constructed in accordance with §71.51.

(b) Except as provided in paragraph (c) or (g) of this section, a package used for the shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system, or liquid contents were to leak out of the containment system so that, under the following conditions, maximum reactivity of the fissile material would be attained:

(1) The most reactive credible configuration consistent with the chemical and physical form of the material;

(2) Moderation by water to the most reactive credible extent; and

(3) Close full reflection of the containment system by water on all sides, or such greater reflection of the containment system as may additionally be provided by the surrounding material of the packaging.

(c) The Commission may approve exceptions to the requirements of paragraph (b) of this section if the package incorporates special design features that ensure that no single packaging error would permit leakage, and if appropriate measures are taken before each shipment to ensure that the containment system does not leak.

(d) A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in §71.71 ("Normal conditions of transport")—

(1) The contents would be subcritical;

(2) The geometric form of the package contents would not be substantially altered;

(3) There would be no leakage of water into the containment system unless, in the evaluation of undamaged packages under §71.59(a)(1), it has been assumed that moderation is present to such an extent as to cause maximum reactivity consistent with the chemical and physical form of the material; and

(4) There will be no substantial reduction in the effectiveness of the packaging, including:

(i) No more than 5 percent reduction in the total effective volume of the packaging on which nuclear safety is assessed;

(ii) No more than 5 percent reduction in the effective spacing between the fissile contents and the outer surface of the packaging; and

(iii) No occurrence of an aperture in the outer surface of the packaging large enough to permit the entry of a 10 cm (4 in) cube.

(e) A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in §71.73 ("Hypothetical accident conditions"), the package would be subcritical. For this determination, it must be assumed that:

(1) The fissile material is in the most reactive credible configuration consistent with the damaged condition of the package and the chemical and physical form of the contents;

(2) Water moderation occurs to the most reactive credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents; and

(3) There is full reflection by water on all sides, as close as is consistent with the damaged condition of the package.

(f) For fissile material package designs to be transported by air:

(1) The package must be designed and constructed, and its contents limited so that it would be subcritical, assuming reflection by 20 cm (7.9 in) of water but no water inleakage, when subjected to sequential application of:

(i) The free drop test in §71.73(c)(1);

(ii) The crush test in §71.73(c)(2);

(iii) A puncture test, for packages of 250 kg or more, consisting of a free drop of the specimen through a distance of 3 m (120 in) in a position for which maximum damage is expected at the conclusion of the test sequence, onto the upper end of a solid, vertical, cylindrical, mild steel probe mounted on an essentially unyielding, horizontal surface. The probe must be 20 cm (7.9 in) in diameter, with the striking end forming the frustum of a right circular cone with the dimensions of 30 cm height, 2.5 cm top diameter, and a top edge rounded to a radius of not more than 6 mm (0.25 in). For packages less than 250 kg, the puncture test must be the same, except that a 250 kg probe must be dropped onto the specimen which must be placed on the surface; and

(iv) The thermal test in §71.73(c)(4), except that the duration of the test must be 60 minutes.

(2) The package must be designed and constructed, and its contents limited, so that it would be subcritical, assuming reflection by 20 cm (7.9 in) of water but no water inleakage, when subjected to an impact on an unyielding surface at a velocity of 90 m/s normal to the surface, at such orientation so as to result in maximum damage. A separate, undamaged specimen can be used for this evaluation.

(3) Allowance may not be made for the special design features in paragraph (c) of this section, unless water leakage into or out of void spaces is prevented following

application of the tests in paragraphs (f)(1) and (f)(2) of this section, and subsequent application of the immersion test in §71.73(c)(5).

(g) Packages containing uranium hexafluoride only are excepted from the requirements of paragraph (b) of this section provided that:

(1) Following the tests specified in §71.73 ("Hypothetical accident conditions"), there is no physical contact between the valve body

and any other component of the packaging, other than at its original point of attachment, and the valve remains leak tight;

(2) There is an adequate quality control in the manufacture, maintenance, and repair of packagings;

(3) Each package is tested to demonstrate closure before each shipment; and

(4) The uranium is enriched to not more than 5 weight percent uranium-235.

Fissile Material Arrays: § 71.59 establishes standards for arrays of fissile material packages.

(a) A fissile material package must be controlled by either the shipper or the carrier during transport to assure that an array of such packages remains subcritical. To enable this control, the designer of a fissile material package shall derive a number "N" based on all the following conditions being satisfied, assuming packages are stacked together in any arrangement and with close full reflection on all sides of the stack by water:

(1) Five times "N" undamaged packages with nothing between the packages would be subcritical;

(2) Two times "N" damaged packages, if each package were subjected to the tests specified in §71.73 ("Hypothetical accident conditions") would be subcritical with optimum interspersed hydrogenous moderation; and

(3) The value of "N" cannot be less than 0.5.

(b) The CSI must be determined by dividing the number 50 by the value of "N" derived using the procedures specified in paragraph (a) of this section. The value of the CSI may be zero provided that an unlimited number of packages are subcritical, such that the value of "N" is effectively equal to infinity under the procedures specified in paragraph (a) of this section. Any CSI greater than zero must be rounded up to the first decimal place.

(c) For a fissile material package which is assigned a CSI value—

(1) Less than or equal to 50, that package may be shipped by a carrier in a nonexclusive use conveyance, provided the sum of the CSIs is limited to less than or equal to 50.

(2) Less than or equal to 50, that package may be shipped by a carrier in an exclusive use conveyance, provided the sum of the CSIs is limited to less than or equal to 100.

(3) Greater than 50, that package must be shipped by a carrier in an exclusive use conveyance, provided the sum of the CSIs is limited to less than or equal to 100.

Hypothetical Accident Conditions: § 71.73 Hypothetical accident conditions include:

(a) *Test procedures.* Evaluation for hypothetical accident conditions is to be based on sequential application of the tests specified in this section, in the order indicated, to determine their cumulative effect on a package or array of packages. An undamaged specimen may be used for the water immersion tests specified in paragraph (c)(6) of this section.

(b) *Test conditions.* With respect to the initial conditions for the tests, except for the water immersion tests, to demonstrate compliance with the requirements of this part during testing, the ambient air temperature before and after the tests must remain constant at that value between −29 °C (−20 °F) and +38 °C (+100 °F) which is most unfavorable for the feature under consideration. The initial internal pressure within the containment system must be the maximum normal operating pressure, unless a lower internal pressure, consistent with the ambient temperature assumed to precede and follow the tests, is more unfavorable.

(c) *Tests.* Tests for hypothetical accident conditions must be conducted as follows:

(1) *Free drop.* A free drop of the specimen through a distance of 9 m (30 ft) onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected.

(2) *Crush.* Subjection of the specimen to a dynamic crush test by positioning the specimen on a flat, essentially unyielding horizontal surface so as to suffer maximum damage by the drop of a 500-kg (1100-lb) mass from 9 m (30 ft) onto the specimen. The mass must consist of a solid mild steel plate 1 m (40 in) by 1 m (40 in) and must fall in a horizontal attitude. The crush test is required only when the specimen has a mass not greater than 500 kg (1100 lb), an overall density not greater than 1000 kg/m^3 (62.4 lb/ft^3) based on external dimension, and radioactive contents greater than 1000 A_2 not as special form radioactive material. For packages containing fissile material, the radioactive contents greater than 1000 A_2 criterion does not apply.

(3) *Puncture.* A free drop of the specimen through a distance of 1 m (40 in) in a position for which maximum damage is expected, onto the upper end of a solid, vertical, cylindrical, mild steel bar mounted on an essentially unyielding, horizontal surface. The bar must be 15 cm (6 in) in diameter, with the top horizontal and its edge rounded to a radius of not more than 6 mm (0.25 in), and of a length as to cause maximum damage to the package, but not less than 20 cm (8 in) long. The long axis of the bar must be vertical.

(4) *Thermal.* Exposure of the specimen fully engulfed, except for a simple support system, in a hydrocarbon fuel/air fire of sufficient extent, and in sufficiently quiescent ambient conditions, to provide an average emissivity coefficient of at least 0.9, with an average flame temperature of at least 800 °C (1475 °F) for a period of 30 minutes, or any other thermal test that provides the equivalent total heat input to the package and which provides a time averaged environmental temperature of 800 °C. The fuel source must extend horizontally at least 1 m (40 in), but may not extend more than 3 m (10 ft), beyond any external surface of the specimen, and the specimen must be positioned 1 m (40 in) above the surface of the fuel source. For purposes of calculation, the surface absorptivity coefficient must be either that value which the package may be expected to possess if exposed to the fire specified or 0.8, whichever is greater; and the convective coefficient must be that value which may be demonstrated to exist if the package were exposed to the fire specified. Artificial cooling may not be applied after cessation of external heat input, and any combustion of materials of construction, must be allowed to proceed until it terminates naturally.

(5) *Immersion—fissile material.* For fissile material subject to §71.55, in those cases where water inleakage has not been assumed for criticality analysis, immersion under a head of water of at least 0.9 m (3 ft) in the attitude for which maximum leakage is expected.

(6) *Immersion—all packages.* A separate, undamaged specimen must be subjected to water pressure equivalent to immersion under a head of water of at least 15 m (50 ft). For test purposes, an external pressure of water of 150 kPa (21.7 lbf/in^2) gauge is considered to meet these conditions.

6. Department of Transportation (DOT) Packaging Requirements

10 CFR 173.410 General design requirements.

In addition to the requirements of subparts A and B of this part, each package used for the shipment of Class 7 (radioactive) materials must be designed so that—

(a) The package can be easily handled and properly secured in or on a conveyance during transport.

(b) Each lifting attachment that is a structural part of the package must be designed with a minimum safety factor of three against yielding when used to lift the package in the intended manner, and it must be designed so that failure of any lifting attachment under excessive load would not impair the ability of the package to meet other requirements of this subpart. Any other structural part of the package which could be used to lift the package must be capable of being rendered inoperable for lifting the package during transport or must be designed with strength equivalent to that required for lifting attachments.

(c) The external surface, as far as practicable, will be free from protruding features and will be easily decontaminated.

(d) The outer layer of packaging will avoid, as far as practicable, pockets or crevices where water might collect.

(e) Each feature that is added to the package will not reduce the safety of the package.

(f) The package will be capable of withstanding the effects of any acceleration, vibration or vibration resonance that may arise under normal conditions of transport without any deterioration in the effectiveness of the closing devices on the various receptacles or in the integrity of the package as a whole and without loosening or unintentionally releasing the nuts, bolts, or other securing devices even after repeated use (see §§173.24, 173.24a, and 173.24b).

(g) The materials of construction of the packaging and any components or structure will be physically and chemically compatible with each other and with the package contents. The behavior of the packaging and the package contents under irradiation will be taken into account.

(h) All valves through which the package contents could escape will be protected against unauthorized operation.

(i) For transport by air—

(1) The temperature of the accessible surfaces of the package will not exceed 50 °C (122 °F) at an ambient temperature of 38 °C (100 °F) with no account taken for insulation;

(2) The integrity of containment will not be impaired if the package is exposed to ambient temperatures ranging from −40 °C (−40 °F) to +55 °C (131 °F); and

(3) Packages containing liquid contents will be capable of withstanding, without leakage, an internal pressure that produces a pressure differential of not less than 95 kPa (13.8 lb/in^2).

§ 173.412 Additional design requirements for Type A packages.

In addition to meeting the general design requirements prescribed in §173.410, each Type A packaging must be designed so that—

(a) The outside of the packaging incorporates a feature, such as a seal, that is not readily breakable, and that, while intact, is evidence that the package has not been opened. In the case of packages shipped in closed transport vehicles in exclusive use, the cargo compartment, instead of the individual packages, may be sealed.

(b) The smallest external dimension of the package is not less than 10 cm (4 inches).

(c) Containment and shielding is maintained during transportation and storage in a temperature range of −40 °C (−40 °F) to 70 °C (158 °F). Special attention shall be given to liquid contents and to the potential degradation of the packaging materials within the temperature range.

(d) The packaging must include a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by pressure that may arise within the package during normal transport. Special form Class 7 (radioactive) material, as demonstrated in accordance with §173.469, may be considered as a component of the containment system. If the containment system forms a separate unit of the package, it must be securely closed by a positive fastening device that is independent of any other part of the package.

(e) For each component of the containment system account is taken, where applicable, of radiolytic decomposition of materials and the generation of gas by chemical reaction and radiolysis.

(f) The containment system will retain its radioactive contents under the reduction of ambient pressure to 25 kPa (3.6 psi).

(g) Each valve, other than a pressure relief device, is provided with an enclosure to retain any leakage.

(h) Any radiation shield that encloses a component of the packaging specified as part of the containment system will prevent the unintentional escape of that component from the shield.

(i) Failure of any tie-down attachment that is a structural part of the packaging, under both normal and accident conditions, must not impair the ability of the package to meet other requirements of this subpart.

(j) When evaluated against the performance requirements of this section and the tests specified in §173.465 or using any of the methods authorized by §173.461(a), the packaging will prevent—

(1) Loss or dispersal of the radioactive contents; and

(2) A significant increase in the radiation levels recorded or calculated at the external surfaces for the condition before the test.

(k) Each packaging designed for liquids will—

(1) Be designed to provide for ullage to accommodate variations in temperature of the contents, dynamic effects and filling dynamics;

(2) Meet the conditions prescribed in paragraph (j) of this section when subjected to the tests specified in §173.466 or evaluated against these tests by any of the methods authorized by §173.461(a); and

(3) Either—

(i) Have sufficient suitable absorbent material to absorb twice the volume of the liquid contents. The absorbent material must be compatible with the package contents and suitably positioned to contact the liquid in the event of leakage; or

(ii) Have a containment system composed of primary inner and secondary outer containment components designed to assure retention of the liquid contents within the secondary outer component in the event that the primary inner component leaks.

(l) Each package designed for gases, other than tritium not exceeding 40 TBq (1080Ci) or noble gases not exceeding the A_2 value appropriate for the noble gas, will be able to prevent loss or dispersal of contents when the package is subjected to the tests prescribed in §173.466 or evaluated against these tests by any of the methods authorized by §173.461(a).

§ 173.413 Requirements for Type B packages.

Except as provided in §173.416, each Type B(U) or Type B(M) package must be designed and constructed to meet the applicable requirements specified in 10 CFR part 71.

APPENDIX B. OVERVIEW OF RESEARCH REACTORS

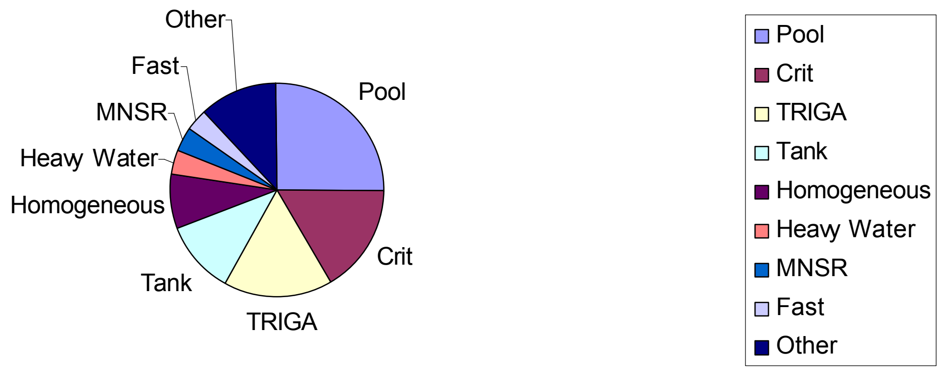
Many of the world's nuclear reactors are used for research and training, materials testing, production of radioisotopes for medicine and industry. These research reactors are much smaller than power reactors or those propelling ships, and many are on university campuses. Research reactors comprise a wide range of different reactor types, which are generally not used for power generation. The primary use of research reactors is to provide a neutron source for research and other applications. They are small relative to power reactors whose primary function is to produce electricity. Their power is designated in megawatts and their output ranges up to 100 MW_{th}, compared with 3000 MW_{th} (i.e., 1000 MW_e) for a typical power reactor. Research reactors are simpler than power reactors and operate at lower temperatures. They need far less fuel, and far less fission products build up as the fuel is used. (Reference: http://www-naweb.iaea.org/napc/physics/ACTIVITIES/Research_Reactors.htm.)

The IAEA Research Reactor Database (RRDB) online at http://nucleus.iaea.org/RRDB/Reports/CategoryList.aspx provides summary information and reports on the status of research reactors internationally. Of the 677 reactors shown in a recent search of the database, 249 are listed as currently in operation or in temporary shutdown. More than 60% of the listed reactors have been permanently shutdown or decommissioned.

Status	Developed Countries	Developing Countries	Subtotal
Operational	146	91	237
Shutdown	203	21	224
Decommissioned	177	16	193
Temporary shutdown	7	5	12
Cancelled	1	4	5
Planned	1	1	2
Under construction	2	1	3
Unverified information	0	1	1
Total	537	140	677

The 237 operational research reactors included in the RRDB are shown in the listing at the conclusion of this document and represent a wide variety of design and fuel types. Note that the total thermal power of all of these 237 reactors, 2400, MW_{th}, is less than the 3000 MW_{th} of a typical power reactor. A summary chart of the distribution of basic design types is shown in the following figure.

Operational Research Reactor Types



A summary report from the RRDB of the various fuel types is shown in the following table (Assumedly, the total of 245 includes the 237 operational as well as most of those that are in temporary shutdown.)

Fuel Type	#	%	Notes
MTR	65	26	Materials Test Reactor - pool type; Plate type fuel clad in Al
TRIGA	43	21	Training, Research, Isotopes, General Atomic - UZrH fuel rods clad in Al
VVR	10	4	Russian light water
IRT	7	3	Russian (uses EK-10 type rods)
EK-10	18	6	UO2 fuel rods clad in Al
ROD	16	7	fuel rods?
SUR-100	9	2	Manufactured by Siemens (Germany)
UO2	7	3	Uranium dioxide
MOX	3	1	Mixed oxide
Other	67	27	
TOTAL	245	100	

The following discussion of research reactor fuel storage is taken essentially verbatim from an article by Iain G. Ritchie, "GROWING DIMENSIONS - Spent Fuel Management at Research Reactors," available online at

http://www.iaea.org/Publications/Magazines/Bulletin/Bull401/article7.html.

A large variety of fuel types and fuel assembly geometries are in use in research and test reactors. Consequently, special storage conditions are often necessary, as well as different types of transport casks and different techniques for dealing with failed fuel. The distribution of fuel types among the reactors in the RRDB shows that a significant percentage (27%) are classified as "other" types. This underlines the fact that many experimental and exotic fuels exist at research reactors around the world, posing problems for their continued storage, transportation, and ultimate disposal.

By far the most commonly used form of spent fuel storage is the at-reactor pool, pond or basin. Since the average age of these facilities in the RRDB is 25 years, the success of wet storage where the water chemistry has been well controlled is remarkable. In fact, many aluminum clad Material Test Reactor fuels and aluminum pool liners show few, if any, signs of either localized or general corrosion after more than 30 years of exposure to research reactor water. In contrast, when water quality was allowed to degrade aluminum clad, fuel is seriously corroded.

Data also show that many facilities also have an auxiliary away-from-reactor pool or dry well. At away-from-reactor facilities, the trend is to transfer fuel from wet storage to dry storage, which avoids some of the expense of water treatment facilities and their maintenance.

Clearly, dry storage requires less monitoring and maintenance than wet storage and at most dry storage facilities the operators monitor the activity continuously. Several, however, are recognizing the importance of assessing the moisture content of dry storage facilities.

An IAEA survey also addressed the concerns expressed by reactor operators about their spent fuel management programs. Not surprisingly, the majority are concerned about the final disposal of their fuel. This is followed by concerns about limited storage capacity, and materials degradation. (Reference: http://www.iaea.org/Publications/Magazines/Bulletin/Bull401/article7.html.)

The World Nuclear Association provides summary descriptions of the more common research reactor designs at http://www.world-nuclear.org/info/inf61.html , including the information below.

There is a much wider array of designs in use for research reactors than for power reactors, where 80% of the world's plants are of just two similar types. They also have different operating modes, producing energy which may be steady or pulsed.

A common design (67 units) is the pool type reactor, where the core is a cluster of fuel elements sitting in a large pool of water. Among the fuel elements are control rods and empty channels for experimental materials. Each element comprises several (e.g. 18) curved aluminum-clad fuel plates in a vertical box. The water both moderates and cools the reactor, and graphite or beryllium is generally used for the reflector, although other materials may also be used. Apertures to access the neutron beams are set in the wall of the pool. Tank type research reactors (32 units) are similar, except that cooling is more active.

The TRIGA reactor is another common design (40 units). The core consists of 60-100 cylindrical fuel elements about 36 mm diameter with aluminum cladding enclosing a mixture of uranium fuel and zirconium hydride (as moderator). It sits in a pool of water and generally uses graphite or beryllium as a reflector. This kind of reactor can safely be pulsed to very high power levels (e.g. 25,000 MW) for fractions of a second. Its fuel gives the TRIGA a very strong negative temperature coefficient, and the rapid increase in power is quickly cut short by a negative reactivity effect of the hydride moderator.

Other designs are moderated by heavy water (12 units) or graphite. A few are fast reactors, which require no moderator and can use a mixture of uranium and plutonium as fuel. Homogenous type reactors have a core comprising a solution of uranium salts as a liquid, contained in a tank about 300 mm diameter. The simple design made them popular early on, but only five are now operating. (Reference: http://www.world-nuclear.org/info/inf61.html.)

Listing of Operational Research Reactors from the IAEA Research Reactor Database

The 237 operational research reactors listed in a recent search of the RRDB are shown in the following table.

Country	Facility Name	Type	Thermal Power(kW)
Algeria	NUR	POOL	1000
Algeria	ES-SALAM	HEAVY WATER	15000
Argentina	RA-1 ENRICO FERMI REACTOR	TANK	40
Argentina	RA-3	POOL	10000
Argentina	RA-0	TANK	0.01
Argentina	RA-4 (EX. SUR-100)	HOMOG (S)	0.001
Argentina	RA-6	POOL	500
Australia	OPAL	POOL	20000
Austria	TRIGA II VIENNA	TRIGA MARK II	250
Bangladesh	TRIGA MARK II	TRIGA MARK II	3000
Belarus	YALINA-Thermal	SUBCRIT	0
Belarus	YALINA-Booster	SUBCRIT	0
Belgium	BR-1	GRAPHITE	4000
Belgium	BR-2	TANK	100000
Brazil	IEA-R1	POOL	5000
Brazil	IPR-RI	TRIGA MARK I	100
Brazil	ARGONAUTA	ARGONAUT	0.2
Brazil	IPEN/MB-01	POOL	0.1
Canada	NRU	HEAVY WATER	135000
Canada	MNR MCMASTER UNIV	POOL, MTR	3000
Canada	ZED-2	TANK	0.2
Canada	SLOWPOKE-2, MONTREAL	SLOWPOKE-2	20
Canada	SLOWPOKE-2, HALIFAX	SLOWPOKE-2	20
Canada	SLOWPOKE, ALBERTA	SLOWPOKE-2	20
Canada	SRC SLOWPOKE, SASKATCHEWAN	SLOWPOKE-2	16
Canada	SLOWPOKE-2, RMC	SLOWPOKE-2	20
Chile	RECH-1	POOL	5000
China	HWRR-II	HEAVY WATER	15000
China	ZPR FAST	CRIT FAST	0.05
China	HFETR	TANK	125000
China	SPR IAE	POOL	3500
China	MNSR IAE	MNSR	27
China	PPR PULSING	POOL, UZRH	1000
China	HFETR CRITICAL	CRIT ASSEMBLY	0
China	SPRR-300	POOL	3000
China	NHR-5	HEATING PROT	5000
China	TSINGHUA UNIV.	POOL-2 CORES	1000
China	MJTR	POOL	5000
China	MNSR-SZ	MNSR	30
China	MNSR-SD	MNSR	33
China	MNSR-SH	MNSR	30
China	HTR-10	HIGH TEMP GAS	10000
China	CARR	TANK IN POOL	60000

Country	Facility Name	Type	Thermal Power(kW)
China	CFER	FAST BREEDER	65000
Colombia	IAN-R1	POOL	30
Czech Republic	LVR-15 REZ	TANK WWR	10000
Czech Republic	VR-1	POOL	5
Czech Republic	LR-0	POOL - VARIABLE CORE	5
Dem. P.R. of Korea	IRT-DPRK	POOL, IRT	8000
Egypt	ETRR-1	TANK WWR	2000
Egypt	ETRR-2	POOL	22000
Finland	FIR-1	TRIGA MARK II	250
France	MINERVE	POOL	0.1
France	EOLE	TANK IN POOL	0.1
France	OSIRIS	POOL	70000
France	ISIS	POOL	700
France	MASURCA	CRIT FAST	5
France	HFR	HEAVY WATER	58300
France	CABRI	POOL	25000
France	PHEBUS	POOL	38000
France	ORPHEE	POOL	14000
France	SILENE	HOMOG (L)	1
France	PILE AZUR	CRIT ASSEMBLY	0.1
Germany	FRG-1	POOL	5000
Germany	FRMZ	TRIGA MARK II	100
Germany	SUR STUTTGART	HOMOG (S)	0
Germany	SUR AACHEN	HOMOG (S)	0
Germany	SUR ULM	HOMOG (S)	0
Germany	BER-II	POOL	10000
Germany	SUR HANNOVER	HOMOG (S)	0
Germany	SUR FURTWANGEN	HOMOG (S)	0
Germany	AKR	HOMOG (S)	0.002
Germany	FRM II	POOL	20000
Ghana	GHARR-1	MNSR	30
Greece	GR-B SUBCRITICAL ASSEMBLY	CRIT ASSEMBLY	0
Hungary	NUCLEAR TRAINING REACTOR	POOL	100
Hungary	BUDAPEST RES. REACTOR	TANK WWR	10000
India	APSARA	POOL	1000
India	CIRUS	HEAVY WATER	40000
India	DHRUVA	HEAVY WATER	100000
India	FBTR	FAST BREEDER	40000
India	KAMINI	U-233 FUELLED	30
India	CRIT. FAC.-AHWR AND 500 MW PHWR	TANK	0.1
Indonesia	TRIGA MARK II, BANDUNG	TRIGA MARK II	2000
Indonesia	KARTINI-PTAPB	TRIGA MARK II	100
Indonesia	GA SIWABESSY MPR	POOL, MTR	30000
Iran	TRR	POOL	5000
Iran	ENTC LWSCR	SUBCRIT	0
Iran	ENTC GSCR	SUBCRIT	0

Country	Facility Name	Type	Thermal Power(kW)
Iran	ENTC HWZPR	CRIT ASSEMBLY	0.1
Iran	ENTC MNSR	MNSR	30
Israel	IRR-1	POOL	5000
Italy	LENA, TRIGA II PAVIA	TRIGA MARK II	250
Italy	TRIGA RC-1	TRIGA MARK II	1000
Italy	RSV TAPIRO	FAST SOURCE	5
Italy	AGN 201 COSTANZA	HOMOG (S)	0.02
Jamaica	UWI CNS SLOWPOKE	SLOWPOKE	20
Japan	UTR KINKI	ARGONAUT	0.001
Japan	TCA TANK TYPE CRIT. ASSBLY	CRIT ASSEMBLY	0.2
Japan	JRR-3M	POOL	20000
Japan	TOSHIBA NCA	CRIT ASSEMBLY	0.2
Japan	JRR-4	POOL	3500
Japan	FCA	CRIT FAST	2
Japan	JMTR	TANK	50000
Japan	YAYOI	TANK	2
Japan	KUCA	CRIT ASSEMBLY	0.1
Japan	NSRR	TRIGA ACPR	300
Japan	HTTR	HIGH TEMP GAS	30000
Japan	STACY	HOMOG	0.2
Japan	TRACY	PULSING	10
Kazakhstan	WWR-K ALMA ATA	POOL	6000
Kazakhstan	IGR	PULSING	
Kazakhstan	EWG 1	TANK	35000
South Korea	AGN-201K	HOMOG (S)	0.01
South Korea	HANARO	POOL	30000
Libya	IRT-1	POOL, IRT	10000
Malaysia	TRIGA PUSPATI (RTP)	TRIGA MARK II	1000
Mexico	TRIGA MARK III	TRIGA MARK III	1000
Mexico	CHICAGO MODELO 9000	SUBCRIT	0
Mexico	NUCLEAR CHICAGO MOD 2000	SUBCRIT	
Morocco	MA-R1	TRIGA MARK II	2000
Netherlands	LFR	ARGONAUT	30
Netherlands	HOR	POOL	2000
Netherlands	HFR	TANK IN POOL	45000
Nigeria	NIRR-0001	MNSR	30
Norway	HBWR	HEAVY WATER	20000
Norway	JEEP II	TANK	2000
Pakistan	PARR-1	POOL	10000
Pakistan	PARR-2	MNSR	30
Peru	RP-0	CRIT ASSEMBLY	0.001
Peru	RP-10	POOL	10000
Poland	MARIA	POOL	30000
Portugal	RPI	POOL	1000
Romania	TRIGA II PITESTI - SS CORE	TRIGA DUAL CORE	14000
Romania	TRIGA II PITESTI - PULSED	TRIGA DUAL CORE	500

Country	Facility Name	Type	Thermal Power(kW)
Russian Federation	OP-M	TANK WWR	300
Russian Federation	IR-8	POOL, IRT	8000
Russian Federation	IRT	POOL, IRT	2500
Russian Federation	WWR-M	TANK WWR	18000
Russian Federation	IVV-2M	POOL	15000
Russian Federation	MIR.M1	POOL/CHANNELS	100000
Russian Federation	IRT-T	POOL, IRT	6000
Russian Federation	GIDRA (HYDRA)	HOMOG (L)	10
Russian Federation	ARGUS	HOMOG (L)	20
Russian Federation	WWR-TS	TANK WWR	15000
Russian Federation	RBT-10/2	POOL	7000
Russian Federation	RBT- 6	POOL	6000
Russian Federation	F-1	GRAPHITE PILE	24
Russian Federation	SM-3	PRESSURE VESSEL	100000
Russian Federation	PIK PHYSICAL MODEL	CRIT ASSEMBLY	0.1
Russian Federation	BOR-60	FAST BREEDER	60000
Russian Federation	IGRIK, PULSED HOMOG	HOMOG	30
Russian Federation	YAGUAR (NHUAR)	HOMOG PUL	10
Russian Federation	FBR-L, FAST BURST-LASER	FAST BURST	5
Russian Federation	B-6	PROMPT BURST	20
Russian Federation	VK-50	BWR-PROTOTYPE	200000
Russian Federation	MAKET	CRIT ASSEMBLY	0.1
Russian Federation	FS-1M	CRIT ASSEMBLY	1
Russian Federation	AMBF-2	CRIT ASSEMBLY	0.1
Russian Federation	MATR-2	CRIT ASSEMBLY	0.4
Russian Federation	BFS-1	CRIT ASSEMBLY	0.2
Russian Federation	BFS-2	CRIT ASSEMBLY	1
Russian Federation	RBMK	CRIT ASSEMBLY	0.025
Russian Federation	SF-1	CRIT ASSEMBLY	0.1
Russian Federation	SF-7	CRIT ASSEMBLY	0.1
Russian Federation	KVANT	CRIT ASSEMBLY	1
Russian Federation	ASTRA	CRIT ASSEMBLY	0.1
Russian Federation	GROG	CRIT ASSEMBLY	0.1
Russian Federation	P	CRIT ASSEMBLY	0.2
Russian Federation	EMPHIR-2M	CRIT ASSEMBLY	0.1
Russian Federation	DELTA	CRIT ASSEMBLY	0.1
Russian Federation	SK PHYSICAL	CRIT ASSEMBLY	0.6
Russian Federation	NARTSISS-M	CRIT ASSEMBLY	0.01
Russian Federation	CA MIR.M1	CRIT ASSEMBLY	0.005
Russian Federation	CA-SM	CRIT ASSEMBLY	0.02
Russian Federation	STEND-4	CRIT ASSEMBLY	0.03
Russian Federation	STEND-5	CRIT ASSEMBLY	0.3
Russian Federation	659	CRIT ASSEMBLY	0.1
Russian Federation	1125	CRIT ASSEMBLY	0.6
Serbia	RB	HEAVY WATER	0
Slovenia	TRIGA- MARK II LJUBLJANA	TRIGA MARK II	250

Country	Facility Name	Type	Thermal Power(kW)
South Africa	SAFARI-1	TANK IN POOL	20000
Switzerland	CROCUS	CRIT ASSEMBLY	0.1
Switzerland	AGN 211 P	HOMOG (S)	2
Switzerland	PROTEUS	CRIT ASSEMBLY	1
Syria	SRR-1	MNSR	30
Taiwan, China	THOR	TRIGA CONV	2000
Thailand	TRR-1/M1	TRIGA MARK III	2000
Turkey	ITU-TRR, TECH UNIV	TRIGA MARK II	250
Ukraine	WWR-M KIEV	TANK WWR	10000
Ukraine	SNI, IR-100	POOL, IRT	200
Ukraine	SPh IR-100	CRIT ASSEMBLY	0.002
United Kingdom	NEPTUNE	POOL	0.1
USA	ARRR	TRIGA CONV	250
USA	NRAD	TRIGA MARK II	250
USA	AFRRI TRIGA	TRIGA MARK F	1000
USA	DOW TRIGA	TRIGA MARK I	300
USA	NTR GENERAL ELECTRIC	GRAPHITE	100
USA	ATR	TANK	250000
USA	ATRC	POOL	5
USA	AGN-201 IDAHO ST. UNIV.	HOMOG (S)	0.005
USA	KSU TRIGA MK II	TRIGA MARK II	250
USA	MITR-II MASS. INST. TECH.	TANK	5000
USA	NBSR	HEAVY WATER	20000
USA	PULSTAR N.C. STATE UNIV.	POOL, PULSTAR	1000
USA	HFIR	TANK	85000
USA	OSURR OHIO ST. UNIV.	POOL	500
USA	OSTR, OREGON STATE UNIV.	TRIGA MARK II	1100
USA	PSBR PENN ST. UNIV.	TRIGA MARK CONV	1000
USA	PUR-1 PURDUE UNIV.	POOL	1
USA	RRF REED COLLEGE	TRIGA MARK I	250
USA	RPI RENSSELAER	CRIT ASSEMBLY	0.1
USA	RINSC RHODE ISLAND NSC	POOL	2000
USA	ANN. CORE RES. REACTOR (ACRR)	TRIGA ACPR	4000
USA	SPR II	FAST BURST	5
USA	SPR III	FAST BURST	10
USA	AGN-201 TEXAS A&M UNIV.	HOMOG (S)	0.005
USA	NSCR TEXAS A&M UNIV.	TRIGA CONV	1000
USA	GSTR GEOLOGICAL SURVEY	TRIGA MARK I	1000
USA	UNIV. ARIZONA TRIGA	TRIGA MARK I	100
USA	UCI, IRVINE	TRIGA MARK I	250
USA	UFTR UNIV. FLORIDA	ARGONAUT	100
USA	UMLR UNIV. MASS. LOWELL	POOL	1000
USA	MUTR UNIV. MARYLAND	TRIGA MODIFIED	250
USA	MURR UNIV. OF MISSOURI	TANK IN POOL	10000
USA	UMRR	POOL, MTR	200
USA	AGN-201 UNIV. NEW MEXICO	HOMOG (S)	0.005

Country	Facility Name	Type	Thermal Power(kW)
USA	TRIGA UNIV. UTAH	TRIGA MARK I	100
USA	UWNR UNIV. WISCONSIN	TRIGA CONV	1000
USA	WSUR WASHINGTON ST. UNIV.	TRIGA CONV	1000
USA	WPI	POOL	10
USA	UC DAVIS	TRIGA MARK II	2000
USA	TRIGA II UNIV. TEXAS	TRIGA MARK II	1100
USA	FAST BURST (FBR)	FAST BURST	10000
Uzbekistan	WWR-SM TASHKENT	TANK WWR	10000
Uzbekistan	IIN-3M, FOTON	HOMOG PUL	20
Vietnam	DALAT RESEARCH REACTOR	POOL	500