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Spent Nuclear Fuel Integrity During Dry Storage—Performance Tests and Demonstrations

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A. L. Doherty

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Summary

This report summarizes the results of fuel integrity surveillance determined from gas sampling during and after performance tests and demonstrations conducted from 1983 through 1996 by or in cooperation with the U. S. Department of Energy (DOE) Office of Commercial Radioactive Waste Management (OCRWM). The cask performance tests were conducted at Idaho National Engineering Laboratory (INEL) between 1984 and 1991 and included visual observation and ultrasonic examination of the condition of the cladding, fuel rods, and fuel assembly hardware before dry storage and consolidation of fuel, and a qualitative determination of the effects of dry storage and fuel consolidation on fission gas release from the spent fuel rods. The performance tests consisted of 6 to 14 runs involving one or two loadings, usually three backfill environments (helium, nitrogen, and vacuum backfills), and one or two storage system orientations. The nitrogen and helium backfills were sampled and analyzed to detect leaking spent fuel rods. At the conclusion of each performance test, periodic gas sampling was conducted on each cask as part of the cask surveillance and monitoring activity. A spent fuel behavior project (i.e., enhanced surveillance, monitoring, and gas sampling activities) was initiated by DOE in 1994 for intact fuel in a CASTOR V/21 cask and for consolidated fuel in a VSC-17 cask. The results of the gas sampling activities are included in this report.

Information on spent fuel integrity is of interest in evaluating the impact of long-term dry storage on the behavior of spent fuel rods. Spent fuel used during cask performance tests at INEL offers significant opportunities for confirmation of the benign nature of long-term dry storage. Supporting cask demonstrations included licensing and operation of an independent spent fuel storage installation (ISFSI) at the Virginia Power (VP) Surry reactor site. A CASTOR V/21, an MC-10, and a Nuclear Assurance NAC-I28 have been loaded and placed at the VP ISFSI as part of the demonstration program.

Contents

Figures

Tables

Acronyms and Initialisms

BWR	boiling water reactor
CP&L	Carolina Power and Light
DOE	U.S. Department of Energy
DOE-HQ	U.S. Department of Energy-Headquarters
DOE-RL	U.S. Department of Energy-Richland Operations Office
DSC	dry shield canister
EPRI	Electric Power Research Institute
FRDS	fuel rod detection system
FTT	fuel temperature test
GE	General Electric Company
INEL	Idaho National Engineering Laboratory
LANL	Los Alamos National Laboratory
LCSS	lid closure and seal system
LITCO	Lockheed Martin Idaho Technology Company
LLNL	Lawrence Livermore National Laboratory
MSB	multi-assembly sealed basket
NUHOMS	NUTECH horizontal modular storage
NWPA	Nuclear Waste Policy Act of 1982
PNNL	Pacific Northwest National Laboratory
PSN	Pacific Sierra Nuclear Associates (now SNC)

PWR	pressurized water reactor
SCAP	solicitation for cooperative agreement proposal
SFS	spent fuel storage
SNC	Sierra Nuclear Corporation (formerly PSN)
TAN	Test Area North
TCs	thermocouples
UT	ultrasonic
VCC	ventilated concrete cask
VP	Virginia Power
WEPC	Wisconsin Electric Power Company

1.0 Introduction

The need for additional storage capacity for spent fuel from commercial nuclear power reactors is near for some utilities. The consequences of failure to provide additional storage capacity are significant if reactors are forced to terminate operations until required storage expansions can be provided. At-reactor storage capacity requirements can be expected to increase in the foreseeable future until spent fuel repositories are established. Therefore, a proven method of interim dry storage of spent fuel is needed in the near term to avoid reactor shutdowns and, in the long term, to provide contingency storage capability in the event that implementation of a repository is delayed.

The Nuclear Waste Policy Act of 1982 (NWPA) assigns the U.S. Department of Energy (DOE) the responsibility for assisting utilities with their spent fuel storage problems. In response to the NWPA, DOE Richland Operations Office (DOE-RL) issued a solicitation for cooperative agreement proposal (SCAP) to help the private sector with their spent fuel storage problems in May 1983, and proposals were received in August 1983. Virginia Power (VP) proposed that pressurized water reactor (PWR) spent fuel storage (SFS) cask performance testing be conducted at a federal site in support of their at-reactor licensed demonstration. The performance test was to be followed by a demonstration at the Surry reactor site. VP and DOE signed a cooperative agreement in March 1984, and VP signed a separate agreement with the Electric Power Research Institute (EPRI), essentially establishing a three-party cooperative agreement. Prior to the solicitation for cooperative agreements, DOE initiated performance testing of boiling water reactor (BWR) spent-fuel assemblies at the General Electric (GE) Morris facility in Illinois. Carolina Power and Light (CP&L) proposed a demonstration of the NUHOMS concept at their H. B. Robinson site. The cooperative agreement between DOE and CP&L was also signed in 1984.

The scope of the cooperative agreements included performance testing of three different metal storage casks loaded with unconsolidated spent nuclear fuel. The test were conducted at Idaho National Engineering Laboratory (INEL) with the Gesellschaft fur Nuklear Service (GNS) CASTOR V/21, the Transnuclear TN-24P, and the Westinghouse MC-10 casks. After the cask performance testing with unconsolidated fuel was completed in the VP/DOE cooperative program, a decision was made by DOE and EPRI to extend the performance testing to include dry rod consolidation and cask testing with the Transnuclear Inc. TN-24P cask.

At the conclusion of the metal cask testing, a cooperative agreement was established between DOE and Pacific Sierra Nuclear (PSN)[a] to test a ventilated concrete cask, the VSC-17, at INEL in 1990. The primary objective of PWR spent fuel storage cask performance testing was to obtain the heat transfer, shielding, and limited spent fuel integrity data needed to support at-reactor licensing efforts.

Prior to each dry storage system performance test, 1) dry (cold) runs were performed with a nonirradiated dummy fuel assembly to gain operating experience and finalize handling and test procedures; 2) the PWR spent fuel assemblies were ultrasonically examined and videotaped to ensure integrity; 3) the

(a) Pacific Sierra Nuclear Associates (PSN) is currently known as Sierra Nuclear Corporation (SNC).

exterior cask surface was instrumented with thermocouples (TCs) and radiation dose rate sensors; and 4) TCs were inserted into selected fuel assembly guide tubes after the cask was loaded with fuel assemblies to monitor temperatures throughout the test. The backfill environments, vacuum, nitrogen, and helium, were sampled and analyzed to detect leaking fuel rods. Where possible, vertical and horizontal orientations were investigated, and test runs were performed inside under controlled conditions. At the conclusion of testing, selected fuel assemblies were videotaped and photographed, and smear samples were collected and analyzed.

Participants in the various programs included GE, VP, SNC, EPRI, CP&L, Wisconsin Electric Power Company (WEPC), Transnuclear, Inc., Lawrence Livermore National Laboratory (LLNL), EG&G Idaho Inc., the Pacific Northwest National Laboratory (PNNL), and Los Alamos National Laboratory (LANL). Spent fuel storage (SFS) systems included in the performance testing included a Ridihalgh, Eggers & Associates REA-2023 cask (currently available from Mitsubishi of Japan as an MSF-IV), a GNS CASTOR-V/21 cask, a Transnuclear, Inc. TN-24P cask, a Westinghouse MC-10 cask, a NUTECH horizontal modular storage system (NUHOMS), and an SNC ventilated vertical concrete storage cask.

2.0 Conclusions

Dry storage systems can be satisfactorily handled in many reactor facilities with only minor modifications to the supplied handling equipment and procedures. Performance testing of CASTOR-V/21, TN-24P, MC-10, and VSC-17 PWR SFS casks was successfully completed at the INEL Test Area North (TAN). An REA-2023 BWR SFS cask was performance tested at GE facilities in Morris, Illinois. A similar performance test of NUHOMS was conducted at the H. B. Robinson reactor site in Florida.

The tests demonstrated that the storage systems could be satisfactorily handled and loaded dry. They also demonstrated the heat transfer and shielding performance of the system when loaded with intact or consolidated PWR spent fuel. Radiochemical gamma analysis of gas samples from cask performance tests and subsequent cask surveillance and monitoring activities provide an indication for determination of spent fuel integrity during dry storage. The gas sampling analysis indicates that dry storage of spent fuel in an inert atmosphere is benign. In general, fuel handling activities have a more significant impact on fuel rods than does extended dry storage in an inert atmosphere.

The following are significant findings and conclusions:

Fuel Characterization and Integrity

- Results of pretest in-basin sipping of each Cooper spent fuel assembly indicated that no failed fuel was loaded in the REA-2023 cask. Results of pretest and in-test fuel integrity activities (pretest ultrasonic, photographic, and video examinations) led to the conclusion that no failed fuel rods were loaded in the CASTOR-V/21, TN-24P, or MC-10 casks.

- Pre- and post-test inspections of selected assemblies revealed that no noticeable changes occurred during the testing. Expected uneven rod growth during irradiations and slight rod bowing was observed in some fuel assemblies during video scans and photography before and after cask performance testing. Video scans and photography indicated the presence of an intermittent "crud" layer on the fuel assemblies.

- Only two leaking fuel rods have been detected during cask performance testing with unconsolidated fuel, one in the REA-2023 cask and one in the TN-24P cask. Several (about 10) fuel rods began to leak after the fuel was consolidated. No leaking fuel rods were detected during the consolidation process at INEL. The leaks were detected through determination of the concentration of krypton-85 in gas samples. The cladding penetrations were estimated to be extremely small, and they had no adverse effects on the testing effort.

- Continued post-test gas sampling of casks containing unconsolidated and consolidated fuel are recommended at INEL to determine the long-term impact of consolidation on fuel integrity.

- Smear samples from selected fuel assemblies indicated that large quantities of cobalt-60 were present, but no fission product species were detected.

3.0 Dry Storage Performance Tests

This section of the report contains a description of the dry spent nuclear fuel storage systems followed by a summary of the thermal performance of each system. The thermal performance data included indicate the temperature variation that was experienced during performance testing and provide an indication of the initial storage temperature at the beginning of long-term storage.

3.1 Cask Descriptions

Six spent nuclear fuel dry storage systems are briefly described in this section. Additional details on each of the systems can be found in the cited references.

3.1.1 REA-2023 Cask

The REA-2023 (McKinnon et al. 1986; Wiles et al. 1986) spent fuel storage cask consists of a double containment design with silicone rubber O-rings for sealing the primary lid of the inner cavity and a welded final closure on the secondary cover. The REA-2023 cask, shown in Figure 3.1, has a smooth, painted, stainless steel outer skin; a lead/stainless steel gamma shield; and a water/glycol neutron shield. The fuel basket is constructed of stainless steel for criticality control, copper plates to conduct heat to the cask wall, and stainless steel for structural strength. The loaded cask is approximately 4.9 m (16 ft) tall, measures 2.22 m (7.3 ft) in diameter, and weighs approximately 100 tons. The basket is configured to hold 52 BWR spent fuel assemblies. The test fuel assemblies were of the GE 7 x 7 rod design. The REA-2023 BWR spent fuel storage cask design and manufacturing rights have been acquired by Mitsubishi of Japan, and the cask model designation has been changed to MSF IV. The cask was performance tested at GE-Morris (Illinois) and is located at INEL.

3.1.2 CASTOR-V/21 Cask

The Castor-V/21 cask (Creer et al. 1986) shown in Figure 3.2 is a one-piece cylindrical structure composed of ductile cast iron in nodular graphite form. This material exhibits good strength and ductility and provides effective gamma shielding. The overall external dimensions of the cask body include a height of 4.9 m (16 ft) and a diameter of 2.4 m (8 ft). The external surface has 73 heat transfer fins that circumvent the cask; it is coated with epoxy paint for corrosion protection and ease of decontamination.

The spent fuel basket is a cylindrical structure composed of welded stainless steel plate and borated stainless steel plate. The basket comprises an array of 21square fuel tubes/channels that provide structural support and positive positioning of the fuel assemblies. Stainless-steel primary and secondary lids are provided. The test fuel assemblies were of the Westinghouse 15 x 15 rod design.

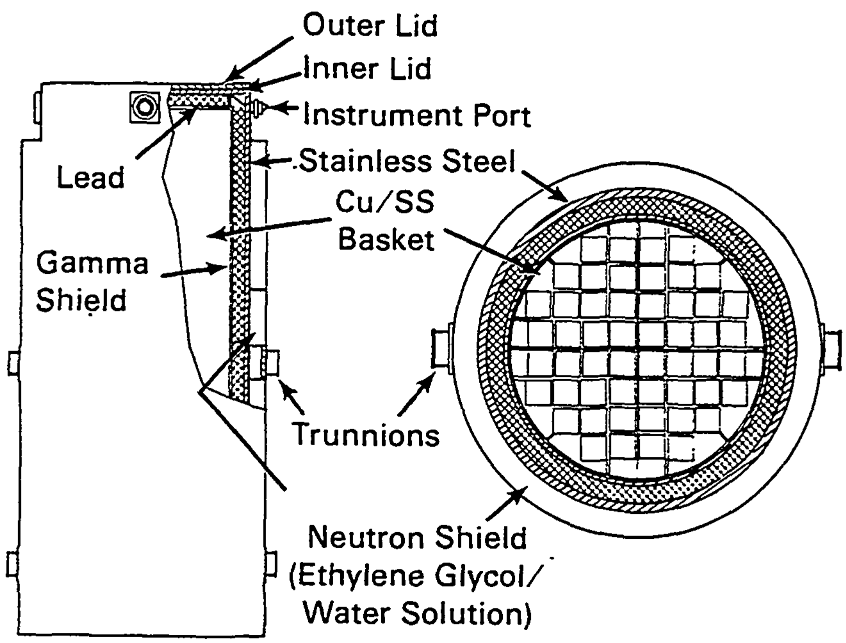

Figure 3.1. REA-2023 BWR Cask

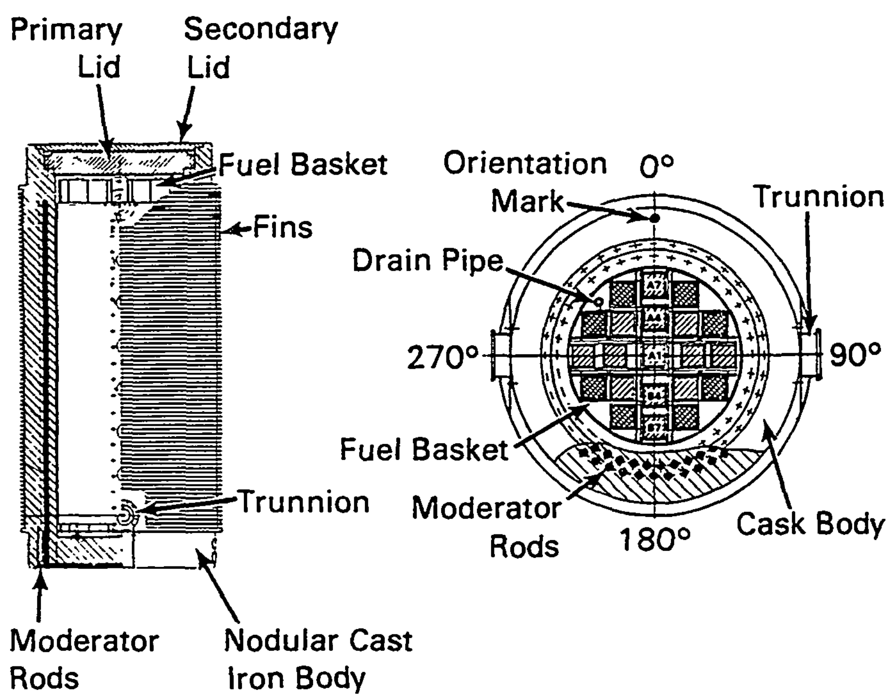

Figure 3.2. CASTOR V/21 PWR Cask

The secondary lid was not used during the CASTOR-V/21 performance test because it interfered with fuel assembly instrumentation leads. This cask was performance tested at INEL and demonstrated as part of the VP/DOE cooperative agreement at VP's Surry Reactor.

3.1.3 TN-24P Cask

The TN-24P (McKinnon et al. 1987a) cask has a forged steel body for structural integrity and gamma shielding, surrounded by a resin layer for neutron shielding, and is enclosed in a smooth steel outer shell. The TN-24P cask, shown in Figure 3.3, is 5.0 m (16 ft) long and measures 2.3 m (7.5 ft) in diameter; it weighs approximately 100 tons when loaded with unconsolidated PWR spent fuel. The cask has a cylindrical cavity that holds a fuel basket designed to accommodate 24 intact or consolidated PWR fuel assemblies. The basket is made of a neutron-absorbing material, borated aluminum, to control criticality. The cavity atmosphere is designed to be nitrogen or helium at a positive pressure.

The cask is sealed with a single lid with double, metallic O-ring seals. A protective cover, bolted to the body, provides weather protection for the lid penetrations. The test fuel assemblies are of the standard Westinghouse 15 x 15 rod design. This cask was performance tested at INEL with intact PWR fuel as part of the VP/DOE cooperative agreement. It was later performance tested with consolidated fuel by DOE and EPRI.

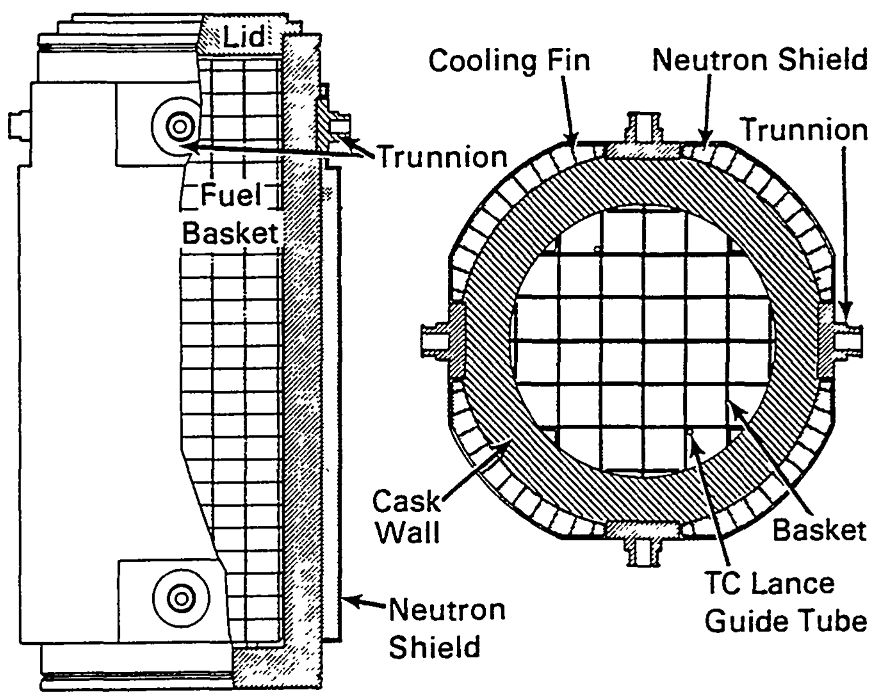

Figure 3.3. TN-24P PWR Cask

3.1.4 MC-10 Cask

The MC-10 PWR spent fuel storage cask (McKinnon et al. 1987b) consists of a low-alloy forged steel body. The MC-10 cask, shown in Figure 3.4, is 4.8 m (15.7 ft) long and 2.7 m (8.9 ft) in diameter; it weighs approximately 110 tons when loaded with unconsolidated PWR spent fuel. Neutron shielding has been placed around the outside of the cask and vertical carbon-steel heat transfer fins pass through the neutron shield to augment cooling. The fuel basket within the cask is constructed of aluminum and is configured to hold 24 PWR spent fuel assemblies or 24 consolidated fuel canisters. Each of the 24 basket locations contains a removable stainless steel enclosure and neutron poison material for criticality control. The Surry spent fuel assemblies used during testing were of a standard Westinghouse 15 x 15 rod design. The cask is closed with two lids and a seal cover having both elastomer and metallic O-rings to seal the cask cavity from the environment.

The cask lid closure and seal system (LCSS) consists of four covers: shield, primary, seal, and protective covers. The LCSS was replaced with a single test lid for the performance test. This cask was tested at INEL and demonstrated at VP's Surry reactor as part of the VP/DOE cooperative agreement.

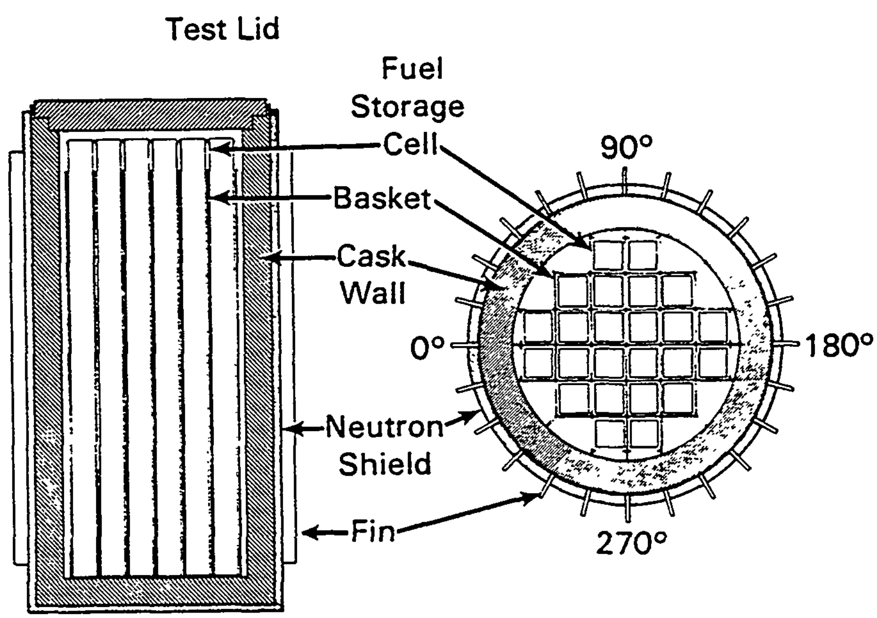

Figure 3.4. Westinghouse MC-10 PWR Cask

3.1.5 VSC-17 Cask

The VSC-17 spent fuel storage system (McKinnon et al. 1992) is a passive device for vertical storage of 17 assemblies/canisters of irradiated nuclear fuel. The VSC-17 cask is shown in Figure 3.5. The commercial version of the cask is designed to hold 24 PWR fuel assemblies. The VSC-17 system consists of a ventilated concrete cask (VCC) and a multi-assembly sealed basket (MSB). Decay heat generated by the spent fuel is transmitted through the containment wall of the MSB to a cooling air flow. Natural circulation drives the cooling air flow through an annular path between the MSB and the VCC and carries the heat to the environment without undue heating of the concrete cask. The annular air flow cools the outside of the MSB and the inside of the VCC.

The cask weighs approximately 80 tons empty and 110 tons loaded with 17 canisters of consolidated fuel. The VCC has a reinforced concrete body with an inner steel liner and a weather cover (lid). The MSB contains a guide sleeve assembly for fuel support and a composite shield lid that seals the stored fuel inside the MSB.

The cavity atmosphere is helium at slightly sub-atmospheric pressure. The helium atmosphere inside the MSB enhances the overall heat transfer capability and prevents oxidation of the fuel and corrosion of the basket components.

Consolidated Westinghouse 15x15 PWR spent fuel assemblies were used in the performance test. The performance test was part of a DOE/PSN cooperative agreement

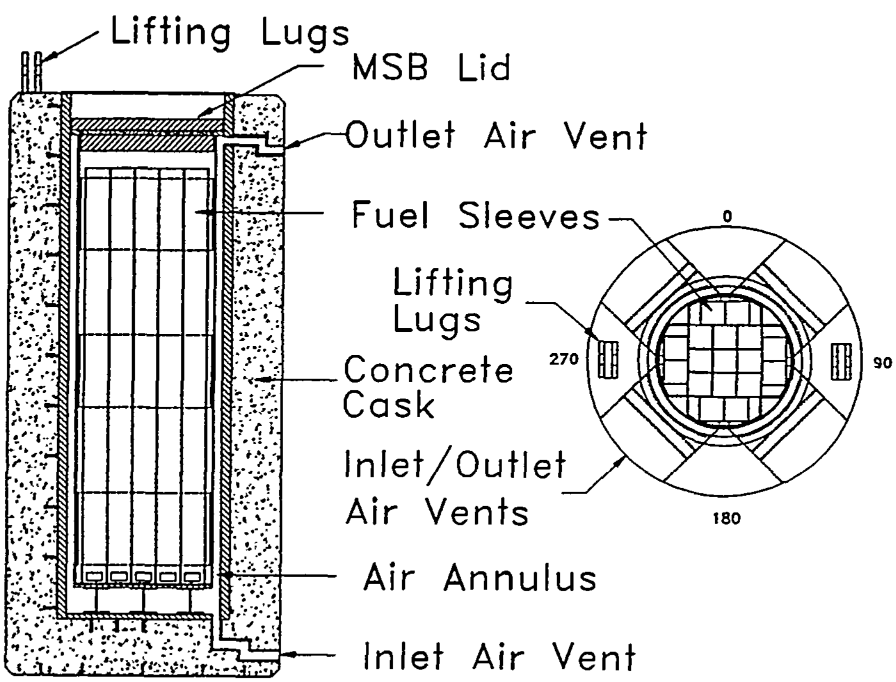

Figure 3.5. VSC-17 Cask

3.1.6 NUHOMS Cask System

The NUHOMS system (Strope et al. 1990) is a passive device for horizontal dry storage of irradiated fuel assemblies. In the NUHOMS system, which is shown in Figure 3.6, the fuel assemblies are confined in a helium atmosphere by a stainless-steel canister. The dry shield canister (DSC) is protected and shielded by a massive reinforced concrete module, which is roughly 6.7 m (22 ft) long, 7.6 m (25 ft) wide, and 3.7 m (12 ft) high. The walls and roof of the module are approximately 1.7 m (3.5 ft) thick, providing the primary biological shield and impact protection for the canister. The generic NUHOMS system consists of a basic unit of two modules arranged back to back. The system is expanded with additional two-module units placed beside the first until the required storage capacity is reached.

The demonstrated seven-assembly DSC consists of a cylindrical shell made of rolled 1.5-cm (0.6-in.)-thick stainless steel. It is 94 cm (37 in.) in diameter and 457 cm (180 in.) long. The internal basket contains seven square fuel tubes made of a boron/aluminum alloy with stainless steel cladding. Commercial systems contain 24 fuel tubes per DSC. The test fuel assemblies were of the standard 15 x 15 rod PWR design. The performance demonstration was conducted as part of a CP&L/DOE cooperative agreement.

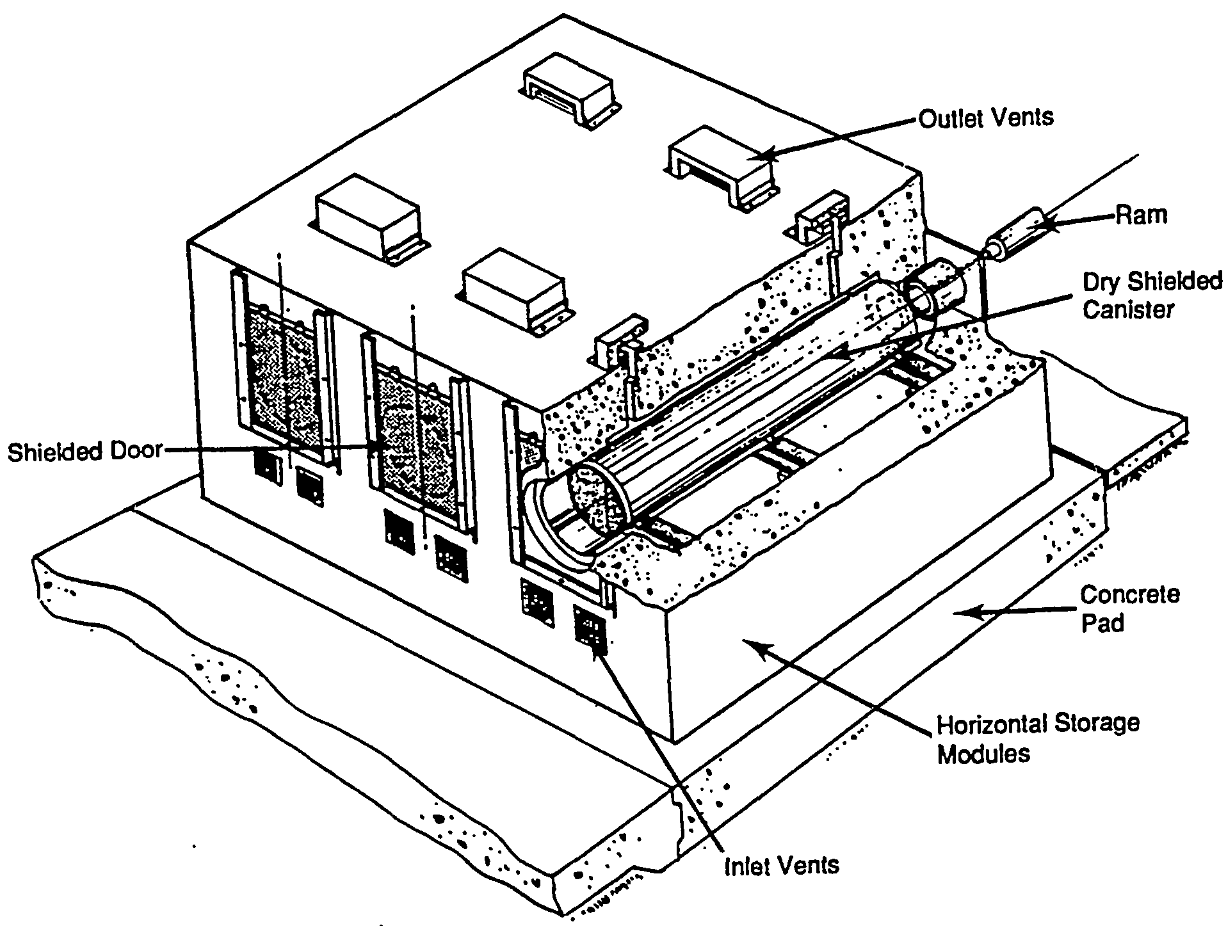

Figure 3.6. H. B. Robinson NUHOMS Dry Storage System

3.2 Thermal Performance

Cask thermal performance tests consisted of 6 to 14 conditions involving one or two loadings, usually three backfill environments, and one or two cask orientations. A test plan specified the order of the runs, spent fuel assembly load patterns, temperature measurement locations, and calibration requirements.

Cask thermal instrumentation consisted of 71 to 106 thermocouples located in the fuel, the cask basket, or the surface of the cask or concrete structure. All of the casks included one or two pressure monitors. The REA cask also included a weather station.

Figure 3.7 provides generic temperature profiles associated with cask orientation and fill gas. A summary of the core test conditions and peak fuel temperatures is found in Table 3.1. Axial and radial temperature profiles corresponding to the location of the peak temperature are contained in the original reports (Creer et al. 1986; McKinnon et al. 1986, 1987a, 1987b; Wiles et al. 1986; Strope et al. 1990; McKinnon and DeLoach 1993).

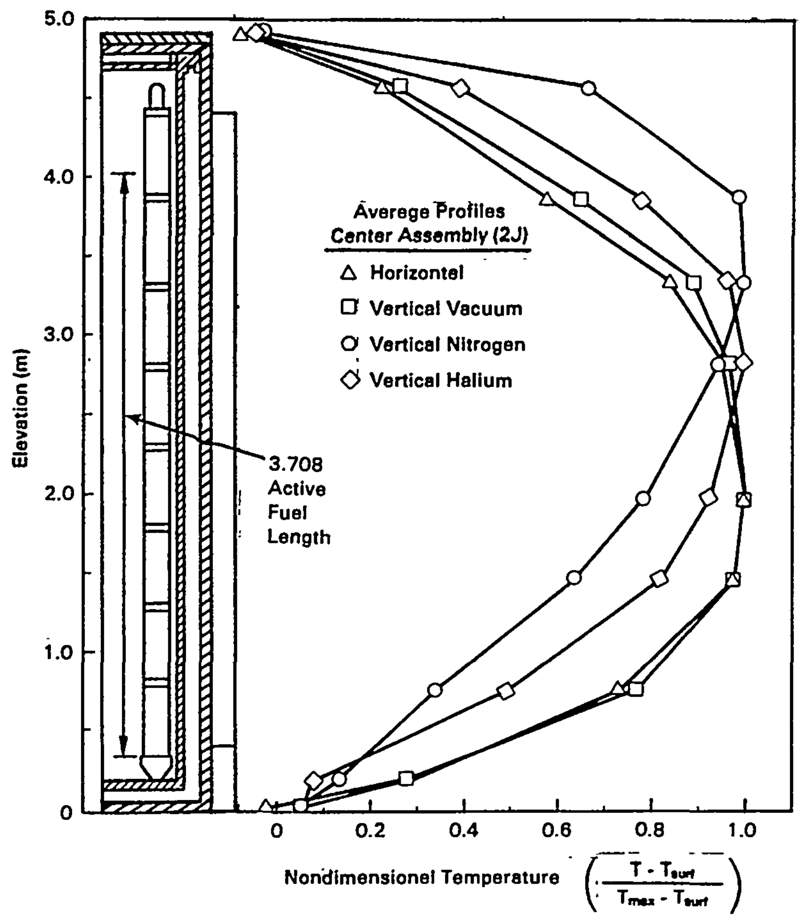

Figure 3.7. Effects of Cask Orientation and Fill Gas on Cask Axial Temperature Profiles

Table 3.1. Peak Temperatures for Fully Loaded Storage System

Metal Casks	Orientation	Backfill	Cask Heat Load, kW Design	Cask Heat Load, kW Actual	Ambient Temperature, °C	Wind Speed, m/s	Estimated Peak Clad Temperature, °C
REA-2023	Vertical	Helium	21	14.6	22		144
52 BWR	Vertical	Helium		14.9	-14	0.7	110
Assemblies	Vertical	Nitrogen		15.1	-4	3.1	151
(hot fuel in	Vertical	Vacuum		15.2	24		227
center of	Vertical	Vacuum		15.2	-10	3.6	200
basket)	Horizontal	Helium		14.8	-8	4.2	113
	Horizontal	Nitrogen		15.0	-4	2.0	164
CASTOR V/21	Vertical	Helium	21	28.4	27		352
21 PWR	Vertical	Nitrogen		28.4	24		368
Assemblies	Vertical	Vacuum		28.4	25		424
(hot fuel on	Horizontal	Helium		28.4	24		365
outside of	Horizontal	Nitrogen		28.4	24		405
basket)							
TN-24P	Vertical	Helium	24	20.6	18		221
24 PWR	Vertical	Nitrogen		20.6	20		241
Assemblies	Vertical	Vacuum		20.6	20		290
(hot fuel in	Horizontal	Helium		20.5	18		215
center of	Horizontal	Nitrogen		20.4	21		256
basket)	Horizontal	Vacuum		20.3	19		280
24 PWR	Vertical	Helium	24	23.3	22		211
Canisters[a]	Vertical	Nitrogen		23.3	16		268
(hot fuel on	Vertical	Vacuum		23.2	22		293
outside of	Horizontal	Helium		23.2	17		205
basket)	Horizontal	Nitrogen		23.2	22		252
	Horizontal	Vacuum		23.1	23		282
	Horizontal	Vacuum		23.1	24		282[b]
MC-10	Vertical	Helium	15	12.7	28		139
24 PWR	Vertical	Nitrogen		12.7	24		181
Assemblies	Vertical	Vacuum		12.6	23		217
(hot fuel on	Horizontal	Helium		12.6	25		138
outside of	Horizontal	Nitrogen		12.6	26		204
basket)	Horizontal	Vacuum		12.6	27		213
Concrete Systems						Heat Source	
NUHOMS	Horizontal	Helium	7	5.3	21	7 PWR fuel	181
(random load	Horizontal	Helium	7	7	23	Electricity	201
of hot fuel)	Horizontal	Helium	7	13	19	Electricity	333
Block Inlets	Horizontal	Helium	7	7	28	Electricity	321
						Vent Blockage	
VSC-17	Vertical	Helium	17	14.9	21	None	321
17 PWR	Vertical	Nitrogen		14.9	24	None	376
Canisters[a]	Vertical	Vacuum		14.9	24	None	397
(hot fuel in	Vertical	Helium		14.9	23	½ inlets	334
center of	Vertical	Helium		14.9	23	All inlets	378
basket)	Vertical	Helium		14.9	22	All vents	381

(a) Consolidated fuel - 2:1 consolidation ratio.
(b) The top and bottom of the cask were insulated during this run.

4.0 Spent Fuel

Three types of spent fuel have been used during the cask performance testing and demonstration program. BWR 7 x 7 spent fuel assemblies were used for the performance test of the REA-2023 cask. Westinghouse 15 x 15 PWR was used in the CASTOR V/21, TN-24P, MC-10, and NUHOMS performance tests. A portion of this fuel was consolidated at INEL and used in performance tests of the TN-24P and VSC-17 casks, also at INEL. Table 4.1 gives a summary of the fuel used in each of the performance tests. A description of the spent fuel and the results of spent fuel monitoring during the performance test follows.

4.1 BWR 7 x 7 Fuel Assemblies

The BWR assemblies used during the REA-2023 cask performance test were of a GE 7 x 7 design and were taken from the Nebraska Power Cooper reactor. The 7 x 7 design specifications are given in Table 4.2. The upper and lower tie plates are 304 stainless-steel castings. The lower tie plate has a nose piece that supports the fuel assembly in the reactor. The upper tie plate has a handle for transferring the fuel assembly.

Table 4.1 Spent Fuel Assembly Characteristics

Cask	REA-2023	CASTOR V/21	TN-24P	TN-24P[a]	MC-10	VSC-17[a]	NUHOMS
Fuel Type	BWR	PWR	PWR	PWR	PWR	PWR	PWR
Assembly Type	7 x 7	15 x 15	15 x 15	Consolidated 15 x 15	15 x 15	Consolidated 15 x 15	15 x 15
Burnup, GWd/MTU	24-28	24-35	29-32	24-35	24-35	26-35	31-34
Cooling Time, years	2.3-3.4	2.2-3.8	4.2	6.2-12.2	4.6-10.1	8.8-14.3	5
Discharge Date(s)	1981 -82	1981 -83	1981	1975 - 81	1975 - 81	1976 - 81	1984
Enrichment, wt%	2.5	2.9-3.1	2.9-3.2	1.9-3.2	1.9-3.2	2.56-3.2	2.9
Assembly Decay Heat, W	235-370	1000-1800	832-919	701-1185	400-700	700-1050	692-834
Average, W	290	1350	860	970	530	877	766
Cask, kW	15.2	28.4	20.6	23.3	12.6	14.9	5.3

(a) Performance test using consolidated fuel in the cask.

Table 4.2 Design Characteristics of Cooper BWR Fuel Rods and Assemblies

Fuel Assembly Data	
Overall length	4.47 m (175.83 in.)
Nominal active fuel length	3.71 m (144 in.)
Fuel rod pitch	1.87 cm (0.738 in.)
Space between fuel rods	0.445 cm (0.175 in.)
Fuel bundle heat transfer area	8.04 m^2 (86.52 ft2)
Fuel rod array, 7 x 7	-- --
Zr-2 weight	48.000 kg/ass. (105.8 lb/ass.)
304 stainless steel	8.600 kg/ass. (18.96 lb/ass.)
Fuel Rod Data	
Average linear rod power	23.2 kW/m (7.079 kW/ft)
Outside diameter	1.43 cm (0.563 in.)
Cladding thickness	0.081 cm (0.032 in.)
Pellet outside diameter	1.24 cm (0.487 in.)
Fission gas plenum length	40.6 cm (16 in.)
Pellet immersion density	10.42 g/cc
Cladding material	zircaloy-2
Helium fill gas pressure	1 atm
Fuel	UO_2

In addition to having standard BWR fuel rods, each assembly has eight tie rods that thread into the lower tie plate casting. The upper ends of the tie rods extend through and are fastened to the upper tie plate with stainless steel hexagonal nuts and locking tabs. These tie rods support the weight of the assembly during fuel-handling operations when the assembly hangs by the handle. The center rod of each fuel assembly has been designed to maintain the position of the fuel rod spacers. It is inserted into the fuel assembly and rotated to lock the spacers into their respective locations. The spacers have Inconel-X springs to maintain rod-to-rod spacing. The fuel rods are pressurized with helium and sealed by welding end plugs on each end.

Additional information on the Cooper fuel assemblies is contained in the REA-2023 performance report (McKinnon et al. 1986). Information in that report includes burnup and burnup history. Each assembly was initially enriched to 2.5 wt% uranium-235 averaged over all rods in the assembly.

4.2 PWR 15 x 15 Fuel Assemblies

The intact PWR assemblies used in the performance tests conducted at INEL were taken from the VP's Surry reactors. They are Westinghouse 15 x 15 PWR fuel assemblies, which are square in cross-section, nominally 214 mm (8.426 in.) on a side, and have a total length of 4058 mm (159.765 in.). The fuel column is 3658-mm (144-in.) long. The overall configuration is shown in Figure 4.2.

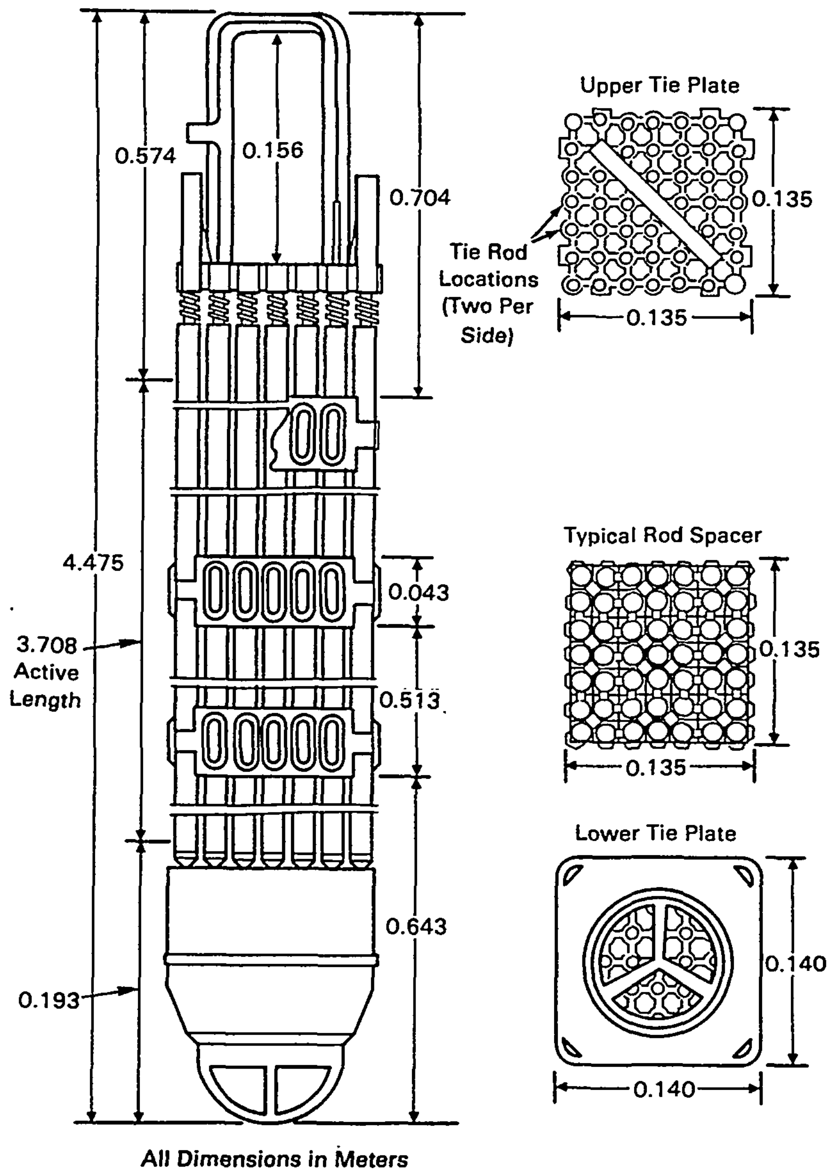

Figure 4.1. Cooper Spent Fuel Assembly

The fuel rods in a fuel assembly are arranged in a square array with 15 rod locations per side and a nominal rod-to-rod centerline pitch of 14.3 mm (0.56 in.), as shown in Figure 4.3. Of the total possible rod locations per assembly (225), 20 were occupied by guide tubes for the control rods and burnable poison rods, and one central thimble was reserved for in-core instrumentation. The remaining 204 locations contained fuel rods. In addition, a fuel assembly included a top nozzle, a bottom nozzle, and seven grid assemblies. The guide tubes, central thimble, grid assemblies, and the top and bottom nozzles provide the basic structure for the fuel assembly.

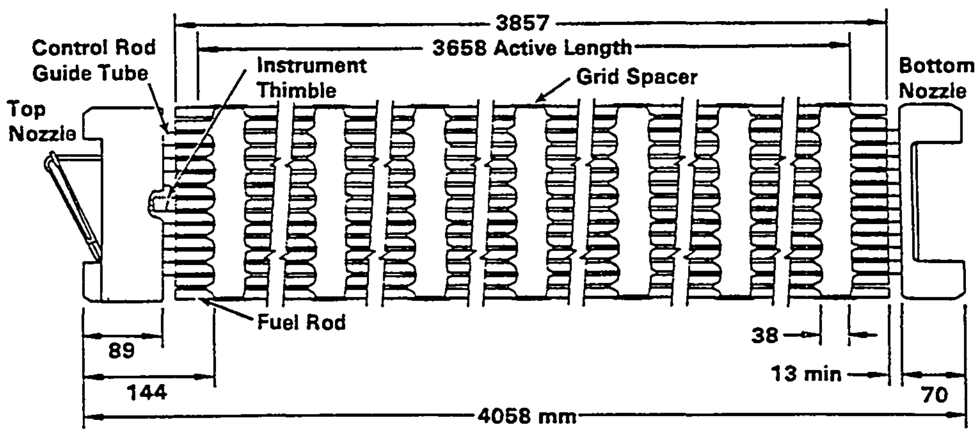

Figure 4.2. 15 x 15 PWR Fuel Assembly

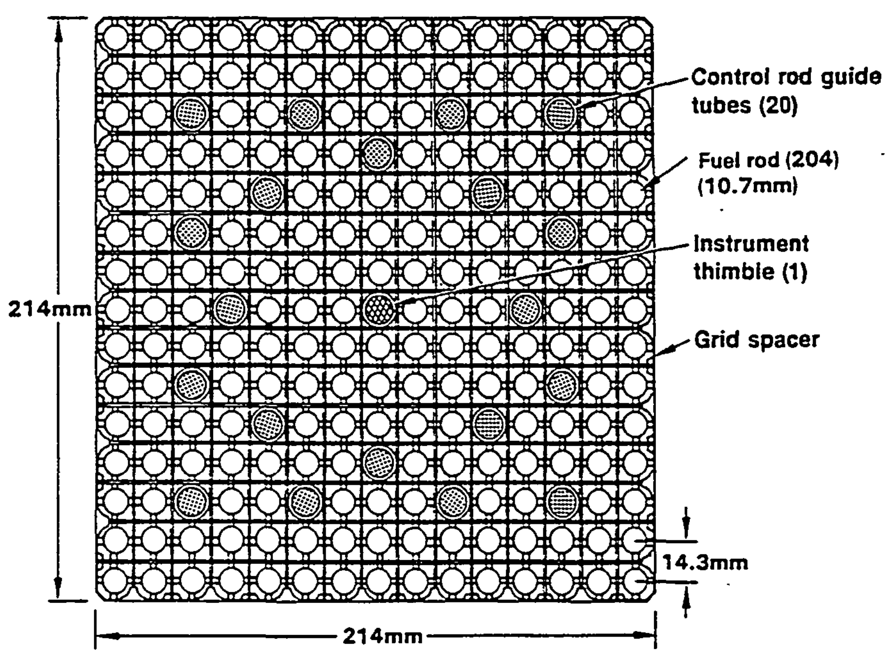

Figure 4.3. 15 x 15 PWR Fuel Assembly Cross-Section

The fuel rods consist of uranium oxide (UO_2) ceramic pellets contained in slightly cold-worked and partially annealed Zircaloy 4™ tubing, which is plugged and seal-welded at the ends to clad the fuel. Nominal dimensions include a 9.29-mm (0.37-in.) pellet diameter, 10.71-mm (0.42-in.) tube outside diameter, 0.62-mm (0.024-in.) tube thickness, and 3860-mm (152-in.) length.

Sufficient void volume and clearances are provided within the rod to accommodate fission gases released from the fuel, differential thermal expansion between the cladding and the fuel, and fuel swelling due to accumulated fission products without overstressing of the cladding or seal welds. Shifting of the fuel within the cladding is prevented during handling or shipping before core loading by a carbon-steel helical compression spring that bears on the top of the fuel pellet column. The hold-down force to prevent fuel shifting is obtained by compressing the spring between the top end plug and the top fuel pellet of the stack.

During assembly, the pellets are stacked in the cladding to the required fuel height. The compression spring is then inserted into the top end of the fuel, and the end plugs are pressed into the ends of the tube and welded. During the welding process, the fuel rods are internally pressurized with helium to between 20.7 and 27.6 bar (300 and 400 psia).

The fuel rod void space is sized to ensure adherence to the pressure criterion. The end-of-life pressure is evaluated for the worst rod under expected conditions of fuel operation and at the peak steady-state power. The model used to predict the quantity of fission gas in the gap at end-of-life is based on an extensive comparison to published performance of fuel rods under a variety of conditions. The composition of the gas in the gap at end-of-life is a maximum of approximately 50% fission.

The fuel pellets are right circular cylinders consisting of slightly enriched UO_2 powder, which is compacted by cold pressing and sintering to the required density. The ends of each pellet are dished slightly to allow the greater axial expansion at the center of the pellets to be taken up within the pellets themselves and not in the overall fuel length. The nominal design enrichment ranged from 1.86 wt% to 3.20 wt%. Information on burnup history and enrichment for each fuel assembly used during testing is found within the individual performance reports (Creer et al. 1986; McKinnon et al. 1986, 1987a, 1987b, 1989, 1992; Strope et al. 1990). The nominal density is 95% of theoretical density for all of the fuel pellets.

4.3 Consolidated Fuel Canisters (PWR 15 x 15 fuel)

During the consolidation process, the fuel rods were removed from 48 of the PWR fuel assemblies described in the previous section and placed into 24 canisters. Two-to-one consolidation was consistently achieved, because each canister was able to hold 410 fuel rods and two fuel assemblies provided 408 rods. This left two extra fuel rod storage locations per canister. Simulated guide tubes with funnel-shaped tops were placed in seven canisters to provide locations for inserting TC lances during performance testing. The simulated guide tubes displaced three fuel rod locations. The overflow fuel rod caused by inserting a guide tube in a canister was placed in the next canister of fuel.

A stainless steel fuel canister (see Figure 4.4) consists of a base and a top-locking cover. A series of spacers, support bars, and tines is attached to the base of the canister to align and hold the fuel rods during consolidation. Once the fuel has been placed on the base, the top cover is placed over the fuel and locked into place. The design of the top cover, the sliding fit between the top cover and base, and the canister locking mechanism do not seal the canister but do limit gas flow into and out of the canister. The loaded canister is 216 mm (8.5 in.) square by 4053 mm (159.57-in.) long. The lower end plate and support angles attached to the top cover raise the fuel 41.5 mm (1.64 in.) off the bottom of the cask.

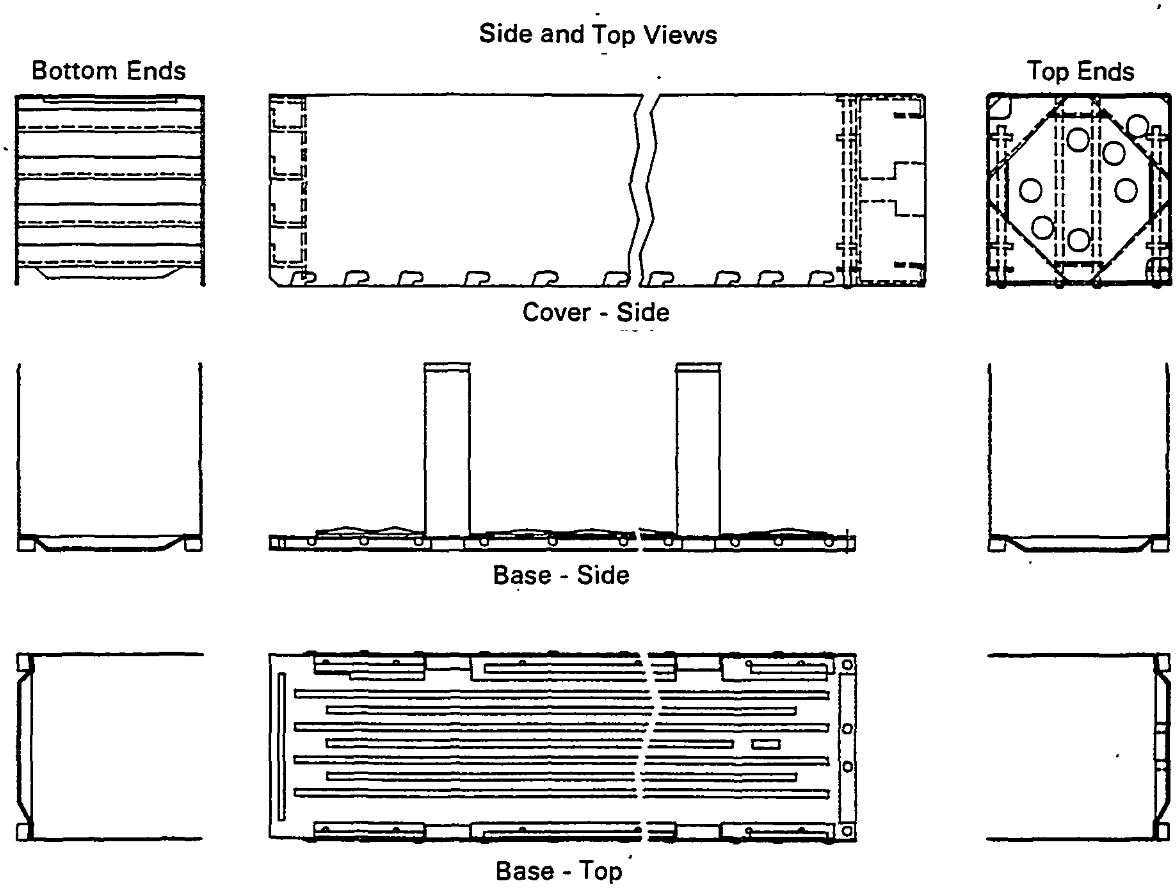

Figure 4.4. Consolidated Fuel Canister

5.0 Spent Fuel Integrity

This report combines gas sampling information from the spent fuel dry storage metal cask performance tests and from cask monitoring activities. It documents the condition of the BWR fuel from Nebraska Power's Cooper Station and the PWR fuel from VP's Surry reactor before testing and the effect of testing on fuel integrity, as ascertained through gas sampling during cask performance tests at GE-Morris and at INEL and subsequent cask surveillance and monitoring at INEL. Results are presented for tests using both intact and consolidated PWR spent fuel. The pretest and post-test fuel conditions are described as are the significant results obtained from gas sampling during and after the cask performance testing.

Before testing, the Surry PWR spent fuel assemblies used in the cask performance tests were characterized using in-basin ultrasonic examinations and video scans. The BWR fuel was characterized using in-basin sipping and video scans. Cask internal cover gas samples were taken during testing. After testing, selected fuel assemblies were videotaped and photographed. Then, fuel assemblies used in the TN-24P and MC-10 cask performance tests, along with a few Turkey Point (Florida) reactor spent fuel assemblies, were consolidated and loaded into the TN-24P cask for another DOE-funded performance test. Later, a cooperative agreement was established with SNC, and 17 of the consolidated fuel canisters from the TN-24P cask were used in a performance test of SNC's ventilated concrete cask.

Performance tests involved a combination of cover gases and cask orientations. The backfill environments were vacuum, nitrogen, and helium; nitrogen and helium were sampled and analyzed to detect leaking spent fuel rods. The integrity of the fuel assemblies was determined from cover gas sampling (Creer et al. 1986; McKinnon et al. 1986, 1987a, 1987b, 1989). After the conclusion of each performance test, periodic gas sampling was conducted on each cask as part of a surveillance and monitoring activity.

Data on fuel integrity, which has been addressed in individual reports on each cask and performance test, have been consolidated in this report. This report also contains surveillance data taken after the conclusion of each performance test that have not been reported previously. Information on spent fuel integrity is of interest in evaluating the impact of dry storage and fuel-handling operations associated with dry storage on the behavior of spent fuel rods during long-term dry storage. The main areas of interest include the integrity of the fuel cladding, the condition of the spent fuel assembly hardware, and the character and condition of the crud. Specific information obtained through the cask performance tests resulted from visual observation and ultrasonic examination of the condition of the cladding, fuel rods, and fuel assembly hardware before dry storage and consolidation of the fuel; and a qualitative determination of the effect of dry storage and fuel consolidation on fission gas release from the spent fuel rods.

The results of the gas sampling during testing and after surveillance indicate that dry storage of unconsolidated fuel at the temperatures associated with the performance tests does not seem to have an adverse effect on fuel integrity. However, consolidation of spent fuel rods appeared to produce pin holes or hair-line cracks in the fuel cladding and resulted in more leaking rods during dry storage.

Four examination methods were used to assess the integrity of the spent fuel used in the cask performance tests. Methods common to the PWR and BWR fuel included visual observations, full-length black and white videos, color photographs, and analysis of the cover gas in the cask. In addition to these methods, the BWR spent fuel was examined by in-basin sipping, and the PWR spent fuel from VP's Surry reactor was examined using in-pool ultrasound.

5.1 Pre-Test Fuel Inspections

This section of the report is divided into two subsections: the first describes the condition of the fuel before testing based on in-basin ultrasonic evaluations, in-basin sipping, and visual examinations; the second presents results of gas sampling during and after cask performance testing.

5.1.1 In-Basin Sipping

The 52 BWR fuel assemblies from the Cooper reactor used in the REA-2023 cask performance test were sipped "in-basin" and "in-vessel" to investigate fuel rod integrity before dry storage and to determine whether any of the fuel rods developed leaks during testing. In-basin sipping was conducted in the fuel storage basin, while in-vessel sipping was conducted in the calorimetry vessel located in the smaller unloading basin. The assemblies were sipped both before and after testing, and one assembly was resipped each time fuel assemblies from a new basket were sipped, to ensure appropriate consistency. In-basin sipping consisted of placing a hood over the selected BWR assembly and analyzing the water that was drawn off the top of the assembly. All the sipping data were compared with background readings to assess fuel integrity.

Detailed results for the in-basin sipping are given in McKinnon et al. (1986). Data are provided on the date, time, basin temperature, and background radiation levels for both cesium-137 and cobalt-60. The cesium-137 levels detected during pretest and post-test sipping are summarized in that report. Although there is some variation in the differences between the pretest and post-test radionuclide concentrations, the values were lower than would exist if the assembly contained leaking fuel rods. The sipping results did not indicate any leaking fuel rods in any of the fuel assemblies used in the cask either before or after cask testing.

5.1.2 In-Basin Ultrasonic Inspections

In-basin ultrasonic inspections were performed on the PWR fuel at VP's Surryreactor using the Babcock & Wilcox Failed Fuel Rod Detection System (FRDS). This portable system is designed to be used under water in spent fuel pools. The system consists of an underwater manipulator, an ultrasonic probe, electronic controls, recording equipment, and a support plate. The FRDS system uses ultrasonic techniques to differentiate between nonleaking and leaking rods by detecting the presence of moisture in the latter.

Several factors must be considered when interpreting the X-Y plots of each fuel assembly, including pulse height, crud deposits on the fuel rods, fuel assembly, and fuel rod bowing. Because the proximity of the ultrasonic (UT) probe to the fuel rods may vary with fuel rod bowing, pulse amplitudes would also tend to vary in height. Fuel assemblies that had "suspect" traces were re-examined from all four faces. In each case, fuel rods that were "suspect" during the initial examination were found to be either "clear" or "indication" (failed) during re-examination. Only Surry reactor fuel assemblies with "clear" examinations were used in the cask performance testing.

5.1.3 Visual, Video, and Photographic Examinations

The PWR fuel assemblies were examined visually to establish their general condition after shipment from VP, after handling at the INEL hot shop, after cask performance testing, and during consolidation.

Similar exams were made of the Cooper BWR fuel during the REA-2023 performance tests at GE-Morris. Two kinds of visual examinations were used, black-and-white videos and color photography of selected fuel assemblies.

The black-and-white videos taken at GE-Morris, VP, and INEL did not provide sufficient detail to characterize the crud or very small features on the fuel rods. They did not reveal any indication of significant variations in the fuel rods after shipment, handling, and performance testing. The resolution of the videotapes did not provide enough information to adequately determine the integrity and condition of the fuel and fuel cladding. Examination of the video scans showed that all the fuel assemblies and fuel rods look basically the same when viewed from outside the assemblies. There was some discoloration of the fuel rod cladding in the area of the grid spacers, which was expected.

Color photographs showed that a typical orange/reddish crud (probably Fe_2O_3) was evenly deposited on all of the zircaloy-2 cladding and fuel assembly hardware. There were no noticeable changes in the characteristics or adherence of the crud during handling operations involving the spent fuel assemblies at GE-Morris or INEL. Some scratches and worn spots were apparent on the spacer grids and some fuel rods, but these features did not change as a result of examination or handling operations. In general, the fuel rods were in excellent condition with a very adherent crud layer.

More visual examinations of the fuel were conducted during the dry rod consolidation program. According to Vinjamuri et al. (1988a, 1988b):

> No noticeable cladding defects in the rod surfaces were observed for any of the fuel processed. The oxide layer on the surface of the fuel rods appears to be intact and firmly attached to the cladding. The oxide layer does not appear to be loose, thick, soft, or powdery. However, the oxide layer and some of the zirconium cladding was scraped from the rod surface by the spacer grids as the rod was pulled during fuel consolidation. Very little crud buildup on the surfaces of the rods was observed. The surfaces of the rods displayed only a thin oxide layer, which had the appearance of surface discoloration rather than any rough or loose material. The rod surfaces are discolored near the spacer grids. The discoloration has an appearance of a dark mottling of the surface and is progressively more predominant from the middle of the rod length toward the rod bottom. The rods are generally clean, with limited amounts of clad discoloration and oxidation.... The evidence of fuel rod growth since fabrication was visually obvious during the consolidation process.... Length variation between rods appears to be as much as 2 cm (0.8 inch). The rods that grew longer than others appeared to be randomly located within the fuel assembly. (Ellipses by McKinnon.)

5.2 Performance Test and Post-Test Cask Cover Gas Sampling

The cask cover gas was sampled several times during each cask performance test to evaluate the integrity of the spent fuel rods. Each sample was collected in a separate 500-cc stainless steel cylinder. The cylinders were checked for leaks before sampling. Initially, during the CASTOR-V/21 cask performance test, the cylinders were equipped only with quick disconnect-fittings and no bellows-sealed valves as part of the closure. During the early sampling efforts with the CASTOR-V/21 cask, the cover gas samples in the cylinders were diluted with ambient air from the vicinity of the sampling apparatus, air that leaked into the cylinder during shipment, and argon introduced at Lawrence Livermore National

Laboratory (LLNL) (where some of the samples were analyzed). In many cases, this dilution was made more severe by the collection of small amounts of cask cover gas, presumably due to short equilibration times between the cask and the sample bottle during the actual cask cover gas collection procedure. The end effects of small, diluted samples on the cask cover gas analyses were to increase detection limits and measurement uncertainties and to introduce questions of sample validity. Once bellows-sealed valves were added to the sampling cylinders, the problem of air leaking into the sampling cylinders was eliminated.

Gas sample analysis included mass spectroscopy and radiochemical gamma analysis. The gas sample analyses were performed at LLNL and INEL. Initially, at LLNL, and later, as upgrades were made in INEL's gas analysis capabilities, the gas sampling analysis was shifted from LLNL to INEL. The results of the gas analyses are presented in Tables 5.1, 5.2, 5.3, 5.4, and 5.5. Mass spectra were analyzed for all common fixed gases with masses less than 100 to verify the purity of the backfill gas composition. Only nitrogen (N_2), oxygen (0_2), helium (He), argon (Ar), and carbon dioxide (CO_2) concentrations above 0.01% can be detected in any of the samples. It is obvious from values in the table that significant amounts of air were introduced into many of the CASTOR V/21 gas samples, as previously discussed. TN-24P sample 4C-PT (Table 5.2) shows the effect of leaving a valve open; some krypton-85 gas was detected shortly after the sample was taken, but by the time the gas was analyzed, the cylinder content had reverted to air.

The integrity of the fuel rods was assessed from the radionuclide concentration based on gamma spectroscopy. Initial indications of the presence of radionuclides (screening analysis) were determined at the test site using a multichannel analyzer before shipment of the gas samples to the measurement laboratory. During surveillance and monitoring activities, which began in the fall of 1994, gas samples were analyzed by LITCO and ANL-W at INEL. These samples were also analyzed for carbon-14. The amount found was on the order of a nanocurie/mL, which is near the lower level of detectability of the instruments used.

Table 5.1. REA-2023 Cask Gas Sample Identification (BWR fuel)

Sample			Volume Percent			Radionuclide Concentration, pCi/mL			
No.	Date	Cover Gas	N_2	O_2	He	Screening Analysis	^{85}Kr	$^{14}CO_2$	^{14}CO
1A	11/27/84	N_2	99.96	0.018	--	<1	0.11 ± 0.02	--	--
2A	12/08/84	N_2	99.98	--	--	<1	0.19 ± 0.01	<0.08	--
3A	12/08/84	He	0.25	0.06	99.68	<1	<0.06	0.08 ± 0.01	--
4A	01/02/85	He	0.20	0.04	99.75	<1	<0.02	<0.02	--
5A	01/21/85	N_2	99.98	--	--	20.1 ± 0.3	18.70 ± 0.2	<0.02	--
6A	02/05/85	N_2	99.99	--	--	14680 ± 160	14680 ± 170	0.28 ± 0.01	0.110 ± 0.03
7A	02/05/85	He	0.31	0.07	99.61	17.0 ± 0.2	12.90 ± 0.2	0.20 ± 0.01	--
8A	03/12/85	He	0.02	--	99.98	35600 ± 400	34600 ± 400	<0.02	0.031 ± 0.005
13A	03/12/85	N_2	99.99	--	--	19.6 ± 0.2	15.8 ± 0.1	--	--
13C	03/15/85	N_2	99.98	0.01	--	3360 ± 40	3220 ± 60	0.06 ± 0.01	--
14A	04/11/85	He	0.05	--	99.93	36040 ± 630	35670 ± 390	<0.03	0.736 ± 0.004

Table 5.2. Gas Samples from the VSC-17 Cask Loaded with Consolidated Spent Fuel

Sample Number	Collection Time/Date	Volume Percent He	N_2	O_2	Ar	CO_2	H_2	Radionuclide Concentration, nCi/mL Screening Analysis	^{85}Kr	Cover Gas
TN-24	09-13-90	61.5	30	8	0.36	0.08	<0.01		62.0±13	He
TN-24A	09-24-90	99.9	0.03	<0.01	<0.01	0.08	<0.01	139.0		He
VSC-40A	0845/10-09-90	99.7	0.2	0.05	<0.01	<0.01	<0.01	ND[a]		He
VSC-50A	0900/10-09-90	99.97	0.02	ND	ND	<0.01	<0.01	ND		He
VSC-40A	1500/10-15-90	99.5	0.02	<0.01	ND	0.15	0.3	ND		He
VSC-50B	1505/10-16-90	99.5	0.02	<0.01	<0.01	0.14	0.3	0.9		He
VSC-40	1335/10-24-90	98.3	0.41	0.1	<0.01	0.27	0.91	3.0		He
VSC-5	1330/10-24-90	98.5	0.21	0.05	<0.01	0.27	0.91	2.5		He
VSC-17	0745/10-29-90	95.4	0.27	0.05	<0.01	0.8	3.5	6.0	6.4±0.7	He
VSC-17.C5	0800/10-31-90	91.5	1.9	0.14	0.01	1.0	5.5	5.3		He
VSC-17.C40	/082010-31-90	91.7	1.4	0.3	0.02	1.0	5.5	5.2	5.0±6	He
TN8L-2	1405/11-06-90	1.7	98.2	<0.01	0.05	<0.01	0.06	ND		N2
#40	1410/11-06-90	0.11	99.7	0.08	0.05	<0.01	0.07	ND		N2
TN8L-2	1320/11-13-90	0.06	99.2	<0.01	0.05	0.02	0.62	ND		N2
#40	1330/11-13-90	0.06	99.2	0.08	0.05	0.02	0.62	ND		N2
none	--/11-13-90									Vacuum
none	--/11-15-90									Vacuum
VSC-5	1500/11-15-90	97.4	0.3	0.01	<0.01	0.09	2.2	1.4		He
VSC-40	1505/11-15-90	97.4	0.31	0.01	<0.01	0.09	2.2	1.5		He
#40	1700/12-12-90	99.97	0.02	<0.01	<0.01	<0.01	<0.01	4.8		He

Table 5.2 (contd)

				Volume Percent				Radionuclide Concentration, nCi/mL		
Sample Number	Collection Time/Date	He	N_2	O_2	Ar	CO_2	H_2	Screening Analysis	^{85}Kr	Cover Gas
TN8L-2	1700/12-12-90	99.96	0.03	<0.01	<0.01	<0.01	<0.01	5.0		He
TN8L-2	/01-29-91	99.49	0.06	<0.01	<0.01	<0.01	0.44	1.1		He
Enhanced Surveillance and Monitoring										
#40	/01-29-91	99.49	0.06	<0.01	<0.01	<0.01	0.44	1.4		He
								LITCO Analysis	ANL-W Analysis	
VSC-17 #3	/01-95								17	He
VSC-17 #4	/01-95								12.6	He
VSC-17 #1	/03-22-95	96.96	0.76	0.17	<0.01	<0.01		8.4		He
VSC-17 #2	/03-22-95	97.75	0.14	0.01	<0.01	<0.01		9.5		He
VSC-17	/3-22-95	95.51	1.91	0.44	0.03	<0.01	2.11		10.3	He
VSC-17	/3-22-95	96.12	1.41	0.32	0.02	<0.01	2.12		11.9	He
VSC-17	/06-95							6.8		He
VSC-17	/06-95							6.8		He
VSC-17 #3	/6- 95	97.62	0.12	≤0.01	≤0.01	≤0.01	2.25		8.28	He
VSC-17 #4	6-95	97.65	0.12	≤0.01	≤0.01	≤0.01	2.22		8.01	He
VSC-17	7-17-96	97.41	0.13	0.01	≤0.01	≤0.01	2.44	7.59		He
VSC-17	7-17-96	97.43	0.12	≤0.01	≤0.01	≤0.01	2.44	7.97		He
VSC-17	/7-17-96	97.54	0.05	<0.01	<0.01	<0.01	2.40		7.67	He
VSC-17	/7-17-96	97.57	0.04	<0.01	<0.01	<0.01	2.38		10.8	He

(a) ND = not determined.

Table 5.3. Gas Samples from the CASTOR-V/21 Cask

Sample Number	Collection Time/Date	Cover Gas	Volume Percent: He	N_2	O_2	Ar	CO_2	H_2	Radionuclide Concentration, pCi/mL: Screening Analysis	^{85}Kr
1A	1400/9-06-85	He	69.30	21.90	5.967	2.798	0.029	<0.01	4.78 ± 0.39	0.09 ± 0.05
1B	1330/9-11-85	He	62.51	25.57	6.692	5.148	0.033	<0.01	4.60 ± 0.57	--
1D	1300/9-11-85	He	59.59	17.59	4.110	18.66	0.048	<0.01	15.9 ± 3.5	0.37 ± 0.13
2A	1920/9-11-85	N_2	<0.01	95.34	1.180	3.457	0.017	<0.01	<0.13	<0.05
2B	1930/9-11-85	N_2	<0.01	90.58	6.81	2.581	0.024	<0.01	0.22 ± 0.08b	--
2C	1345/9-13-85	N_2	0.031	68.79	3.134	27.95	0.059	<0.01	7.15 ± 0.69	0.26 ± 0.09
2D	1430/9-13-85	N_2	0.042	70.08	8.43	21.29	0.153	<0.01	5.98 ± 0.36	--
4B	1410/9-16-85	He	70.59	14.23	5.95	9.09	0.060	0.018	5.77 ± 0.58	--
4A	1419/9-16-85	He	32.38	50.55	13.31	3.700	0.053	<0.01	0.23 ± 0.16b	0.07 ± 0.04
4C	1045/9-20-85	He	65.69	1.560	0.156	32.58	<0.01	<0.01	1.92 ± 0.16	0.05 ± 0.01
4D	1050/9-20-85	He	<0.01	76.94	20.26	2.746	0.054	<0.01	<0.08	--
5A	NA/9-20-85	N_2	0.050	74.92	11.44	13.43	0.120	0.016	1.38 ± 0.46	<0.17
5B	NA/9-20-85	N_2	<0.01	73.95	18.77	7.217	0.060	<0.01	<0.08	--
5C	1300/9-23-85	N_2	0.048	85.63	5.559	8.533	0.065	0.159	4.01 ± 0.41	0.18 ± 0.07
5D	1300/9-23-85	N_2	0.068	91.06	0.158	8.462	0.044	0.207	8.28 ± 0.75	0.19 ± 0.11

Post Cask Performance Test Gas Sampling

Sample Number	Collection Time/Date	Cover Gas	He	N_2	O_2	Ar	CO_2	H_2	Screening Analysis	^{85}Kr
GN-3A	3-05-86	He	99.0	0.82	0.2	≤0.01	≤0.01	--	--	--
GN-3B	3-05-86	He	99.7	0.27	0.06	≤0.01	≤0.01	--	--	--
GN-4A	3-17-86	He	99.86	0.11	0.03	≤0.01	≤0.01	--	--	--
GN-4B	3-17-86	He	99.81	0.15	0.04	≤0.01	≤0.01	--	--	--
GN-5A	3-24-86	He	99.84	0.13	0.03	≤0.01	≤0.01	--	--	--
GN-5B	3-24-86	He	99.83	0.14	0.03	≤0.01	≤0.01	--	--	--
GN-8A	6-27-86	He	65.3	29.1	4.2	0.36	1.1	--	--	--
GN-9A	8-15-86	He	99.8	0.05	0.002	0.001	0.001	0.15	--	<0.0274
GN-9B	8-15-86	He	99.78	0.05	0.006	0.001	0.002	0.16	--	<0.0289
GN-10A	9-26-86	He	99.8	0.03	<0.01	<0.01	<0.01	0.17	--	<0.024
GN-11A	12-08-86	He	99.8	0.06	0.01	≤0.01	≤0.01	0.10	--	<0.057

Note: 9/20/94 No mask spectroscopy taken - casks were refilled with fresh helium on or about 10-11-94.

Table 5.3 (contd)

Sample Number	Collection Time/Date	Cover Gas	Volume Percent: He	N_2	O_2	Ar	CO_2	H_2	Radionuclide Concentration, pCi/mL: Screening Analysis	^{85}Kr
Samples analyzed by Lockheed Idaho Technologies Company										
V-21 #1	3-28-95	He	99.43	0.56	<0.01	≤0.01	≤0.01	≤0.01	--	ND[(a)]
V-21 #2	3-28-95	He	99.28	0.60	0.03	≤0.01	≤0.01	≤0.01	--	ND
GN-14A	7-10-96	He	98.51	1.33	0.11	≤0.01	≤0.03	ND	--	ND
GN-14B	7-10-96	He	98.86	1.07	0.04	≤0.01	≤0.03	ND	--	ND
Samples analyzed by Argonne National Laboratory - West										
V-21 #3	3-95	He	99.37	0.60	0.02	<0.01	<0.01	<0.01		ND
V-21 #5	3-95	He	99.42	0.56	0.01	<0.01	<0.01	<0.01		ND
V-21 #3	6-95	He	99.40	0.55	0.01	≤0.01	0.01	0.02	--	ND
V-21 #4	6/95	He	99.41	0.55	0.01	≤0.01	0.01	0.02	--	ND
V-21 #3	7/96	He	98.42	1.42	0.10	0.01	0.02	0.02		ND
V-21 #4	7/96	He	98.93	1.02	0.02	0.01	0.02	0.01		ND

(a) ND = Not determined.

Table 5.4. Gas Samples from the Transnuclear TN-24P Cask

Sample Number	Collection Time/Date	Cover Gas	Volume Percent: He	N_2	O_2	Ar	CO_2	H_2	Radionuclide Concentration, pCi/mL: Screening Analysis	^{85}Kr
In Test Westinghouse 15 X 15 PWR Assemblies										
1A-PR	1815/1-10-86	He	99.97	<0.01	0.020	<0.01	<0.01	<0.01	≤0.08	
1B-PR	1830/1-10-86	He	99.96	0.037	<0.01	<0.01	<0.01	<0.01	≤0.06	≤0.01
1C-PT	0945/1-14-86	He	99.92	0.056	0.017	<0.01	<0.01	<0.01	0.44 ± 0.06	≤0.02
1D-PT	0955/1-14-86	He	99.93	0.057	<0.01	<0.01	<0.01	<0.01	0.65 ± 0.24	≤0.02
2A-PR	1600/1-14-86	N2	<0.01	99.99	<0.01	<0.01	<0.01	<0.01	≤0.07	--
2B-PR	1605/1-14-86	N2	<0.01	99.99	<0.01	<0.01	<0.01	<0.01	≤0.07	≤0.02
2C-PT	1030/1-17-86	N2	<0.01	99.94	<0.01	<0.01	<0.01	<0.01	2.05 ± 0.29	--
2D-PT	1020/1-17-86	N2	<0.01	99.98	<0.01	<0.01	<0.01	0.011	2.65 ± 0.13	≤0.02
4A-PR	1258/1-24-86	He	99.99	<0.01	<0.01	<0.01	<0.01	<0.01	≤0.07	--
-PR	1303/1-24-86	He	99.93	0.077	<0.01	<0.01	<0.01	<0.01	≤0.07	≤0.02
4C-PT[a]	1245/1-27-86	He	0.379	77.73	20.83	0.931	0.058	<0.01	2657 ± 37	--
4D-PT	1255/1-27-86	He	87.42	9.792	2.592	0.118	0.014	<0.01	7269 ± 166	--
5A-PR	1939/1-27-86	N2	0.063	98.93	0.937	0.063	<0.01	<0.01	277 ± 7	--
5B-PR	1947/1-27-86	N2	0.058	98.68	1.162	0.074	<0.01	<0.01	251 ± 6	252 ± 3
5C-PT	1115/1-31-86	N2	<0.01	99.95	<0.01	0.028	0.012	<0.01	2132 ± 30	--
5D-PT	1125/1-31-86	N2	<0.01	99.93	<0.01	0.029	<0.01	<0.01	2110 ± 24	2077 ± 38
Post Cask Performance Test Gas Sampling										
TN-1A	1300/2-28-86	He	99.99	<0.01	<0.01	<0.01	<0.01	<0.01	7.31 ± 0.10	--
TN-1B	1310/2-28-86	He	99.99	<0.01	<0.01	<0.01	<0.01	<0.01	7.09 ± 0.1	7.24 ± 0.54
TN-1C	2-28-86	He	99.9	<0.03	<0.01	<0.001	<0.005	--	--	--
TN-1D	2-28-86	He	99.9	<0.02	<0.01	<0.001	<0.006	--	--	--
TN-2A	8-13-86	He	99.77	0.16	0.04	0.001	0.020	--	--	ND[b]
TN-2B	8-13-86	He	99.82	0.12	0.03	0.001	0.018	--	--	ND
TN-3A	12-11-86	He	99.35	0.49	0.13	0.007	0.016	0.009	--	12.9 ± 0.6
TN-31A	5-07-87	He	99.83	0.11	0.03	<0.01	0.03	<0.01	--	20.7

Table 5.4 (contd)

Sample Number	Collection Time/Date	Cover Gas	Volume Percent He	N_2	O_2	Ar	CO_2	H_2	Radionuclide Concentration, pCi/mL Screening Analysis	^{85}Kr
Dry rod consolidation began May 1987; finished loading cask with consolidated fuel September 28, 1987										
Pretest	12-18-87	He							86,320	
B1-P	1505/1-07-88	He	99.9	0.03	<0.01	<0.01	<0.01	<0.01		
A1-PT	1000/1-12-88	He	99.9	0.03	<0.01	<0.01	<0.01	<0.01	24,530	21,300 ± 1700
A2-PR	1700/1-13-88	N2	0.08	99.86	0.01	0.04	<0.01	<0.01		
A2-PT	1000/1-18-88	N2	0.10	99.82	<0.01	0.04	0.04	<0.01	7,680	6000 ± 500
A4-PR	1330/1-27-88	He	99.95	0.04	0.01	<0.01	<0.01	<0.01		
A4-PT	0930/2-01-88	He	99.98	0.02	<0.01	<0.01	<0.01	<0.01	2,100	1300 ± 140
A5-PR	0900/2-03-88	N2	0.10	99.84	0.01	<0.01	<0.01	<0.01		
A5-PT	0900/2-08-88	N2	0.02	99.95	0.02	<0.01	0.01	<0.01	7,160	4500 ± 400
A7-PT	1540/2-29-88	He	99.97	0.02	<0.01	<0.01	<0.01	<0.01	17,130	14,100 ± 1000
Post-test	8-11-88	He	99.77	0.15	0.04	<0.01	0.04	ND	118,000	69,890 ± 5170

(a) Bottles received with one open valve on each sample.
(b) Not detected.

Table 5.5. Gas Samples from the Westinghouse MC-10 Cask

Sample Number	Collection Time/Date	Cover Gas	Volume Percent						Radionuclide Concentration, pCi/mL	
			He	N_2	0_2	Ar	CO_2	H_2	Screening Analysis	^{85}Kr
1A-PR	5/29/86	He	99.96	0.03	0.003	--	<0.01	<0.01		
1B-PR	1600-5/29/86	He	99.89	0.092	0.019	<0.01	<0.01	<0.01	≤0.09	--
1C-PR	1605-5/29/86	He	99.95	0.039	<0.01	<0.01	<0.01	<0.01	≤0.10	≤0.02
1A-PT	6/02/86	He	99.85	0.09	0.02	--	0.004	0.03		
1B-PT	1400-6/02/86	He	99.93	0.040	<0.01	<0.01	<0.01	0.016	≤0.09	--
1C-PT	1410-6/02/86	He	99.94	0.034	<0.01	<0.01	<0.01	0.018	≤0.09	≤0.01
2A-PR	6/03/86	N2	--	99.97	<0.01	0.016	--	<0.01		
2B-PR	1450-6/03/86	N2	0.370	99.60	<0.01	0.015	<0.01	0.018	≤0.08	--
2C-PR	1500-6/03/86	N2	<0.01	99.97	<0.01	0.015	<0.01	0.018	≤0.08	≤0.02
2A-PT	6/06/86	N2	0.07	99.87	0.01	0.015	0.009	0.02		
2B-PT	0915-6/06/86	N2	0.033	99.92	<0.01	0.015	<0.01	0.032	0.24 ± 0.16	--
2C-PT	0930-6/06/86	N2	<0.01	99.95	<0.01	0.015	<0.01	0.033	0.30 ± 0.11	≤0.02
4A-PR	6/13/86	He	99.96	0.03	<0.01	--	--	<0.01		
4B-PR	1320-6/13/86	He	99.99	<0.01	<0.01	<0.01	<0.01	<0.01	≤0.07	--
4C-PR	1325-6/13/86	He	99.99	<0.01	<0.01	<0.01	<0.01	<0.01	≤0.08	<0.01
4B-PT	1420-6/16/86	He	99.99	<0.01	<0.01	<0.01	<0.01	0.01	≤0.06	--
4C-PT	1540-6/16/86	He	99.99	<0.01	<0.01	<0.01	<0.01	<0.01	≤0.08	≤0.02
5A-PR	6/18/86	N2	0.8	99.2	<0.01	0.02	--	--		
5B-PR	1545-6/18/86	N2	0.707	99.25	<0.01	0.019	<0.01	0.011	≤0.08	--
5C-PR	1550-6/18/86	N2	0.051	99.92	<0.01	0.019	<0.01	<0.01	≤0.08	≤0.02
5A-PT	6/23/86	N2	0.05	99.89	<0.01	0.02	--	0.03		
5B-PT	0920-6/23/86	N2	0.687	99.22	<0.01	0.018	0.010	0.033	≤0.07	--
5C-PT	0930-6/23/86	N2	0.386	99.55	<0.01	0.019	<0.01	0.030	≤0.07	≤0.02
Post Cask Performance Test Gas Sampling										
MC-1A	1330-8/06/86	He	99.67	0.195	<0.01	<0.01	<0.01	0.119	≤0.08	--
MC-1B	1345-8/06/86	He	99.71	0.156	<0.01	<0.01	<0.01	0.119	≤0.08	≤0.02
MC-3A	9-30-86	He	99.6	0.18	0.01	<0.01	0.01	0.24	--	<0.031
MC-4A	12-16-86	He	99.97	0.17	0.004	0.001	ND(a)	0.005	--	<0.02
MC-31A	5-21-87	He	95.88	3.09	0.80	0.04	0.02	0.18	--	--
Dry Rod Consolidation from May 1987 to September 1987										
	3-03-88	He	98.7	1.0	0.24	0.01	0.02	≤0.01	--	--
	8-10-88	He	99.0	0.82	0.02	0.01	0.12	ND	--	ND

(a) Not detected.

The differences between the screening analysis and the more exact laboratory measurements are apparent in Table 5.3. It was generally expected that the screening analysis would agree with the laboratory-measured krypton-85 result, but in these samples the screening counts were significantly greater than the laboratory-measured krypton results. The disparity between the concentrations remains unexplained; however, the relatively low amounts detected indicate that no leaking fuel rods were present in the GNS CASTOR-V/21 and MC-10 casks during performance testing with unconsolidated fuel and up to about a year after testing had been completed. This is particularly significant, because the first few assemblies loaded in the CASTOR-V/21 cask were exposed to air for approximately 200 hours during incremental loading of the cask and fuel assembly/ basket inspections at a reduced temperature. In addition, after testing was completed and long-term surveillance started, all the fuel assemblies were in a 70% helium and 30% air environment for approximately four months because a quick disconnect fitting on the CASTOR-V/21 cask lid had not sealed shut.

Krypton gas was found in samples 6A, 6B, 8A, 8B, 14A, and 14B for the REA-2023 cask (Table 5.1). The estimated amount of krypton-85 released to the cask during each sample period was determined, and these amounts were accumulated and plotted as a function of total cask storage time, as shown in Figure 5.1. The release of krypton-85 to the cask was essentially linear during the 2.5 months of testing and indicates that the defect in the cladding was very small.

The krypton releases during the REA-2023 cask test are compared in Figure 5.2 with the releases observed in assembly B02 in the fuel temperature test (FTT) (Johnson and Gilbert 1983) that was conducted to assess dry storage of spent fuel. The background level (3.086 x 10-8 Ci) in the FTT, 1% release of the krypton-85 produced in a single fuel rod, and 20% release of that produced in a single fuel

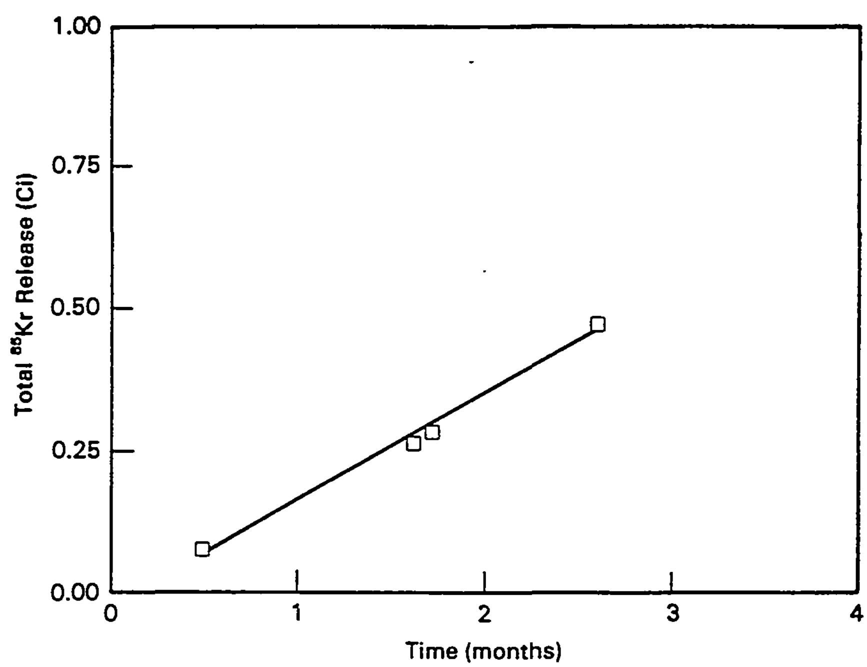

Figure 5.1 Release of Krypton-85 Fission Gas During the REA-2023 BWR Cask Performance Test

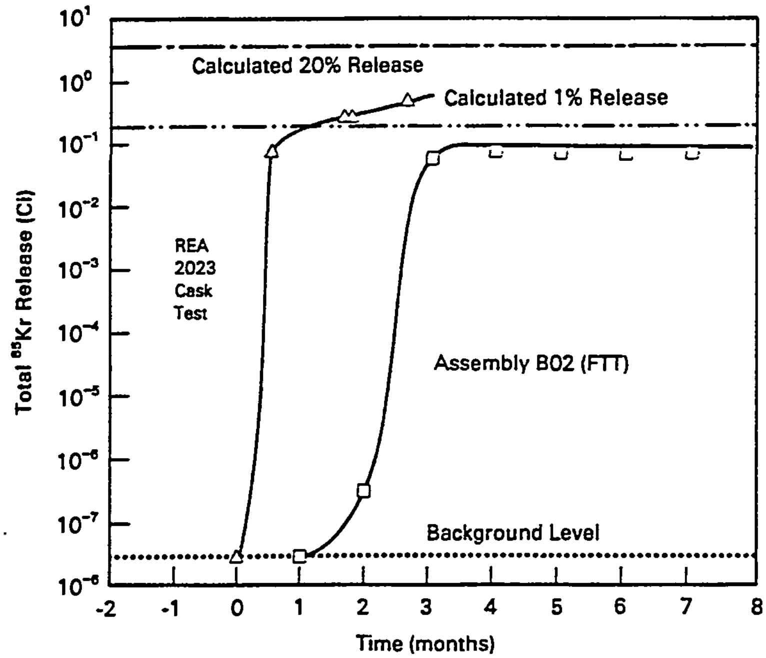

Figure 5.2. Comparison of Krypton-85 Fission Gas Release During the REA-2023 BWR Cask Performance Test and the Fuel Temperature Test

rod are plotted in Figure 5.2 for comparison with the release observed in the cask test. The krypton levels observed in the cask test are less than those expected for a single fuel rod with 20% release of krypton-85 from the fuel. The krypton levels in the FTT were less than that from a fuel rod with 1% release. The higher release during the cask test would be consistent with that expected from failure of a corner rod that probably experienced higher power during reactor operation. If the background level from the FTT is also used with the cask test data, it appears that the release of fission gas from both tests would have similar behaviors. With increasing time, the release of krypton in the cask would probably have decreased, as indicated by the curve for the FTT data. The difference in release rates during the first few months of testing in the REA cask and the FTT may have resulted from a difference in the size of the defects or in the number of failed rods.

There was no confirmation of a leaking fuel rod either by visual inspection or sipping of the fuel assemblies after the cask test. An extremely small defect may have opened up, perhaps at a previous cladding crack. The LLNL gas analyses provided the only indication of a leaking fuel rod. In any event, leaking fuel rods had no impact on the basin operation or handling of the fuel assemblies after the cask test.

During the performance test with the TN-24P cask loaded with unconsolidated fuel, the amount of krypton-85 detected during test runs 4 and 5 (samples 4D-PT and 5D-PT, Table 5.4) indicated a leaking fuel rod was present during this portion of the test. The relatively low amounts of krypton-85 in the other samples are what would be expected to result from crud, not a leaking fuel rod. The increase in krypton-85 content coincides with the cask being rotated from vertical to horizontal, which suggests that a fuel rod(s) may have started leaking because of the change in position. The decay in the leak rate, sample 5D-PT being less than sample 4D-PT and sample TN-1B being less than the previous two samples, indicates that the leak was small (or most of the gas had already been lost) and it took several days to vent the gas from the fuel rod(s).

In May 1987, 36 of the 48 intact fuel assemblies in the TN-24P and MC-10 casks and 12 intact assemblies that had been in the Turkey Point Reactor were consolidated into 24 consolidated fuel canisters as part of INEL's Dry Rod Consolidation Technology Project. The consolidated fuel canisters were placed in the TN-24P cask. The remaining 12 intact assemblies from the TN-24P and MC-10 cask were placed in the MC-10 cask. Gas samples taken from the MC-10 cask after dry rod consolidation suggest that there were no leaking fuel rods in that cask (see Table 5.5).

During the fuel rod consolidation process, the exhaust gases from the consolidation area were monitored to detect the release of radioactive gases from the fuel that would indicate a cladding failure. One of the conclusions reached was that all fuel rods from the 48 assemblies were pulled and canisterized without rod failures. Pulling forces and rod profiles were recorded during the consolidation process. The minimum and maximum breakaway forces were 80 and 350 Newtons (8.2 and 35.8 kgf), respectively. The average X and Y diameters were 10.669 and 10.668 mm. The as-fabricated fuel rod diameter was 10.719 $\pm$ 0.0025 mm, indicating there was fuel rod cladding creep-down (Vinjamuria et al. 1988b).

Later, during the performance test of the TN-24P cask using consolidated fuel, krypton-85 was released to the cask. Based on a combination of ORIGEN2 predictions and experimental measurements (Barner 1985; Guenther et al. 1988), it was estimated that four or more fuel rods may have developed leaks between the end of cask loading and the beginning of cask performance testing, three or more fuel rods during cask performance testing, and another five fuel rods in the six-month period following testing. The rate of release was observed to decrease with time from cask loading. Shortly after the last gas sample was taken from the fully loaded TN-24P cask, 17 canisters of consolidated fuel were removed from the TN-24P cask and loaded into the VSC-17 cask. The performance tests for the VSC-17 cask showed a nominal amount of krypton-85 but not enough to indicate a new leaking fuel rod. From the end of the VSC-17 performance testing in early 1991 until September 1994, the VSC-17 was undisturbed. Recent gas samples, taken since September 1994, indicate that the atmosphere in the VSC-17 has not changed significantly. There has been a small amount of krypton-85 release, below the quantity expected for a single rod release, and there has been some buildup of hydrogen in the cask. The amount of hydrogen is consistent with off-gassing of the RX277 neutron shield material in the lid. Similar amounts of hydrogen were observed during cask performance testing.

As can be seen from Table 5.4, significant amounts of krypton-85 were released from the consolidated fuel in the TN-24P cask. Additional test data that permitted calculations of the cumulative release of krypton are given in Table 5.6. The cumulative release is shown in Figure 5.3. The single-rod release of krypton for a PWR rod is based on a combination of ORIGEN2 predictions of total krypton-85 gas available and experimental measurements. The experimental measurements indicated that no more than 0.5% of the available krypton-85 gas was released. The rest of the gas was captured in the fuel (Barner 1985; Guenther et al. 1988). For the fuel used in the testing, the expected single rod release of krypton-85 would be between 0.034 and 0.056 Ci. An average value of 0.045 Ci has been used for determining the number of breached rods in Figure 5.3, which accounts for most of the time helium or nitrogen backfills

Table 5.6. Release of Krypton-85 in the TN-24P Cask Loaded with Consolidated Fuel

Sample Number	Estimated Cask Gas Tempera-ture°C	Cask Gas Volume m^3	Cask Pressure mBar	Krypton-85 Concentration: Sample, nCi/cc: Screen	Krypton-85 Concentration: Sample, nCi/cc: CPP	Krypton-85 Concentration: Cask, Ci: Screen	Krypton-85 Concentration: Cask, Ci: CPP	Gas Residence Time, days
Pretest	150	2.61	1500	86.32		0.231	0.000	80.00
A1-PT	150	2.61	1500	24.53	21.30	0.066	0.057	4.79
A2-PT	190	2.61	1500	7.68	6.00	0.019	0.015	4.65
A4-PT	150	2.61	1500	2.10	1.30	0.006	0.003	4.69
A5-PT	200	2.61	1500	7.17	4.50	0.017	0.011	5.75
A7-PT	150	2.61	1500	17.13	14.10	0.046	0.038	6.05
Post Test	150	2.61	1500	118.00	69.89	0.316	0.187	161.00
Pre VSC-17	140	2.61	1500	139.00	100.80	0.381	0.276	1288.00

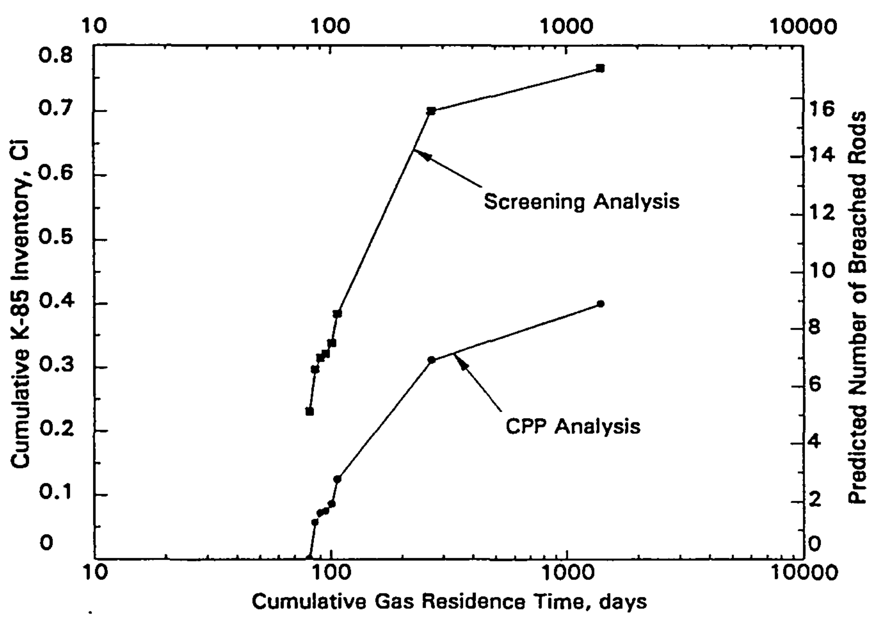

Figure 5.3 Cumulative Release of Krypton-85 Gas from the TN-24P Cask Loaded with Consolidated Fuel

were in the cask, from the time the cask was fully loaded with consolidated fuel until six months after the TC lances were removed at the end of testing. The krypton-85 release shown in Figure 5.3 does not include any that may have been released during the vacuum runs. The data indicate that four or more rods may have developed leaks before the pre-test TC lance insertion, three or more rods during testing, and another five in the six months after testing.

The amount of krypton-85 released during and after the TN-24P cask performance test with consolidated fuel is significantly higher than that released in previous cask testing with unconsolidated fuel. Before this test, four cask performance tests of similar duration and scope had been performed, and only two indications of krypton release were observed. The magnitude of the releases in the previous tests and surveillance periods indicated that each was limited to a single rod cladding breach. The previous tests involved about 16,700 spent fuel rods, whereas this test involved about 9800 rods. It is hypothesized that the larger amount of krypton-85 released in this test and post-test surveillance results from additional cladding leaks caused by enlargement of incipient cladding flaws during pulling and flexing of the fuel rods during the consolidation process. The enlarged cladding flaws combined with cladding creep during cask testing and surveillance periods allowed leak paths to develop. The leakage has not affected operations.

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L. A. Albertson
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P.O. Box 1625, MS 9208
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R. Anderson
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220 Stoneridge Drive
Columbia, SC 29210

R. O. Anderson
Northern States Power Company
15 South Church Street
Minneapolis, MN 55401

W. B. Andrews
SAIC
Valley Bank Center
101 Convention Center Drive,
Suite 407
Las Vegas, NV 89198

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