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ABSTRACT

A performance test of a Pacific Sierra Nuclear VSC-17 ventilated concrete storage cask loaded with 17 canisters of consolidated PWR spent fuel generating approximately 15 kW was conducted. [a] The performance test included measuring the cask surface, concrete, air channel surface, and fuel temperatures, as well as cask surface gamma and neutron dose rates. Testing was performed using vacuum, nitrogen, and helium backfill environments. Pretest predictions of cask thermal performance were made using the COBRA-SFS computer code. Analysis results were within 15°C of measured peak fuel temperature. Peak fuel temperature for normal operation was 321°C. In general, the surface dose rates were less than 30 mrem/h on the side of the cask and 40 mrem/h on the top of the cask.

INTRODUCTION

Since the enactment of the Nuclear Waste Policy Act in 1982, large metal spent fuel storage casks have been evaluated through tests and analyses for interim or lag storage at utility reactors and for use in Federal facilities associated with storage/disposal systems. Conceptual designs and analyses for interim or lag storage indicate that concrete spent fuel storage casks may be economically and technically competitive with metal casks. Therefore, it was desirable to conduct a Concrete Cask Testing (CCT) Project to evaluate the benefits of concrete as a storage cask material and to provide assistance in licensing activities if appropriate.

In 1987, DOE Office of Civilian Radioactive Waste Management initiated activities to obtain concrete storage modules for testing that could be used at reactors or Federal interim storage or disposal sites. The modules/casks were to be different from those previously tested at GE-Morris by DOE and EPRI, at INEL by DOE, Virginia Power, and EPRI at H.B. Robinson Nuclear Reactor by DOE, Carolina Power and Light, and EPRI[1-6] but were to be similar to those concepts that could be employed at nuclear reactor sites and Federal facilities. In late 1988 DOE entered into a Cooperative Agreement with Pacific Sierra Nuclear Associates (PSN) to demonstrate the performance of a small ventilated concrete storage cask.

This summary report focuses on a heat transfer and shielding performance test conducted on a Pacific Sierra Nuclear VSC-17 pressurized-water reactor (PWR) spent fuel storage cask loaded with consolidated spent fuel. Additional information of the performance test can be found in reference 7. The performance testing consisted of pretest preparations, performance testing, and post-test activities. Pretest preparations involved performing predictive code analyses and conducting cask-handling dry (cold) runs. The performance tests included six runs with the cask in a vertical orientation, four vent blockage conditions, and three backfill environments.

CASK DESCRIPTION

The PSN VSC-17 spent fuel storage cask system is a passive device for storing 17 assemblies/canisters of spent nuclear fuel. The VSC-17 system consists of an outer Ventilated Concrete Cask (VCC) and an inner Multi-Assembly Sealed Basket (MSB). The cask system is shown in Figure 1. Most of the decay heat, gener-

[a] The work was performed by the Pacific Northwest Laboratory (PNL) and the Idaho National Engineering Laboratory (INEL) for a cooperative agreement between the U.S. Department of Energy, Office of Civilian Radioactive Waste Management and Pacific Sierra Nuclear with support from the Wisconsin Electric and Power Company and the Electric Power Research Institute. PNL is operated for DOE by Battelle Memorial Institute under contract DE-AC06-76RLO 1830.

ated in the spent fuel, is removed by the cooling air flow in the annulus between the MSB and the VCC. Natural circulation drives the cooling air flow vertically through the annular path and carries the heat to the outside 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 MSB cavity is filled with helium at slightly sub-atmospheric pressure. The helium atmosphere enhances the overall heat transfer capability and prevents oxidation of the fuel and corrosion of the basket components.

The VCC is a one-piece reinforced cylindrical structure, which provides structural support, shielding, and natural convection cooling for the MSB. The concrete wall thickness is sufficient to limit exterior surface radiation dose rates to less than 30 mrem/h. The air inlet and outlet vents are steel lined penetrations that are non-planar paths to minimize radiation streaming. The internal cavity of the reinforced concrete cask is formed by a steel cylindrical liner and a flat bottom plate. Two lifting lug assemblies are embedded at the top face of the cask and are designed to lift the cask vertically. A cask weather cover plate provides additional shielding and provides a cover to protect the MSB from the environment. The cover is bolted in place and has a sheet rubber seal.

SPENT FUEL

The spent fuel used in this test came from two PWR reactor sources, Surry and Turkey Point. During a consolidation process[8] the fuel rods were removed from the Westinghouse 15 x 15 PWR spent fuel assemblies and placed into fuel canisters. Two-to-one consolidation was consistently achieved and the canisters had sufficient room so that a simulated guide tube with funnel-shaped tops could be placed in several of the canisters to provide locations for inserting TC lances during performance testing.

PRETEST ANALYSIS

The ORIGEN2 code[9] was used to predict decay heat generation rates of the PWR spent fuel used in the VSC-17 cask performance test. Based on pretest ORIGEN2 predictions, fuel rod decay heat generation rates totaled approximately 14.9 kW during testing (Table 1). The decay heat output of the canisters of consolidated rods ranged from 700 to 1050 W, with an average output per canister of 875 W at the start of testing. The fuel had cooling times of 9 to 15 years. The fuel loading pattern was selected to create a temperature profile that would be peaked in the center; the hottest canisters of fuel were loaded in the center of the cask with cooler fuel canisters loaded around the outside.

Figure 2 shows the axial decay heat profile used for the consolidated fuel canisters. Measured axial power profiles were not available for predicting axial decay heat profiles; therefore, axial gamma radiation scans previously obtained on Turkey Point reactor fuel assemblies were used to predict the fuel's axial burnup

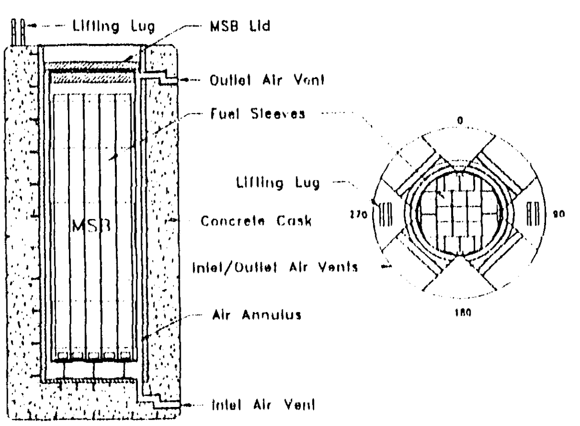

Fig. 1. VSC-17 Cask

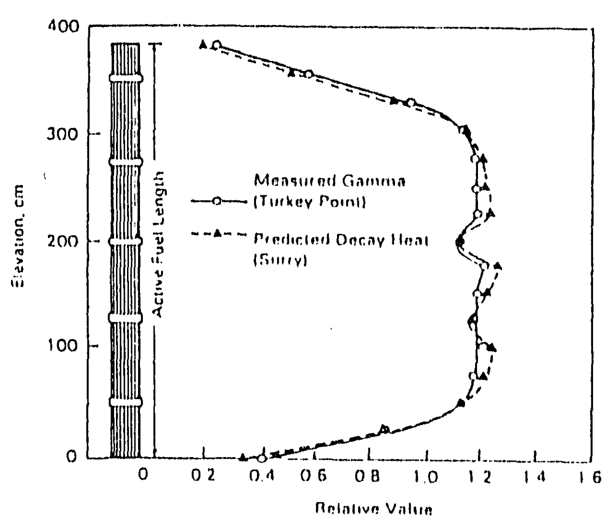

Fig. 2. Measured Gamma and Predicted Decay Heat Axial Profiles

Table 1. VSC-17 Cask Fuel Canister Composition and Loading Arrangement

Canister Location & ID	Assembly IDs	Assembly Source	Burnup GWd/MWT	Initial Enrich. %	Discharge Date	8/15/90 Cool Canister Time Years	Decay Heat, W
P15/A1/12	D01/D04	T-P	28.43	2.56	Nov-77	12.7	744
P01/A2/21	N05/N11	S-MC10	26.8/27.0	2.56	Apr-76	14.3	707
P06/A3/07	W10/W02	S-TN24P	29.80	3.20	Nov-81	8.8	956
P03/A4/16	N16/N35	S-MC10	26.82	2.56	Apr-76	14.3	704
P04/A5/24	R01/R15	S-MC10	35.44	3.10	Feb-79	11.5	1050
P05/A6/08	W52/W49	S-TN24P	29.99	3.20	Nov-81	8.8	959
P07/B1/10	D06/D15	T-P	28.4/27.9	2.56	Nov-77	12.7	744.20
P09/B5/17	R34/R35	S-MC10	35.33	3.10	Feb-79	11.5	1047
P08/B6/09	W38/W01	S-TN24P	29.99	3.20	Nov-81	8.8	962
P17/C1/11	D35/D40	T-P	28.43	2.56	Nov-77	12.7	744
P11/C2/15	N37/N17	S-MC10	27.04	2.56	Apr-76	14.3	710
P02/C3/06	W19/W16	S-TN24P	29.80	3.20	Nov-81	8.8	955
P10/C6/02	W44/W46	S-TN24P	29.99	3.20	Nov-81	8.8	962
P16/D1/13	D47/D46	T-P	28.43	2.56	Nov-77	12.7	744
P12/D3/04	W34/W27	S-TN24P	30.52	3.20	Nov-81	8.8	981
P14/D5/23	W09	S-MC10	28.28	3.20	Nov-81	8.8	970
	R41	S-MC10	35.33	3.10	Feb-79	11.5	
P13/D6/03	W28/W17	S-TN24P	29.99	3.20	Nov-81	8.8	9.62

distribution. ORIGEN2 was used with the assembly axial burnup distribution and the reactor operating history to determine the predicted axial decay heat profile shown in Figure 2. The axial decay heat profiles were smoothed for the heat transfer analysis. Axial decay heat profiles are important input to heat transfer computer codes because they strongly influence the shape of predicted axial fuel temperature profiles.

Pretest heat transfer predictions using the COBRA-SFS computer code [10-12] indicated that peak cladding temperatures in vacuum, nitrogen, and helium with open air cooling vents would be below or near 400, 394, and 315°C, respectively.

PERFORMANCE TESTING

The VSC-17 cask performance testing consisted of six test runs which included three internal environments and four vent blockage conditions. The cask orientation was vertical for all test runs. The first test run was representative of the normal storage/operating configuration for the cask with a helium cover gas and all air vents open. The next three test runs represented various vent blockage conditions: test run two had one-half the inlet vents blocked, test run three had all inlet vents blocked, and test run four had all of the inlet and outlet vents blocked. Test run five repeated test run one conditions but had a nitrogen cover gas instead of helium. Finally, test run six tested the cask with a vacuum environment with all the air cooling vents open. During performance testing, the cask was double pumped and double backfilled each time it was backfilled with a new gas or evacuated to a vacuum. Gas purity was verified through gas sampling and gas sample analysis.

Before testing with fuel in the cask, dry/cold runs (trial runs) of cask handling and fuel loading were performed. The objectives of the dry runs were to gain operational experience and to complete handling and test procedures. Each dry run was conducted successfully without unusual problems or significant modifications to the cask or handling equipment.

Instrumentation

A total of 95 thermocouples were used to instrument the cask and fuel. They were located as follows: eleven attached to the outer surface of the cask, four attached to the cask lid, two attached to the weather cover, ten imbedded in the concrete, nine each attached to the outside surface of the MSB and inner liner of the cask, one in each air vent, three to measure ambient temperature, and forty-two in seven TC lances. The TC lances were inserted through the cask lid into six fuel canister and one basket guide tubes. The thermocouple locations are shown in Figure 3.

Neutron/gamma radiation dose rate measurements were taken along the outside surface of the cask and across the top of the cask lid. EG&G Idaho Health Physics technicians conducted the surveys using portable instruments.

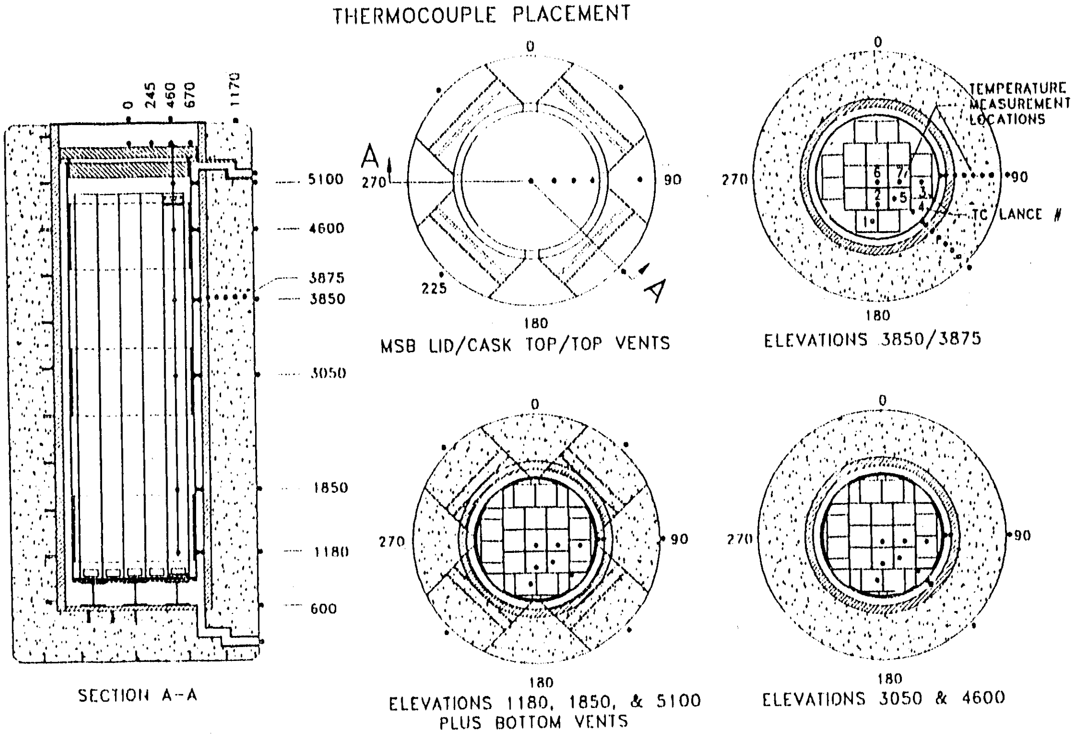

Fig. 3. Thermocouple Locations

Thermal Performance

The cask tests included assessments of performance with a full load of consolidated fuel (17 canisters), four ventilation blockage conditions, as well as vacuum, nitrogen, and helium backfill environments. Four to five days were required for the cask to reach steady state temperatures after initial conditions were established for each test run. The test matrix and corresponding measured peak guide tube temperatures and estimated peak cladding temperatures are presented in Table 2. Peak cladding temperatures were estimated using predictions from the COBRA-SFS computer code.

The data in Table 2 indicate that peak cladding temperatures for all fill gases and ventilation blockage conditions tested were less than 400°C. In general, the cask heat transfer performance was good. The peak concrete temperatures were less than 72°C and were fairly insensitive to partial blockage of the inlet vents. The peak fuel temperatures were less than 315°C for unblocked operation with a helium backfill and the 15 kW cask decay heat load.

Axial and radial temperature profiles for four of the six test runs are shown in Figures 4 and 5. Straight lines have been used to connect the data points to provide ease in interpreting the data and do not represent actual profiles. The axial profiles are for the hot center assembly, and the radial profiles are for the axial location corresponding with the TCs embedded in the concrete.

The axial profiles show the small effect of convection in nitrogen where peak temperatures are skewed toward the top of the cask. In the nitrogen case, convection moves the location of the peak axial temperature from an elevation of about 3.0 m (10 ft) to about 3.4 m (11.2 ft); in the helium case, the change is too small to differentiate.

The effect of convection on axial temperature profiles with consolidated fuel in the cask is much less than previously observed for casks loaded with unconsolidated fuel assemblies.[2,3,5] The consolidated fuel canisters are densely packed and have limited flow areas for the axial flow of gas compared to the open design of unconsolidated fuel assemblies. In general, the consolidated fuel canister blocks most of the gas flow through the basket. The high conductivity of helium masks what little convection may be occurring for the helium run. Only the vertical nitrogen run shows skewing of the axial temperature profile caused by convection.

Table 2. Peak Temperatures for the VSC-17 Cask Loaded with Consolidated Fuel

Run No.	Backfill/ Vent Blockage	Cask Heat Load kW	Amb. Temp. °C	Side Surf. Temp. °C	Est. Max. Conc. Temp. °C	Liner Temp. °C	MSB Surf. Temp. °C	Meas. Peak Fuel Temp. °C	Est. Peak Clad Temp. °C
1	He/None	14.9	21	37	69	82	136	316	321
2	He/Half Inlets	14.9	23	41	76	90	145	329	334
3	He/All Inlets	14.9	23	56	132	152	202	373	378
4	He/In & Outlets	14.9	22	56	141	161	212	376	381
5	Nitrogen/None	14.9	24	40	72	85	145	366	376
6	Vacuum/None	14.9	24	41	72	86	146	384	397

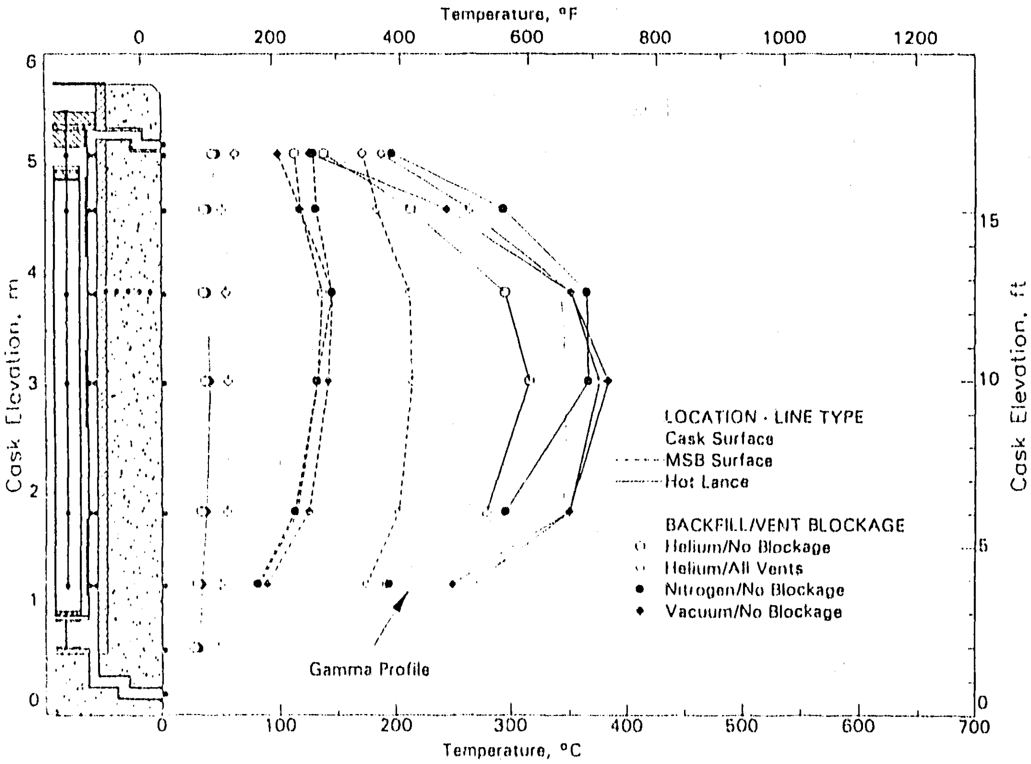

Fig. 4. Effect of Gas Environment and Vent Blockage on Axial Temperature Profiles

Radial temperature profiles for four of the six test runs are shown in Figure 5. At the hot elevation, significant temperature drops occur between the basket and the MSB wall, and between the MSB wall and the inner liner. Improving the heat path between the basket and the MSB wall could increase the heat load capacity of the cask if the heat load capacity is limited by the peak allowable fuel temperatures.

Temperature Predictions

The COBRA-SFS heat transfer code was used to predict temperatures in the cask. The code used a half-section cask model and a thirteen-node lumped-rod model of the consolidated fuel. COBRA-SFS used conservation of mass, momentum, and energy to calculate temperatures throughout the cask. The cask model includes conduction, convection, and radiation heat transfer.

The COBRA-SFS predictions of peak guide tube temperatures agree well with experimental data. The largest variation occurred for the nitrogen run, where a 15°C higher peak temperature was predicted. The mean difference between calculated and measured peak temperatures for the five test runs was less than 10°C. All five simulations either overpredicted the peak fuel temperature or were close to the temperature measurement uncertainty (± 4°C). Selected COBRA-SFS predictions compared to data are shown in Figures 6 and 7.

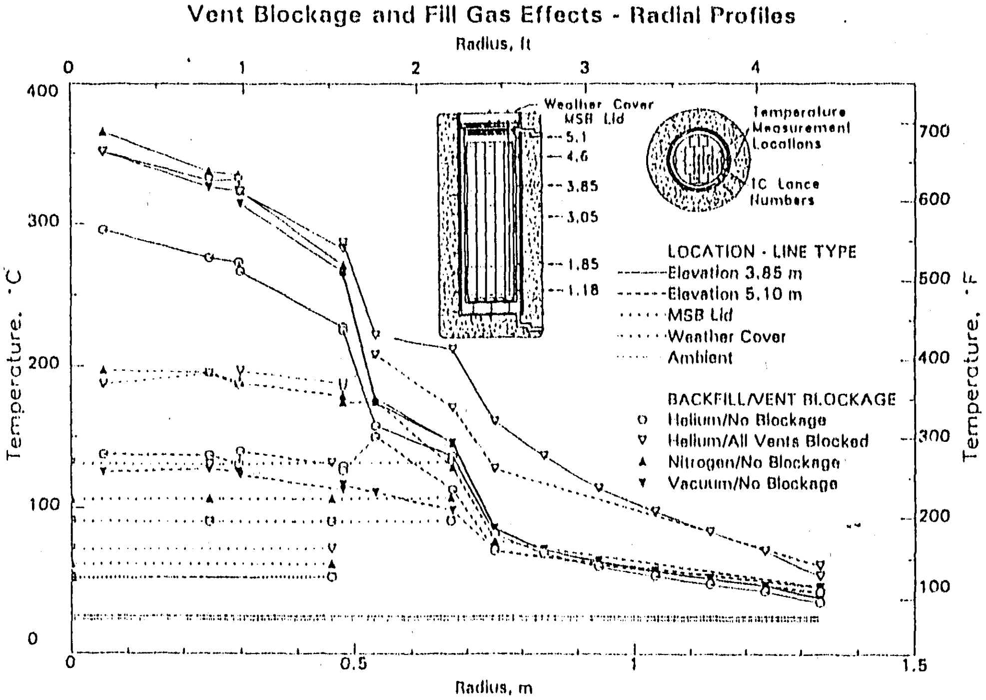

Fig. 5. Radial Temperature Profiles Measured Near Peak Axial Temperatures

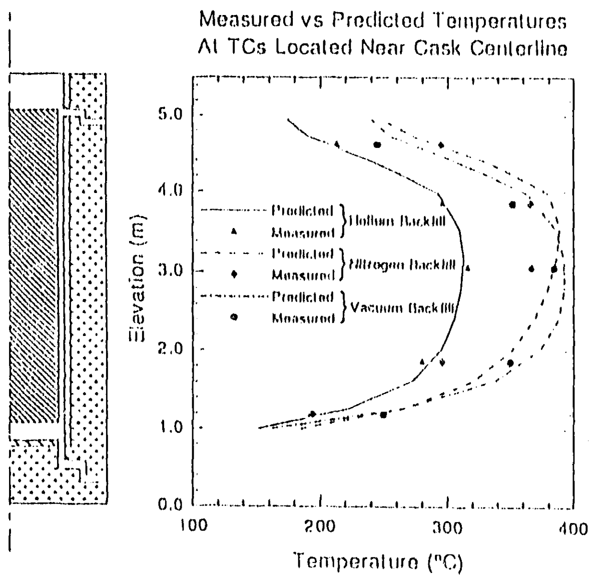

Fig. 6. Pretest Axial Temperature Profile Predictions Compared to Vacuum, Nitrogen, and Helium Data

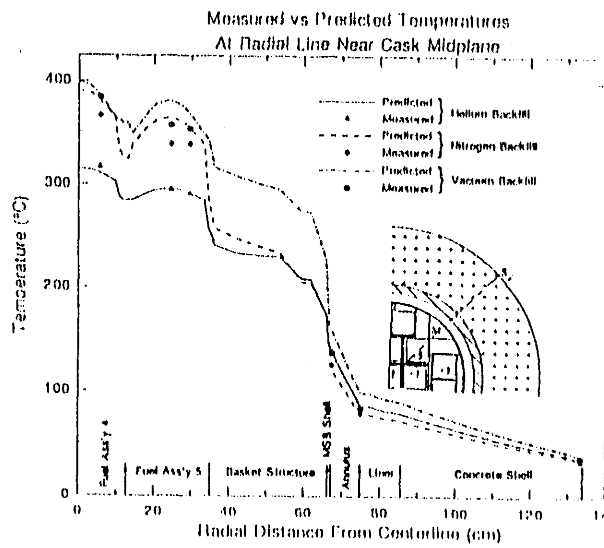

Fig. 7. Pretest Radial Temperature Profile Predictions Compared to Vacuum, Nitrogen, and Helium Data

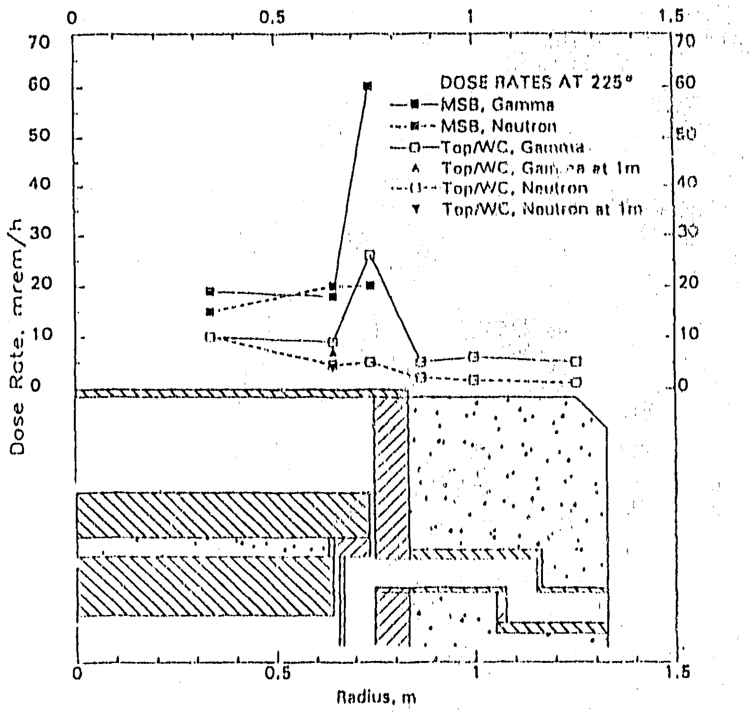

Fig. 8. Gamma and Neutron Dose Rate Profiles Measured on Cask Weather Cover

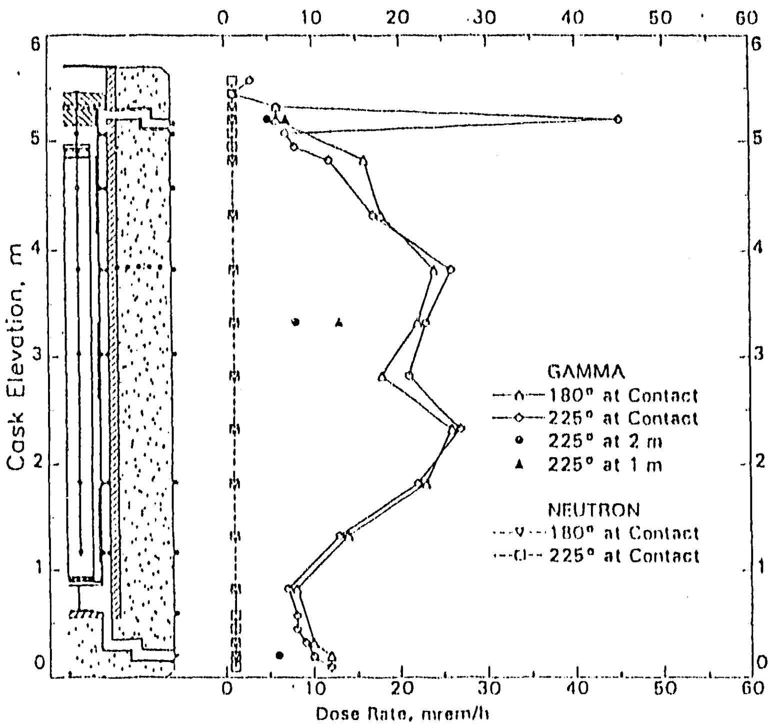

Fig. 9. Gamma and Neutron Dose Rate Profiles Measured on Cask Side

Shielding Performance

Gamma and neutron dose rates on the top and side of the cask are shown in Figures 8 and 9. These measurements were taken with portable survey instruments. Removal of the top and bottom nozzles and spacer grids from the fuel assemblies in the consolidation process removed most of the ^{60}Co gamma source.

The peak total dose rate on the top of the weather cover occurred near the edge of the MSB lid and was 50 mrem/h (40 mrem/h gamma and 10 mrem/h neutron). The total dose rates along most of the cask side were less than 30 mrem/h, predominately gamma (Figure 9). There was a localized radiation peak of 56 mrem/h (predominately gamma) corresponding to the center of the outlet vent. These peaks would disappear if additional gamma shielding was provided in these locations. No gamma peak was located at the inlet vents.

Use of unconsolidated fuel in the cask would increase gamma peaks observed at the edge of the MSB lid and at the outlet air vents. Tests conducted with the TN-24P cask [4,5] (one using unconsolidated fuel and one using consolidated fuel) indicates that removal of the non-fuel-bearing components during fuel consolidation decreased the contact dose rates at the top of the TN-24P cask by a factor of 4. If this same dose reduction factor is used with the VSC-17 cask, then a dose rate of 50 mrem/h for consolidated fuel would correspond to the cask design value of 200 mrem/h for unconsolidated fuel.

The overall shielding performance of the VSC-17 cask was good with consolidated fuel. A minor refinement in the gamma shielding design may be required to keep dose rates below 200 mrem/h for unconsolidated fuel.

Spent Fuel Integrity

The PWR spent fuel assemblies had been well characterized before testing. The results of these examinations indicated the potential of up to 13 leaking fuel rods before the VSC-17 cask performance test. Gas samples taken during the performance test indicate that no additional fuel rods developed leaks.

Operational Experience

The VSC-17 storage cask was loaded with consolidated fuel canisters transferred from the TN-24P cask. The cask-handling operations and consolidated fuel canister loading into the cask were performed without any difficulties attributed to the VSC-17 cask. Cask vacuum pump down required 0.5 to 1.0 hour to go from 85 kPa (12.24 psi) atmospheric pressure to 0.1 kPa (0.01 psi). Backfilling the cask with a cover gas required about 15 minutes. The spread of contamination was not a major problem during the fuel canister air transfers between the TN-24P and VSC-17 cask.

Personnel radiation exposures during the handling, loading, and testing of the VSC-17 cask were 40 mrem for TC lance installation and removal, 10 mrem for cask handling, and 10 mrem for testing. Radiation exposures under actual storage scenarios should be much lower than those encountered during the cask performance testing effort. Storage does not require the installation of TC lances used in the performance test.

During VSC-17 testing, unusual pressure behavior in the MSB was encountered. The pressure behavior was attributed to off-gassing of water vapor and hydrogen from the neutron shield material in the test lid. The TC lance penetration through the lid provided a path for the off-gassed material to enter the cask cavity. The commercial casks do not have the TC lance penetrations and this problem should not be present.

CONCLUSION

The cask performance test demonstrated that the VSC-17 cask could be satisfactorily handled and loaded dry. It was concluded that the heat transfer performance of the cask was good. Peak cladding temperatures with a helium backfill were significantly less than 340°C with a total cask heat load of 15 kW. The shielding performance of the cask met expectations for consolidated fuel. From both heat transfer and shielding perspectives, the VSC-17 cask can be effectively implemented at reactor sites and central storage facilities for safe storage of consolidated spent fuel. Storage of unconsolidated fuel may require some minor modification in gamma shielding design at the edge of the MSB lid and for the outlet vents to keep contact dose rates below 200 mrem/h.

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