ATTACHMENT I to VPN-042-96
TROJAN ISFSI SAFETY ANALYSIS REPORT
DESCRIPTION AND JUSTIFICATION OF CHANGES

PAGE	DESCRIPTION	JUSTIFICATION
1-6	Revised Section 1.5 to clarify reference to TranStor™ SAR.	The Trojan ISFSI basket and transfer cask are of the same design as the basket and transfer cask described in the TranStor™ SAR (Docket No. 71-9268). The text has been revised to clarify the association between the PGE and SNC SAR transmittals to the NRC.
2-7, 2-8, 2-10, 2-44, and 2-45	Deleted references to mileage and direction to surrounding towns and communities in Section(s) 2.1.4, 2.2.1, 2.6.2.3, and 2.6.2.4.	Section 2.1.1 adequately describes the surrounding towns/communities with regard to mileage and direction from the Trojan site. Restating the mileage and direction in subsequent sections does not provide useful information.
2-4 & 7-21	Revised Section 2.1.2.2 and Section 7.5.3.2.2 to correct the description of the boundary for the Restricted Area (10 CFR 20).	The Restricted Area was intended to coincide with the outer of the two fences around the ISFSI. The original text incorrectly stated that the Restricted Area boundary was the same as the Protected Area boundary, which is the inner fence. The description of the boundary for the Restricted Area has been changed to coincide with the Isolation Zone, which is the outer fence.
4-34	Revised Section 4.3.5 to delete the requirement for water to wash down the basket in the Fuel Building.	The original text stated that water was required to wash down the baskets prior to being placed in a cask. This wash down is not a requirement. Small amounts of water may be used to decontaminate the basket in the Cask Wash Pit if decontamination of the basket is determined to be necessary, but a wash down of each basket is not required.
Table 4.2-1a	Added table which defines ASME Code deviations	A table of ASME Code deviations was previously provided in a submittal separate from the ISFSI SAR with a commitment to incorporate the table of ASME Code deviations in the first revision of the SAR.

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DESCRIPTION AND JUSTIFICATION OF CHANGES

PAGE	DESCRIPTION	JUSTIFICATION
Table 4.2-2a	Added Table which defines ACI Code deviations.	A table of ACI Code deviations was previously provided in a submittal separate from the ISFSI SAR with a commitment to incorporate the table of ACI Code deviations in the first revision of the SAR.
5-4	Revised Section 5.1.1.2 to state that approximately 75 gallons of water is removed from the basket before welding the shield lid instead of 16 inches.	Water is drained from the basket to lower the water level approximately 5" from the bottom of the shield lid. This allows sufficient volume for water expansion due to heatup during the welding process. Specifying the number of gallons to be removed from the basket provides a directly measurable parameter that can be used to establish the desired air gap.
Appendix 1	Added non-proprietary drawings for major ISFSI components.	Non-proprietary drawings of the major ISFSI components were previously provided in a submittal separate from the ISFSI SAR with a commitment to incorporate the non-proprietary drawings in the first revision of the SAR.

INSTRUCTION SHEET

The following information is provided as a guide for the insertion of new sheets for changes to the "Trojan Independent Spent Fuel Storage Installation Safety Analysis Report," July 15, 1996 update.

Remove	Insert
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1 - Entire Section (retain figures)	1 - Entire Section
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	Drawings (after Chapter 11)
	PGE-001-Sheet 1/1-Rev 0 PWR Basket Assembly
	PGE-002-Sheet 1/1- Rev 0 Concrete Cask Assembly
	PGE-003-Sheet 1/1- Rev 0 GTCC Basket Assembly
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1.0 INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION

1.1 INTRODUCTION

The Trojan Nuclear Plant (TNP) was operated for approximately 17 years and was shut down for the last time on November 9, 1992. The plant is jointly owned by Portland General Electric Company (PGE), the City of Eugene through the Eugene Water and Electric Board, and Pacific Power and Light/PacifiCorp. PGE is the principal owner and has responsibility for maintaining TNP.

PGE's plans to decommission TNP include prompt decontamination and dismantlement of contaminated structures, systems and components. In order to facilitate decontamination and dismantlement, the contents of the Spent Fuel Pool will be relocated to an Independent Spent Fuel Storage Installation (ISFSI). Use of an ISFSI was determined to be the most economical method for the temporary storage of the TNP spent fuel until a DOE or other offsite facility is available. Relocation of the spent fuel, and other high-level radioactive waste stored in the pool to the ISFSI would allow decontamination and dismantlement of structures, systems, and components throughout TNP to proceed without impacting the safe storage of the spent fuel.

PGE plans to begin construction of the ISFSI following a favorable NRC assessment of the environmental impact. In order to support PGE's overall decommissioning schedule, construction of the ISFSI is planned to occur in 1997. PGE is requesting approval of the Environmental Report in August 1996.

PGE is requesting issuance of the 10 CFR Part 72 license in December 1997, and PGE intends to begin movement of fuel to the ISFSI in January 1998, which would allow greater than five years cooling period for the spent nuclear fuel. PGE is anticipating termination of the 10 CFR Part 50 license for TNP in 2001.

Activities involving loading of the spent nuclear fuel assemblies and Greater Than Class C (GTCC) waste into the storage basket, closure of the basket and placement of the basket into a storage cask are considered licensed activities per PGE's 10 CFR Part 50 license and are not described in detail in this Safety Analysis Report. This Safety Analysis Report is primarily focused on ISFSI operations, including ultimate transfer to a shipping cask.

TNP is located in Columbia County, Oregon, along the west bank of the Columbia River, approximately 42 miles north of Portland, Oregon. Figure 1.1-1 shows the location of TNP. The ISFSI will be located in the northeast portion of the TNP site, as shown in Figure 1.1-2.

1.2 GENERAL DESCRIPTION OF THE INSTALLATION

PGE has selected Sierra Nuclear Corporation's TranStor™ Storage System for the Trojan ISFSI.

The TranStor™ Storage System is a vertical dry storage system which utilizes a ventilated concrete storage cask and a seal-welded steel basket to safely store spent nuclear fuel assemblies, fuel debris, and GTCC waste material.

The ISFSI consists of a reinforced concrete pad, supporting a maximum of 36 Sierra Nuclear Corporation's TranStor™ Storage Systems. It is anticipated that 35 storage baskets and Concrete Casks will be required. Thirty three of the baskets are PWR Baskets, designed to safely store intact spent fuel assemblies, assemblies containing damaged fuel or fuel debris. Two of the baskets are GTCC Baskets, designed to safely store GTCC waste. The storage system is passive and requires minimal surveillances. Significant radioactive waste generation is not anticipated as a result of ISFSI operation.

The system is designed to permit transfer of the basket to a shipping cask once a repository or other facility is available. The TranStor™ shipping cask is being licensed separately. It is also designed to accommodate recovery from postulated off-normal events without reliance on the Spent Fuel Pool.

The principal design criteria are discussed in Chapter 3. Chapter 4 discusses the design of the ISFSI.

1.3 GENERAL SYSTEMS DESCRIPTION

The storage system consists of the baskets, Concrete Casks, Storage Pad and associated transfer equipment necessary for safe placement of spent nuclear fuel assemblies, fuel debris, and GTCC waste into dry storage. The following sections provide an overview of the primary components used during storage and transfer of spent fuel and GTCC waste. Figure 1.3-1 provides an overview of the basket and Concrete Cask.

1.3.1 STORAGE SYSTEM BASKETS

The Trojan ISFSI storage system utilizes two types of baskets, the PWR Basket and the GTCC Basket. The baskets are metal containers that are seal-welded closed. Both baskets serve as a confinement boundary for the materials stored within the baskets.

The PWR Basket is a fuel storage canister designed to provide safe storage of intact spent fuel, failed fuel and fuel debris. The PWR Basket consists of an internal sleeve assembly, an outer shell assembly, a shield lid and a structural lid. The internal sleeve assembly is fabricated from high strength steel plates formed into an array of 24 square storage sleeves, each holding one PWR spent fuel assembly. The cells are sized to accommodate storage of control components within the fuel assembly. The PWR Basket relies only on geometry for subcriticality during storage.

Assemblies containing damaged fuel are placed in a failed fuel can prior to placement in the PWR Basket. Fuel debris is placed in a Fuel Debris Can prior to placement in the PWR Basket. The four peripheral cells in each PWR Basket can accommodate either failed fuel cans or a fuel debris can as well as spent fuel assemblies.

The GTCC Basket is designed to provide safe storage of GTCC waste. GTCC waste is placed in canisters and then placed into the GTCC Basket. The GTCC Basket does not contain an internal sleeve assembly. The GTCC Basket accommodates 28 individual canisters designed for GTCC waste.

A Basket Overpack is provided to be used in the unlikely event of a leaking basket.

1.3.2 STORAGE SYSTEM CONCRETE CASK

The Concrete Cask provides structural support, shielding, and natural circulation cooling for the basket. The basket is stored in the central steel lined cavity of the Concrete Cask. The Concrete

Cask is ventilated by internal air flow paths which allow the decay heat to be removed by natural circulation around the metal basket wall. Air flow paths are formed by the skid channels at the bottom (air entrance), the air inlet ducts, the gap between the basket exterior and the Concrete Cask interior, and the air outlet ducts. Air outlet temperature is monitored to confirm proper decay heat removal.

The air inlet and outlet vents are steel lined penetrations that take non-planar paths to minimize radiation streaming. Side surface radiation dose rates are limited by the thick steel and concrete walls of the cask.

1.3.3 TRANSFER EQUIPMENT

1.3.3.1 Transfer Cask

The Transfer Cask is used to move the loaded basket from the Spent Fuel Pool to the Concrete Cask, to transfer a Basket to the Shipping Cask, or to a Basket Overpack in the unlikely event of a leaking basket. To use, the Transfer Cask is placed above the Storage Cask, the retractable doors at the bottom of the Transfer Cask are opened, and a loaded basket is hoisted into the Transfer Cask. The doors are closed and the loaded Transfer Cask is positioned above the destination Concrete or Shipping Cask, where the basket is lowered in a process reverse to the process described above.

1.3.3.2 Transfer Station

The Transfer Station is utilized for basket transfer operations at the ISFSI site. During transfer to a Shipping Cask, the Concrete Cask and Shipping Cask are placed in the Transfer Station. The Transfer Station provides a sliding collar and structural steel rails to prevent the loaded Transfer Cask from falling or overturning during transfer operations.

1.3.3.3 Auxiliary Systems

To move a loaded Concrete Cask from one location to another, an air pad system is used. Air pads are inserted under the cask and energized with a standard air compressor. A forklift or other small truck can then be used to move the Concrete Cask.

1.3.4 AUXILIARY EQUIPMENT

1.3.4.1 Vacuum Drying System

A skid-mounted vacuum drying system is used to remove the water from the basket (following fuel loading), dry the basket, and backfill the basket with helium. The vacuum drying system is designed to evacuate the basket in a stepwise fashion. During evacuation, the decay heat from the fuel further helps remove residual moisture from the basket.

1.3.4.2 Semi-Automatic Welding System

The baskets are seal-welded using a semi-automatic welding system.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

PGE is the principal owner of the Trojan ISFSI and is responsible for operation, maintenance and surveillance of the ISFSI. PGE is also responsible for overall project management.

Sierra Nuclear Corporation is responsible for design and fabrication activities including quality assurance services.

1.5 MATERIAL INCORPORATED BY REFERENCE

The PWR Basket and Transfer Cask are the same components as described in the Sierra Nuclear Corporation Safety Analysis Report for the TranStor™ Shipping Cask System, Docket No. 71-9268, submitted to the NRC on December 20, 1995. PGE intends to register as a user of the TranStor™ shipping cask once a certification is issued.

As discussed in Chapter 11, the Sierra Nuclear Corporation Quality Assurance Program for the Ventilated Storage Cask System, Pacific Sierra Nuclear Associates and Sierra Nuclear Corporation, October, 1991. Docket Number 72-1007 is incorporated by reference into this document.

2.0 SITE CHARACTERISTICS

This chapter discusses the general characteristics of the ISFSI site and vicinity as they relate to the area's geology, seismology, hydrology, and meteorology. Population distribution, land use, and site activities and controls are also discussed. This chapter presents, in complement with more detailed discussions provided in other ISFSI SAR chapters, information showing the overall adequacy of the site for storage of spent nuclear fuel and GTCC waste.

2.1 GEOGRAPHY AND DEMOGRAPHY

The ISFSI is located at the Trojan Nuclear Plant site. The Trojan Nuclear Plant site was originally selected to minimize hazards to the general public from the operation of Trojan Nuclear Plant. The site environs have low population densities and minimal usage for such activities as farming and recreation. Some of the site characteristics associated with selection of the site for operation of the Trojan Nuclear Plant remain applicable to the storage of spent nuclear fuel and GTCC waste. This chapter provides discussion of those site characteristics applicable to storage of spent nuclear fuel and GTCC waste.

2.1.1 SITE LOCATION

The ISFSI site is in Columbia County, Oregon, and lies along the west bank of the Columbia River at approximately River Mile 72.5, 42 miles north of Portland. The specific geographic location of the site is 46° 02' 25" N latitude and 122° 53' 03" W longitude. In the Universal Transverse Mercator coordinate system, the site location is 5098352 meters N by 509000 meters E, and in the Oregon North Zone Lambert Coordinate, the site location is 874375 N by 1394615 E.

The nearest incorporated communities are Rainier, Oregon, approximately 4-1/2 miles northwest; and across the Columbia River in Washington, Kalama, approximately 3 miles southeast, and Longview, approximately 6 miles northwest. Within a 5-mile radius of the site are three small unincorporated communities with a total population of less than 2000: Prescott, Oregon, 1/2 mile north; Goble, Oregon, 1-1/2 miles south; and across the Columbia River, Carrolls, Washington, 2-1/2 miles northeast.

Other than the Columbia River and tributaries, there are no nearby natural geographic features of prominence offsite. The Kalama River joins the Columbia at River Mile 73.1, about 1/2 mile upstream on the bank opposite the site. Similarly, the confluence of the Cowlitz and Columbia Rivers is about 4-1/2 miles downstream at River Mile 68. Manmade features include the 492-foot natural draft cooling tower, which rises 589 feet above mean sea level (MSL), an approximately 26-acre man-made reflecting lake, and an approximately 28-acre recreational lake.

Figures 1.1-1 and 2.1-1 are maps that show the Trojan Nuclear Plant and ISFSI site location. Figure 2.1-2 is a map that shows the PGE property, the ISFSI site, surrounding topography, and the Controlled Area as defined in 10 CFR 72.106. Figure 2.1-3 shows the ISFSI site layout.

2.1.2 SITE DESCRIPTION

The ISFSI is located on an approximately 634-acre tract of land owned in fee by Portland General Electric Company (PGE) in Sec. 35 and 36, T. 7 N., R. 2 W., W.M., and in Sec. 1 and 2, T. 6 N., R. 2 W., W.M., Columbia County, Oregon. The tract is all-inclusive of individual and separate parcels as described in the following deed records on file in Columbia County: BK 168, Pages 13 and 14, 22, 23 to 26 inclusive, 81 to 83 inclusive, 117 to 121 inclusive; BK 171, Pages 935 and 936; and BK 174, Page 436.

The eastern boundary of the PGE property is the Columbia River. The eastern boundary (owned in fee) extends to mean low water in the southern part of the property and to mean high water in the northern part of the property. By written agreement with the State of Oregon, who is owner of the submerged and remainder of submersible lands in the river, PGE has control of the uses of such areas out to a line at approximately -20 feet MSL. Beyond this line the U. S. Coast Guard has jurisdiction over river operations.

The western boundary of the PGE property is U.S. Highway 30 in the northern one-third of the property. The western boundary in the southern two-thirds of the property extends past (to the west of) U.S. Highway 30 and includes several parcels of land to the west of U.S. Highway 30.

The ISFSI reinforced concrete pad, which is approximately 100 feet by 170 feet and is designed to accommodate up to 36 storage casks, is located inside the ISFSI Protected Area fence near the eastern edge of the PGE property. The ISFSI Protected Area fence represents the boundary of the ISFSI "site" within which 10 CFR 72 activities will be licensed and will occur.

Transportation routes which are in the immediate vicinity of the ISFSI site include the Columbia River, U.S. Highway 30, and Burlington Northern Railroad. The nearest edge of the ISFSI reinforced concrete pad is about 160 feet from the Oregon bank (mean low water) of the Columbia River, about 1/2 mile from the U.S. Highway 30 right of way, and about 700 feet from the railway right-of-way.

Approximately six oceangoing commercial vessels pass the ISFSI site on the Columbia River in a typical day (Reference 1). U.S. Highway 30 is a two-lane roadway that carries moderate passenger and freight traffic between communities along the Columbia River. Burlington Northern Railroad operates an average of two freight trains per day along their railway right-of-way which traverses the PGE property (Reference 2). Railroad property within PGE property boundaries is "operating" property, i.e., not available for lease or other use.

Four 230kV overhead transmission lines terminate in a switchyard approximately 1000 feet from the ISFSI. The switchyard supplies power to the ISFSI site.

The Controlled Area, as defined in 10 CFR 72.106, immediately surrounds the ISFSI and extends out to 325 meters from the edge of the storage pad (Figure 2.1-2). The Controlled Area lies entirely on PGE property with the exception of a portion of the Controlled Area that extends over the Columbia River and the Burlington Northern Railroad right-of-way. U.S. Highway 30 is not within the Controlled Area. PGE has formal agreements with Burlington Northern Railroad to restrict traffic over their right-of-way, with the U.S. Coast Guard to restrict traffic on the Columbia River, and with the state of Oregon to evacuate persons from publicly owned lands (i.e., tidelands) in the event of an emergency at the ISFSI.

The doses that could be anticipated at the Controlled Area boundary from an off-normal event or accident are discussed in Chapter 8 and are below the limits of 10 CFR 72.106 and Oregon Administrative Rule (OAR) 345-26-390.

2.1.2.1 Other Activities Within the ISFSI Site Boundary

No activities unrelated to ISFSI operation are performed within the ISFSI site boundary.

Several major physical facilities, which were used during Trojan Nuclear Plant operation, are grouped to the south and west of the ISFSI site. These facilities are outside the ISFSI Protected Area (ISFSI site boundary) and are intended to be made available for commercial activities upon their release for unrestricted use. Leases issued to commercial users of these facilities will limit activities to ensure that postulated events and accident analyses remain bounding. Access to these facilities will not afford access to the ISFSI.

2.1.2.2 Boundaries for Establishing Effluent Release Limits

The only potential effluent release points are the storage casks themselves located at the ISFSI. No effluents are anticipated for normal and anticipated occurrences at the ISFSI. In addition, no credible accidents or off-normal events result in effluents. The dose resulting from normal operation and anticipated occurrences (i.e., direct radiation) has been estimated at the Controlled Area boundary and is within the limits specified in 10 CFR 72.104 and OAR 345-26-390.

The Restricted Area. as defined in 10 CFR 20, has the same boundaries as the isolation zone that surrounds the ISFSI Protected Area. Physical access to the isolation zone is restricted by the debris fence. Access into the Protected Area is controlled as described in the ISFSI Security Plan. Radiation protection procedures specify when dosimetry is required in the Restricted Area.

The minimum distance from any effluent release point (storage casks) to the Restricted Area boundary is approximately 40 feet. If members of the public have access to the Controlled Area immediately outside the Restricted Area, then the dose to a member of the general public in this area will be shown to comply with the limits of 10 CFR 20.1301.

Recreational uses within the PGE property boundaries include hiking, picnicking, swimming, fishing, and nature observation. In the event of an emergency that could result in a hazard to the general public, members of the general public making recreational or other casual use of the nonrestricted portions of the PGE property or making commercial use of the buildings on the PGE property can be evacuated.

2.1.3 POPULATION DISTRIBUTION AND TRENDS

The 1990 population distribution within 10 miles of the site, shown in Figure 2.1-4, was derived using 1990 census values (Reference 3). Specific place populations were located within the appropriate sectors. Rural population groups were distributed on the basis of the density of roads within each sector.

The population projections for 2000 and 2010, shown in Figures 2.1-5 and 2.1-6, were made using county growth projections based upon three census data points: 1970, 1980, and 1990 (Reference 3). Individual growth projections were developed for Cowlitz and Columbia Counties. Based upon these historical factors, population growth within 10 miles of the site is about 5 percent per decade.

In addition to the resident population, a limited influx of people into the area of the site occurs when river conditions are conducive to fishing and recreation. This influx is primarily on the Columbia and Kalama Rivers and consists of pleasure boaters, boat fishermen and bank fishermen (Reference 5). The Oregon Department of Fish and Wildlife performed aerial surveys of the river from February to October 1995 and estimated that there were 15,335 angler trips on the river from Longview to Prescott (about 6 miles) and 17,236 angler trips on the river from Prescott to Martin Slough (about 9 miles). During the busiest month, September, there were 4,556 angler trips on the river from Longview to Prescott and 6,279 angler trips on the river from Prescott to Martin Slough. Using these estimates, there would be about 80 anglers per day within 5 miles of the ISFSI from February to October. During September, the month of highest utilization, there would be about 241 anglers per day within 5 miles of the site. Because there are no state or federal parks or campgrounds within 10 miles of the site, any increase in the number of people in the area during the summer months is relatively small.

Public facilities and institutions near the site are listed on Table 2.1-1.

2.1.4 USES OF ADJACENT LANDS AND WATERS

The ISFSI site lies in a heavily timbered area, characterized by rough terrain and suited primarily to logging and other forestry operations. One major population center lies within a 50 mile radius, as do several smaller cities; most heavy industry in the smaller cities is related to forest product processing or agriculture (Reference 6). Well over half of the land is suitable for commercial forestry or grazing, with about 20 percent suitable for farming. Less than 10 percent of the land area is unsuitable for any agricultural pursuit, and a fraction of 1 percent is devoted to urban or incorporated areas.

Lands adjacent to the site lie within Columbia County, Oregon, in which the site is located, and Cowlitz County, Washington, across the Columbia River. The area within a radius of 10 miles of the site lies within these two counties. Both have agriculturally based economies, with land use in the vicinity of the site primarily agricultural. Logs, hay and other feed are the predominant crops. Salient agricultural data for these two counties is indicative of small, generalized family farming, with heavy emphasis on grazing and farm animals. Only 41 of 934, less than 5 percent, of the farms in the counties are larger than 100 acres, while more than one-third have fewer than 20 acres. There are no major milk-producing centers on lands adjacent to the ISFSI site, the major milksheds being 50 or more miles distant (Reference 7).

A land use census completed in 1994 indicated that there were milk cows within five miles of the ISFSI (Reference 8). There were several milk producing goats located within five miles of the site during 1994. Milk from these goats was sampled as part of a Trojan radiological environmental surveillance plan associated with the Trojan Nuclear Plant. The 1994 land use census also surveyed the locations of beef cattle and other meat producing animals as well as vegetable gardens within 5 miles of the ISFSI site. The results of the 1994 land use census are shown in Table 2.1-2.

Other than agriculture, the industrial base of the area around the ISFSI site is centered in forest products and primary metals, and most of the industrial activity is on the other side of the Columbia River. Of the eleven Longview-Kelso industrial facilities listed in Table 2.2-1, four are forest products processors producing lumber, plywood, pulp, paper and paper products, and related wood and wood pulp products. The other large manufacturing firm is an aluminum smelter with an annual capacity of 220,000 metric tons. A small steel furnace smelter and pleasure boat manufacturer are the only other major manufacturers in this area. Near Kalama, upstream of the site, are grain elevators, chemical plants, a steel mill under construction, and a few small mills.

There is relatively little recreational land use within the immediate area of the site. There are no State or Federal parks nearby, nor are there any natural or man-made attractions such as mountains or reservoirs (References 9 and 10).

The 26-acre reflecting lake and 28-acre recreational lake located on PGE property are accessible to the general public from the property entrance road. Fishing activity peaks in the spring (about 30 fisherman per day) when the state of Oregon stocks the recreational lake, and then the fishing activity tapers off to a couple of fisherman per day.

A 3-mile portion of U.S. Highway 30 has been designated as a scenic area by the Oregon State Scenic Area Board (Reference 11). Pleasure boat launchings are located in Rainier, Goble, and Prescott, (Reference 12). Recreational vehicle overnight parking is available in Goble, and Prescott Beach is used for camping and fishing. River access is also available on the Washington shore, directly opposite the site.

The lower Columbia River is well suited to recreational fishing and boating, most of which occurs in the 7 months from March to September. As described in Section 2.1.3, the Oregon Department of Fish and Wildlife performed aerial surveys of the river from February to October 1995. From their estimates, there will be about 8 anglers per day per river mile near the ISFSI from February to October. The heaviest concentration of anglers on the river near the ISFSI will be about 24 anglers per day per river mile in September.

Commercial fishing in the Columbia is regulated by both Oregon and Washington. About 270 miles of the Columbia River and tributaries are open to commercial fishing, with Bonneville Dam being the approximate midpoint. The fishery upriver of Bonneville is reserved for Indians only, while downstream is open to commercial fishing license holders as well

The Columbia River is a major navigable channel. Approximately 2300 seagoing cargo vessels pass the site annually, carrying wheat and logs outbound and manufactured iron goods, ores and petroleum inbound (Reference 1). Major port facilities are at Portland, Oregon and Longview, Washington.

2.2 NEARBY INDUSTRIAL, TRANSPORTATION AND MILITARY FACILITIES

Potential accidents as a result of external activities in the vicinity of the ISFSI site have been studied to determine their effect on the safety of the ISFSI. This section outlines the activities of the nearby industrial facilities, transportation arterials, and military installations and their potential effects on ISFSI safety. The risk to the operation of the ISFSI resulting from these activities is shown to be minimal.

2.2.1 LOCATIONS AND ROUTES

Most of the local commerce is related to forest products and is centered in Longview, Washington; Rainier, Oregon; and Kalama, Washington.

Due to the emphasis on forest products, industrial development in the area is heavily oriented to river transportation. An aluminum plant, small smelter, and boat manufacturer in Longview, a steel mill (under construction), chemical plants, and grain elevators in Kalama, and a fertilizer plant in Columbia City are the only large industries not related to the timber or paper industry. There are also several small quarry sites and gravel pits in the area, the closest being in Goble.

Transportation routes consist of two major highways, two railroads, the Columbia River and an airport and airways. U.S. Highway 30 runs north-south adjacent to the PGE property boundary approximately 1/2 mile from the ISFSI, and is a light-duty, two-lane highway connecting Portland on the south to Astoria, at the mouth of the Columbia River. Interstate 5 (I-5) is part of the West Coast north-south interstate system extending from Mexico to Canada. I-5 in this area is across the Columbia River in Washington approximately 1-3/10 miles east of the ISFSI at its nearest point. The Burlington Northern Railroad right-of-way passes through the PGE property, approximately 700 feet from the ISFSI. The main line railroad track between Portland and Seattle is located across the Columbia River in Washington, approximately 1-1/10 miles from the ISFSI.

The Columbia River serves as the deep-sea access channel to the important ports of Portland, Oregon and Vancouver, Washington. A 40-foot channel is maintained for deep-draft ocean vessels as far upriver as Portland. The center line of the 600-foot wide ship channel is approximately 3/10 mile from the ISFSI. Upstream from Portland and Vancouver, a 17-foot channel is maintained for barge traffic, extending to Pasco, Washington and a distance into the Snake River (Reference 3). Locks are provided at each of the dams on the river coincident with the 17-foot channel (Reference 4).

About 2300 oceangoing ships a year pass by the ISFSI site on the Columbia River. The major portion of the cargo exported is wheat and logs. Inbound cargo consists of miscellaneous goods such as petroleum, iron and steel products, automobiles, and ores (Reference 1). Portland is one of the largest ports in terms of tonnage on the Pacific Coast and thus it maintains a large number of supporting facilities.

Longview, Washington, downstream of the site also has facilities for oceangoing ships. The Port of Longview maintains facilities for unloading and storage of ship cargo. Significant facilities are a bulk loader with storage for 14,000 metric tons of talc; storage tanks with capacity for 40,000 tons of calcinated coke; a grain elevator, currently not in use, with a capacity of 7.8 million bushels; and log storage yards. Among the commodities routinely stored at the port are pencil pitch (or coaltar pitch), ammonia sulfate, and potash. Additionally, at the port Wilson Oil (doing business as Wilcox & Flegel) operates a petroleum bulk plant which has 14 storage tanks with a total capacity of 26,190 barrels of storage (Reference 2).

The Kelso-Longview Airport is 5.3-miles north of the site and has a 4,391-foot paved runway oriented northwest-southeast. The airport is not a scheduled airline stop, but is the base for approximately 60 single and twin-engine, private and corporate aircraft. The airport handles about 100 takeoffs and landings per day. The largest planes using the field are a DeHavilland 8 corporate plane, a Siddely Hawker, a Cessna Citation, and a Falcon Jet (Reference 5). The Portland International Airport is located 33 statute miles south of the site, and is the only major airport within a 60-mile radius of the site. Portland inbound and outbound air traffic is controlled for a distance of 30 miles from the airport by Portland Air Traffic Control. Area-wide in-flight traffic control is regulated by Seattle Air Traffic Control (Reference 6).

There are no major military bases in the vicinity of the ISFSI site. The nearest military facilities are Reserve Headquarters for the various branches in Portland and Vancouver (30-40 miles south of the site), and Coast Guard and Naval facilities in Portland, Longview and at the mouth of the Columbia River (Reference 7).

A natural gas main extending to Wauna, Oregon, downriver of the site, runs along the hillside west of the site, approximately 1-1/2 miles from the site. The main is a 16 inch, 3-million foot^3/hour line, buried a minimum of 3 feet (Reference 8). In addition, there is an odorizer station on the line at Goble, a river crossing at Deer Island, 4-1/2 miles south of the site, and a river crossing at Rainier.

U.S. Highway 30 provides highway access to the ISFSI site and serves as the traffic arterial between Portland and the communities on the Oregon bank of the Columbia River, carrying an average of 5300 vehicles per day (Reference 9). The highway runs through the communities of Scappoose, Warren, St. Helens, Columbia City, Deer Island and Goble, south of the site; and Rainier, Clatskanie, Westport and Astoria north and west of the site. A bridge at Rainier connects U.S. Highway 30 with Longview, Washington, and a bridge at Astoria, the western terminus of the highway, connects to Megler, Washington.

U.S. Highway 26 provides a shorter Portland-to-Astoria route; thus it carries the bulk of traffic between the two, leaving U.S. Highway 30 to carry local passenger traffic, log trucks, tourists, farm vehicles and truck deliveries to the river communities. There is some shipment of petroleum products via U.S. Highway 30. Gasoline, diesel and heating oils in tank trucks are regularly delivered to towns beyond the site from suppliers in Portland and St. Helens.

Interstate 5 is the primary north-south traffic route between Portland and the Puget Sound area (Seattle, Tacoma, Olympia) carrying an average of approximately 46,000 vehicles per day. Of this total, approximately 20 percent is made up of truck combinations and the remaining 80 percent is passenger traffic (Reference 10). It is estimated that about one-tenth of the truck traffic could be carrying flammable or hazardous material, of which petroleum products would make up the majority.

An average of two freight trains per day pass through the PGE property on the Burlington Northern Railroad right-of-way, carrying general commodities, with an annual gross tonnage of 6 million tons (Reference 11). Lumber and forest products make up the bulk of the shipping most of the year. During the peak fishing season, some canned and frozen seafood is carried by rail from the Astoria canneries. An average of about 200 shipments per year with 2-3 cars per shipment of chlorine and caustics are shipped to the James River Corporation in Wauna, Oregon, on the lower river via the Burlington Northern line. Other chemicals shipped include preservatives, fertilizer, resins and paints and a small amount of petroleum and propane.

Three railroads use the tracks on the Washington side of the river: Burlington Northern, AMTRAK, and Union Pacific railroads. Thirty-five to forty freight trains and six passenger trains pass the ISFSI site per day on these tracks (Reference 12). The freight carried varies widely with large quantities of wood products, aluminum, paper products, grains, agricultural products and foodstuffs making up the bulk. Chemicals shipped include large quantities of fertilizers, phenols, caustics, propane and various resins, acids, paints and lumber treatments.

Sharply rising ground to the west and similar high ground across the river to the east provide natural barriers for the site. The ISFSI itself is afforded additional protection on the north and east by earthen berms approximately 50 feet high and on the south and west by the buildings ranging from approximately 30 to 100 feet high.

The highest manmade structure at the site is the cooling tower, rising 492 feet above ground level. The cooling tower is marked with lights in accordance with Federal Aviation Administration regulations.

2.2.2 DESCRIPTION OF PRODUCTS AND MATERIALS

Products and byproducts of the timber industry in the area range from unfinished timber to finished construction lumber, cabinetry, plywood and veneer. Some hardwood products are made in Longview on a small-scale operation, while paper and wood fiber products make up a large percentage of the production of the area. Some chemical use and storage is associated with these industries. Chemicals include resins used in plywood, veneer and chipboard production, acids used in paper and pulp production, and lumber pressure treatments and finish coatings (stains and varnishes). Chemicals are stored either in tank cars on sidings, or in storage tanks connected to the industry involved (Reference 13).

The aluminum plant in Longview is an aluminum reduction facility operated by Reynolds Metal Company which produces raw metal in the form of ingots, billet bars, etc. The use of chemicals at this plant correspond to that of any aluminum plant; namely coke, pitch, chlorine and liquefied nitrogen. Chemical storage facilities at the plant consist of stockpiles, tanks and rail tankers and transportation is by rail tank cars (Reference 13).

Kalama Chemical, Inc., produces phenols with some secondary production of benzoates. The facility receives its raw material, toluene from tankers and stores it in an 80,000-barrel tank. The finished product is shipped by rail tank car (Reference 13).

Hoechst Celanese Corporation, Inc., is located approximately 3-miles southeast of the ISFSI in Kalama, Washington and produces a bleaching agent used in the pulp and paper industry. The facility receives sulfur dioxide by rail tank car and has a storage capacity for this chemical of 300,000 pounds.

All Pure Chemical Company is located approximately 2-miles southeast of the ISFSI in Kalama, Washington. The company produces a number of products including sodium hypochlorite, household ammonia, and water treatment chemicals. It is involved in the repackaging and distribution of chlorine gas. The chlorine gas is received in 90-ton rail tank cars and is repackaged into 1-ton cylinders. The 90-ton rail tank car is the maximum storage capacity for the chlorine gas at the facility.

A listing of nearby industrial facilities, supplementing the summarization above, is provided as Table 2.2-1. The geographic locations of the nearby industrial facilities are shown on Figures 2.2-1 and 2.2-2.

2.2.3 EVALUATION OF POTENTIAL ACCIDENTS

This section provides an evaluation of the capability of the ISFSI to safely withstand the effects of an accident at, or as a result of the presence of, industrial, transportation and military installations or operations within 5 miles of the site. Potential accidents considered include explosions of chemicals, flammable (including natural) gases or munitions; industrial and forest fires; accidental releases of toxic gases; and collapse of the cooling tower.

2.2.3.1 Explosions

Shipments of commercial cargo past the site create the possibility of nearby explosions. For the most part, the rugged construction of the storage casks would protect the spent nuclear fuel and GTCC waste from such explosions. In addition, the ISFSI would be shielded from the direct force of these explosions by the earthen berms on the north and east and by the manmade structures and buildings to the south and west.

Explosions unrelated to transportation are not considered significant. The quarry operations south of the site are located in the hills west of the Columbia River. Presently, there is no storage of explosives at the operating quarry which is 2 miles from the site. The quarry is not a large operation and only a limited amount of explosives are used. Because of the distance from the site and the protection afforded by the hillside and ridge between the quarry and the site, the quarry operation does not present a hazard to the safety of the ISFSI. The natural gas main runs along the hillside west of the site, approximately 1-1/2 miles from the site. The operation of this line will not present a hazard to the ISFSI from explosion because of the relatively low explosive capacity of the gas and the distance from the ISFSI.

Explosions related to transportation were extensively analyzed for siting of the Trojan Nuclear plant (same location as the ISFSI). The explosion analysis, which addressed rail, ship, and highway transportation, was described in detail in the Trojan Final Safety Analysis Report (FSAR).

The FSAR analysis used an overpressure limit of 2.2 psi. This overpressure is the maximum overpressure that can be generated by the atmospheric shock from an explosion. As described in Section 8.2.8.2, the storage cask is able to withstand tornado wind pressure up to 2.3 psi and wind pressure as high as 5.8 psi without sliding or overturning.

The FSAR analysis assumed transportation accident rates and numbers of shipments past the site based on estimates from transportation agencies and companies. These agencies and companies were contacted to confirm that the original estimates were still valid for the ISFSI.

The minimum weight of explosives that could cause a 2.2 psi overpressure was calculated by the FSAR analysis as 70,000 (pounds of TNT equivalent). This weight was originally confirmed for Trojan Nuclear Plant operation to exceed any known or planned shipments and has been reconfirmed for ISFSI operation.

In addition, the FSAR analysis calculated that the probabilities of a rail or barge shipment explosion that would cause a 2.2 psi overpressure are less than 10^{-6} per year each. These probabilities would be similar for the ISFSI because the transportation estimates have not changed appreciably.

Therefore, transportation related explosions would not affect the safe storage of spent nuclear fuel and GTCC waste.

2.2.3.2 <u>Toxic Chemicals</u>

The effects of toxic chemicals on human habitability were extensively analyzed for operation of the Trojan Nuclear Plant and addressed in detail in the FSAR. These analyses were predicated on maintaining control room habitability during a toxic gas event. Continuous manning of the ISFSI for operational reasons is not required as in the case for an operating nuclear plant. There are no off-normal events or credible accidents for the ISFSI that require operator action within a prescribed amount of time.

Therefore, a toxic gas event would not affect the safe storage of spent nuclear fuel and GTCC waste.

2.2.3.3 Fires

The ISFSI does not require automatic suppression and detection systems because the site specific fire hazards will not exceed the design temperature limits of the storage casks. The fire main, which was installed for 10 CFR 50 fire protection requirements, may be operable for general property insurance requirements of the surrounding buildings, but the fire main is not required or credited for ISFSI fire protection.

Industries and oil storage facilities in the vicinity of the ISFSI are separated from the ISFSI, either by considerable distance or by the Columbia River. Therefore, fires at these facilities would not pose a hazard to the ISFSI.

Fires resulting from transportation accidents on I-5, the railway near I-5, or the Columbia River would be separated from the ISFSI by considerable distance and the Columbia River. Fires from transportation accidents on Highway 30 would be separated from the ISFSI by the recreation lake and reflecting lake. Fires from transportation accidents on the Burlington Northern railway would be sufficiently far from the ISFSI to not have an effect on the ISFSI. Therefore, fires from transportation related accidents do not pose a hazard to the ISFSI.

The ISFSI is protected from brush or forest fires on two sides by water, the Columbia River to the east and the recreation lake, reflecting lake and Whistling Swan area to the west. The ISFSI is also afforded localized fire protection by the open area immediately surrounding it.

A fire caused by a rupture of the natural gas main west of the ISFSI would be separated from the ISFSI by a considerable distance and by the intervening lake areas. There is the possibility in the future that a natural gas line will be placed in the vicinity of ISFSI to supply gas turbines used to produce electrical power. The potential hazards posed by placing a natural gas line and gas turbine in the vicinity of the ISFSI are addressed in Chapter 8.

In addition to the natural barriers, the Rainier Rural Fire Protection District provides fire protection services for the site.

A fire caused by a diesel fuel oil spill from a mobile crane or other diesel fuel oil tank at the ISFSI or in the immediate vicinity of the ISFSI would not affect the safe storage of spent nuclear fuel and GTCC waste. This type of fire, which is the only credible fire because of the limited number of fire hazards located at the ISFSI itself, would burn for only a few (6-7) minutes. This short burn time would not be sufficient for much heat transfer to the storage cask or basket and the temperatures of the storage cask and basket would not be appreciably raised.

The consequences of a forklift fuel (propane) tank explosion and fire are bounded by the diesel fuel oil spill scenario.

Therefore, fires would not affect the safe storage of spent nuclear fuel and GTCC waste.

2.2.3.4 Aircraft Impacts

An analysis was performed that followed the guidance of NUREG-0800, Section 3.5.1.6, Aircraft Hazards. The analysis demonstrated that the probability of aircraft impacting the ISFSI is less than 10^{-7} per year. Therefore, identification of a design basis aircraft and specific analysis of the impact of the design basis aircraft at the ISFSI is not required.

The aircraft hazard analysis considered the probability of aircraft accidents from airways (P_{FA}), civilian and military airports (P_A), designated airspaces (P_M), and holding patterns (P_H). P_M and P_H are equal to zero because the ISFSI is not located in a designated airspace or a holding pattern which left only P_{FA} and P_A to be calculated.

P_{FA} was calculated using the following equation:

$$P_{FA}=(C)(N)\left(\frac{A}{w}\right)$$

The in-flight crash rate, C, used to calculate P_{FA} was 4 x 10^{-10} per mile, which is the value stated in NUREG-0800 for commercial aircraft.

The number of flights per year, N, used to calculate P_{FA} were for three "V" airways:

1. V112, whose centerline is about 13 statute miles south of the ISFSI;

2. V165, whose centerline is about 8 statute miles west of the ISFSI; and

3. V23-287, whose centerline is about 11 statute miles east of the ISFSI.

The Port of Portland Noise Abatement Department provided the numbers of flights in each airway from December 15, 1994, to December 15, 1995, as follows:

Airway	Jet	Non-Jet	Total
V112	2095	7768	9863
V165	651	6620	7271
V23-287	934	16271	17205

The number of flights in each airway was conservatively doubled to account for random flyovers by jet aircraft that may not be using the "V" airways.

The effective area of the ISFSI, A, used to calculate P_{FA} was 3.12 x 10^{-4} mile^2 for non-jet aircraft and 3.09 x 10^{-3} mile^2 for jet aircraft. The smaller effective area for non-jet aircraft represents the physical dimensions of the ISFSI array (approximately 87 x 100 feet), whereas, the effective area for jet aircraft (287 x 300) adds a 100 foot buffer around the ISFSI array to account for the larger physical size of jet aircraft as well as the higher speed that would result in more violent impacts, explosions, etc. Conservatively, no shadow areas are assumed to reduce the effective area. No skid areas are assumed to increase the effective area because the ISFSI is protected from low angle aircraft approach by a combination of berms, hillsides, forest cover, man-made structures, and a ridge.

The width of the airway, w, used to calculate P_{FA} was conservatively selected as 12 nautical miles (99% probability) rather than the normally recognized 8 nautical miles (95% probability).

P_A was calculated considering that the Kelso-Longview airport is the only airport within 10 miles of the ISFSI and NUREG-0800 only provides in-flight crash rates for airports up to 10 miles away. P_A was calculated using the following equation:

$$P_A = (C)(N)(A)$$

The in-flight crash rate, C, used to calculate P_A was 1.2×10^{-8} per mile, which is the value stated in NUREG-0800 for general aviation aircraft for sites located 4 - 5 miles from the end of the runway. No value was given in NUREG-0800 for sites 5 - 6 miles from the runway (distance from ISFSI to Kelso-Longview airport is 5.3 miles), but the 4 - 5 mile value should be conservative based on the data trend in NUREG-0800, which indicates that the further the aircraft is from the end of the runway, the less likely it is to crash.

The annual number of flights affecting the ISFSI, N, used to calculate P_A is one-half of the 35,000 annual airport flight operations, i.e., departing flights. Only departing flights are counted because the Instrument Flight Rules approach to the Longview - Kelso airport is from the north, hence, the flight path of arriving flights would be from the north which would not be near the ISFSI (ISFSI is located directly south of the airport).

The effective area of the ISFSI, A, used to calculate P_A was 3.12×10^{-4} miles^2. This effective area is used because the Kelso-Longview runway is 4,391 feet in length, hence, only smaller aircraft use this airport and a direct aircraft impact would be required to cause damage to the ISFSI. The effective area was not reduced for shadow areas or increased for skid areas as described above for P_{FA}.

Using the above values, P_{FA}, P_A, and the total probability P_{total} were calculated as:

$P_{total} = P_{FA} + P_A$

$= 7.69 \times 10^{-10}$ per year $+ 6.55 \times 10^{-8}$ per year

$= 6.63 \times 10^{-8}$ per year $< 10^{-7}$ per year

Therefore, the probability of aircraft impact is less than 10^{-7} per year and aircraft impacts are not analyzed as a design basis event.

2.2.3.5 Cooling Tower Collapse

The cooling tower is designed to withstand winds of up to 190 mph and earthquake loads of 0.15g. In the unlikely event of collapse, the hyperbolic design of the structure, coupled with its thin-shell configuration, provide an inherently safe failure characteristic. The structure would tend to collapse inwardly. In addition, the structure is located sufficiently far (over 800 feet) from the ISFSI to prevent damage.

2.2.3.6 Air Pollutants

Air pollutants are not anticipated at the ISFSI site.

2.3 METEOROLOGY

2.3.1 REGIONAL CLIMATOLOGY

2.3.1.1 General Climate

The maritime climate of the region around the ISFSI site is typical of the Pacific coast which is characterized by wet winters and dry summers with mild temperatures year long (References 1 and 2).

This region receives substantial annual rainfall, but the rain showers are of light or moderate intensity and continuous rather than heavy for brief periods. Severe storms and thunderstorms are infrequent. On the average, this region receives 2 inches of snow per year.

Regional temperatures are for the most part, mild throughout the year. The average temperature for the summer season is 65°F and for the winter season 40°F. Surface winds seldom exceed gale force. There have been no major hail storms within a 60-mile radius of the site. Tornados rarely occur (References 1 and 2).

2.3.1.2 Severe Weather

The extreme temperatures for Portland, Oregon have been 107°F on July 30, 1965, and August 8, 1981, and -3 °F on February 2, 1950. The maximum amount of precipitation recorded for a 24-hour period was 7.66 inches in Portland in December 1882. The greatest amount of snowfall ever measured for a 24-hour period was 16.0 inches during January 1937, in Portland. These extremes are based on National Weather Service records for 1880 through 1970 and National Oceanographic and Atmospheric Administration (NOAA) data for 1940 through 1994.

According to Huss (Reference 3), the extreme wind gust expected once in 100 years is 130 mph. However, National Weather Service data for 1928 through 1971 and NOAA data for 1940 through 1994, show that the fastest mile wind speed (1 minute average) at Portland was 88 mph on October 12, 1962. The highest windspeed (1 minute average) in Portland from the windstorm on December 12, 1995, was 52 mph, well below the fastest windspeed.

Tornados have occurred occasionally in the site region, usually associated with the passage of fronts from Pacific storms. From 1916 through 1972, 11 tornados were reported within a 60-mile radius of the site (References 4 and 5). Of these, only four occurred within 30 miles of the site. One occurred near Longview, Washington while the other three occurred in the Portland-Vancouver metropolitan area. Tornados that occur in the northwest region of the United States are usually smaller than tornados typical to the midwestern area.

The series of Mount St. Helens eruptions in 1980 resulted in tephra accumulations at the Trojan site of no more than 1/8 inch. If Mount St. Helens were to have another tephra eruption similar to the May 18, 1980 eruption, only directed towards the ISFSI with winds blowing towards the ISFSI, then the expected ash fall accumulation would be about 1.8 inches.

The greatest air pollution potential in the site region exists during the fall and winter seasons when the tendency is greatest for a quasi stationary anticyclone to develop, associated with wind speeds less than or equal to 5 mph and a shallow mixing depth (References 6 - 9).

2.3.2 LOCAL METEOROLOGY

2.3.2.1 Normal and Extreme Values of Meteorological Parameters

Normals and extremes of available temperature, precipitation, relative humidity and fog for Portland, Oregon and Longview-Kelso, Washington can be found in the "Climatography of the United States No. 20-45, Decennial Census of the United States - Summary of Hourly Observations" (Reference 10).

Meteorological data at the site during the period September 1, 1971, through August 31, 1974, are reported in this section (References 11 - 13). These data compare favorably with National Weather Service data for Portland, Oregon.

The distribution of wind direction and speed is an important factor when considering transport conditions relevant to site diffusion climatology. The topographic features of the site region are a major factor in influencing the wind direction distribution at the ISFSI site. During the 1971-1974 period, the prevailing wind for the 30-foot level was from the south, and south-southeast for the 200-foot level, and the average wind speed at the 30-foot level onsite was 8.2 mph, and was 9.3 mph at Portland.

Wind persistence is extremely important when considering potential effects from a contaminant release. Wind persistence is defined as a continuous flow from a given direction or range of directions. There is only a 5-percent probability of continuous wind direction persistence periods greater than 11.5 hours (References 12 and 13).

Temperatures in the region are generally mild considering its high latitude. During the 1971-1974 period, the annual average temperature onsite was 50°F, the daily annual mean minimum onsite was 44°F, and the annual mean maximum onsite was 58°F.

During the 1971-1974 period, the mean relative humidity onsite was 78 percent, and the annual average precipitation onsite was 62.04 inches.

2.3.2.2 Potential Influence of the ISFSI on Local Meteorology

Operation of the ISFSI is not expected to affect the climate of the region. The natural draft cooling tower is anticipated to remain at the site, but will not be used for ISFSI operation. The physical structure of the cooling tower is expected to locally increase atmospheric turbulence. There is also a potential for somewhat decreased low-level wind speeds in the vicinity of the tower. This effect diminishes rapidly with increasing distance downwind from the cooling tower and is relatively insignificant offsite.

2.3.2.3 Topographic Description

General topography in the vicinity of the ISFSI site is shown in Figure 2.1-2. Topographical cross sections out to 10 miles are provided in Figures 2.3-1 through 2.3-6.

The ISFSI site is located in the Columbia River Valley, which at this location is in a general north-south orientation. North of the site the Columbia River bends to the northwest, and south of the site the river bends to the southeast. Within the immediate vicinity of the site, there is a bluff one-half mile to the west rising sharply to 400-500 feet with a highest peak of 1187 feet MSL. North of the ISFSI, there is a wooded hill which rises to 100 feet. The remaining area in the immediate vicinity of the site is flat and low. The Columbia River Valley is approximately 2 miles wide at the site and widens to 3 miles north of the site at Longview-Kelso. The valley walls at the site rise to an elevation of 1000 feet MSL within approximately 1.8 miles to the west and not quite so high to the east.

The effect of the topographic features on airflow trajectory regimes and dilution is quite significant at the site. Analyses of annual wind roses reveal that the predominant wind flow is in a north-south direction. Winds within the Columbia River Valley will be effectively channeled and therefore will follow the changing orientations of this Valley. Computations of average χ/Q values based on the straight line model for a ground-level release indicate that the greatest potential concentrations would be north and south of the site, corresponding to the predominant wind directions. In addition, a nonbuoyant plume will generally not rise out of the valley for a ground-level release during stable temperature lapse rate conditions. Estimates of dispersion during stable conditions, based on the Gaussian diffusion model, indicate that a plume oriented in a general north-south direction would most likely not intersect with the valley walls. Therefore, the valley walls have only a limited effect as a potential barrier to prevent dispersion of the plume since the width of the valley increases both to the north and south of the site and the plume width is relatively narrow during stable conditions. Turbulence created by the mountainous terrain would increase the dilution of airborne effluents.

2.3.3 ONSITE METEOROLOGICAL MEASUREMENTS PROGRAM

The onsite meteorological program at the site began in October 1969 with wind and temperature instrumentation at four elevations: one 500-foot tower plus a 30-foot satellite tower on the bank of the Columbia River. To more accurately define low wind speed conditions, a Climet system was installed on a 33-foot tower located along the site access road. In addition, one 11-inch rain gauge was installed west of the Turbine Building.

Meteorological data was collected during nuclear plant operation and for a time during defueled operation, but data will not be collected during ISFSI operation. The source terms for ISFSI operation are much lower than the source terms for nuclear plant operation. Accidents and off-normal events do not result in releases that would exceed 10 CFR 72.106 limits and OAR 345-26-390. As a result, meteorological monitoring for the calculation of off-site doses from normal operation and accident conditions is not necessary. These doses can be effectively calculated by using conservative values of χ/Q either from the onsite historical data or from bounding assumptions in the Regulatory Guides.

As stated above, meteorological data for the site during the period September 1, 1971 through August 31, 1974 compared favorably with National Weather Service data for Portland, Oregon. Hence, if real time meteorological data is desired, then data from the National Weather Service for Portland could be used.

2.3.4 DIFFUSION ESTIMATES

Diffusion estimates were made for short-term conditions only. There are no long-term (routine) releases associated with operation of the ISFSI.

For the short-term diffusion estimate, a hypothetical accident was postulated to determine the concentrations and doses that could occur following the release. The short-term diffusion estimate is obtained from the following equation:

$$\frac{\chi}{Q} = \frac{1}{\mu \pi \sigma_y \sigma_z}$$

where

χ = concentration, curies/m^3

Q = source strength, curies/sec

μ = mean wind speed, m/sec

σ_y = horizontal dispersion parameter, m

σ_z = vertical dispersion parameter, m

For conservatism, none of the surrounding terrain or manmade structures are considered to create a wake effect and a correction for wake effect is not included in the equation. Also, the assumptions in Regulatory Guide 1.25 of uniform wind direction, wind speed of 1 meter/second, and category F Pasquill diffusion are used and are conservative when compared to onsite historical meteorological data presented in the Trojan Nuclear Plant FSAR (Section 2.3).

Using the above listed assumptions, the χ/Q at the Controlled Area Boundary (325 meters) is 4.3×10^{-3} sec/m^3.

2.4 HYDROLOGIC ENGINEERING

The site location and design of the storage casks assures that the systems and structures that are important to safety withstand the additional forces that might be imposed by the hydrology of the area without loss of the capability to protect the public.

Hydrologically-related design bases, performance requirements, and the design for important to safety structures, systems and components reflect thorough consideration of the following phenomena:

1. Runoff-type floods up to and including the probable maximum flood (PMF).

2. Surges and wave actions.

3. Tsunamis.

4. Artificial floods due to dam failures or landslides.

5. Ice jam flooding.

6. Dilution and dispersion characteristics of normal and accidental release to the hydrosphere relating to existing and potential future users of surface and groundwater resources.

The following sections discuss the hydrological characteristics of the ISFSI site and their influence on ISFSI design and operation.

2.4.1 HYDROLOGIC DESCRIPTION

2.4.1.1 Site and Facilities

The ISFSI site is on a rock outcropping located on the Oregon bank of the Columbia River at River Mile 72.5 in the northwest section of the 634-acre tract of PGE property. The concrete reinforced pad is about 160 feet from the Oregon bank (mean low water) with an intervening hill. Equipment important to safety is located at or above ground elevation 45 feet MSL, which is above postulated flood levels. There is no potential for flood induced erosion because the ISFSI reinforced concrete storage pad is founded on impervious rock.

The Columbia River is about 1/2 mile wide adjacent to the ISFSI. In the vicinity of the ISFSI there are holes in the river deeper than -120 feet MSL. Directly across from the ISFSI the deepest profile is about -70 feet MSL. Topography nearby and on the site area is shown on Figures 2.1-2 and 2.4-1.

The ISFSI site has excellent drainage. The east side of the rocky ridge drains directly into the Columbia River, while runoff on the west side flows into the old river channel and thence by Carr Slough northward until it joins the Columbia River. Neer Creek, a small stream, flows off the steep hillside west of the site and old river channel. Its flow varies from over 30 cfs at times during the winter to essentially zero during dry summer periods. Neer Creek provides flow through the recreational lake with the outflow passing into Carr Slough as it did prior to construction of the Trojan Nuclear Plant.

The northern, unpaved area of the PGE property drains to the perimeter drainage ditch which is approximately 3 feet lower than the ground elevation of the PGE property. This ditch drains to the reflecting lake, and the south drainage ditch empties into the recreational lake. The roads around the PGE property are sloped so that they drain either to the drainage ditch or toward the river except for those portions of roads shown by cross-hatching in Figure 2.4-2, which drain to the indicated catch basins.

2.4.1.2 Hydrosphere

The Columbia River is the major hydrographic feature in the area. It represents one-third of the potential hydropower of the United States, and has an annual discharge of approximately 180,000,000 acre-ft (59 trillion gallons), and drains an area of 260,000 square miles (Reference 1). The Columbia River has an average flow rate of 230,000 cfs at the site with a corresponding average current velocity of 1.8 fps.

A most important factor in considering flows in the Columbia River is the large amount of storage available for flood control and power use. With the dams constructed in the United States and Canada by 1973, more than 30 million acre-ft of storage (Reference 2) is usable in controlling floods on the lower Columbia River.

Tidal effects on the Columbia River can be seen as far upstream as Bonneville Dam, at River Mile 140. The tides at Astoria are typical of the Pacific Northwest tidal pattern. The tides are of a semidiurnal nature with an average period of approximately 12.4 hours.

The effect of tides at the site is dependent to a large part on the flow of the river at the time. Flow reversal occurs at the site on about one-quarter of the tides during a normal year. The extreme tidal range at the site is less than 5 feet, and a maximum upstream flow of 129,000 cfs with an average current velocity of 1.3 fps. The Columbia River has five significant tributaries near the site. None is large enough to have serious effect upon the hydrology at the site.

2.4.1.2.1 Surface Water Use

To determine the extent to which surface water is used within 5 miles and 20 miles of the ISFSI, a survey was made of registered surface water rights. The survey used a 12-mile by 12-mile area and a 36-mile by 36-mile area surrounding the ISFSI in lieu of a 5-mile and 20-mile circle because the Township/Range coordinate system used to identify the location of water rights in the states of Washington and Oregon uses 6-mile by 6-mile boxes. The survey showed that 824 rights are registered within the 36-mile by 36-mile area with an allowed use of 12,480 cfs. Of these rights, 117 are located within the 12-mile by 12-mile area with an allowed use of 401 cfs. The users of surface water will not be adversely affected by the ISFSI because there are no routine effluent releases associated with normal operation and no credible off normal events or accidents that result in liquid effluents.

2.4.2 FLOODS

2.4.2.1 Flood History

Columbia River floods are generally divided into two categories:

1. Spring floods caused by the melting snowpack usually in the upper reaches of the Columbia Basin east of the Cascades.

2. Winter floods caused by intense rain occasionally augmented by melting snowpack in the Willamette and other basins west of the Cascades.

The maximum natural or unregulated flood on record of the Columbia River is that of 1894, which resulted from a combination of hydro-meteorologic conditions including heavy snowpacks and rapid melt plus rainfall. The peak discharge for the Columbia River was 1,240,000 cfs as measured at The Dalles, Oregon. The large floods which have occurred were spring floods resulting from the melt of a large snowpack combined with the spring rain.

On February 8 and 9, 1996, the Columbia River had an estimated flood flow of 850,000 to 900,000 cfs near the Trojan site. This flow resulted in a peak water surface level at the Trojan site of about 22.5 feet. The flood was caused by warm rainstorms from the mid-Pacific falling on snow in the lower Columbia River basin.

2.4.2.2 Flood Design Considerations

The ISFSI ground elevation of 45 feet MSL is sufficient to be considered safe from projected floods. Equipment important to the safety is situated at or above this level. There is no potential for flood induced erosion because the ISFSI reinforced concrete storage pad is founded on impervious rock.

Water surface elevations were determined for several cases including: the standard project flood (1000-year); the 10,000-year flood; the probable maximum flood; potential dam failures; probable maximum surge flooding; and tsunami.

In addition, studies were made superimposing more than one case and adding surges from wind activity, wave action and tidal effects. Thus, use of highly conservative methods provide a large degree of assurance that the safety of the ISFSI is guaranteed from any potential flooding. The most critical flood level for the site is the combination of an unlikely failure of Grand Coulee Dam and the resultant surge combined with a 25-year flood.

2.4.2.3 Effects of Local Intense Precipitation

The following historical data shows the probable maximum precipitation (PMP) based on the U. S. Weather Bureau Hydro meteorological Report No. 43:

Hour	Cumulative Precipitation (in)	Hour	Cumulative Precipitation (in.)
½	0.45	6	3.37
1	0.83	12	5.62
1-½	1.19	18	7.28
2	1.52	24	8.63
2-½	1.82	36	10.69
3	2.09	48	12.25
4	2.60	60	13.46
5	3.02	72	14.44

The PMP does not create substantial loads because the tops of the storage casks will not collect standing water. Even if a few inches of water accumulated, the loads from this water would be bounded by the analysis for the worst case snow load.

The site drainage, as described in 2.4.1.1, is adequate for the duration and amounts of rainfall listed above.

2.4.3 PROBABLE MAXIMUM FLOOD (PMF) OF STREAMS AND RIVERS

The standard project flood (SPF) and the PMF for the Columbia River were established by the U.S. Army Corps of Engineers. The Corps of Engineers issued their findings in a report dated September 1969 (Reference 1). Because the Columbia River is the major river in the area and is adjacent to the site, the PMF for the Columbia River is the controlling event in studies of natural river and stream flooding for the site.

The 1000-year flood (SPF) with an 850,000 cfs discharge will result in a maximum water surface elevation in the Columbia River at the ISFSI site of 21 feet MSL (Reference 2). The unregulated flow for the 1000-year flood is 1,550,000 cfs at The Dalles or 1.25 times the maximum historical flood of 1894.

The 10,000-year flood with a discharge of 1,050,000 cfs will cause a river water elevation of 24 feet MSL at the site. The PMF, as computed by the North Pacific Division of the Corps of Engineers, will have a discharge of 2,200,000 cfs associated with a water elevation of 36 feet MSL at the site and it can be safely passed by existing dams, and those under construction on the main stem of the Columbia River and its major tributaries, except Bonneville Dam. The effect of a failure of Bonneville Dam, which is the nearest upstream dam, would be, according to the Corps of Engineers, negligible in terms of additional flooding.

The PMF was considered to be caused primarily by snowmelt over an extended period of 2 or 3 months with significant runoff contributions from storm rainfall during the snowmelt period. The combination of conditions for the PMF derivation was the most severe considered "reasonably possible" in the Columbia River Basin.

The wave run up during the PMF, based on an overland wind speed of 40 mph, would be approximately 3.2 feet. For the maximum discharge of 2,200,000 cfs, the highest water surface level at the site, as a result of wave run up, would be 39.2 feet MSL, which is 5.8 feet below site grade.

Further detailed discussions of the model used to determine the flood levels are contained in the Trojan Nuclear Plant FSAR. The Corps of Engineer report of 1969 has not been revised, indicating that the description in the FSAR is applicable for the ISFSI.

2.4.4 POTENTIAL DAM FAILURES

Studies have been made of potential dam failures that could affect the ISFSI site by the Corps of Engineers (References 3 and 4) and by Portland General Electric Company (PGE). Two types of failures were considered: a seismically induced failure and a volcanically induced failure. The potential for the most severe flood caused by an earthquake concerns the Columbia River, while the worst possible volcanically induced dam failure concerns the Lewis River in Washington.

2.4.4.1 Seismically Induced Dam Failure

The maximum artificial flood that can occur at the ISFSI site is a catastrophic, massive and sudden failure of Grand Coulee Dam at Columbia River Mile 597. This event is almost inconceivable. The resulting flood, depending on the exact conditions of breach and natural flow conditions, would cause a discharge ranging from 3,600,000 to 4,400,000 cfs at the site with maximum river elevations ranging from 39 to 46 feet MSL. It would reach the site approximately 2 days after the failure of Grand Coulee. Considering dam failure permutations and routing of the artificial flood, the water surface elevation for the maximum artificial flood (4,400,000 cfs) was calculated to be 41 feet MSL and for the smaller flood (3,600,000 cfs) was calculated to be 40 feet MSL. Wave run up based on an overland wind speed of 20 mph would be 1.75 feet, resulting in a peak water level at the site of 42.75, which is 2.25 feet below grade level. Further details and assumption of this analysis are contained in the Trojan Nuclear Plant FSAR.

2.4.4.2 Volcanically Induced Dam Failure

The Lewis River enters the Columbia River approximately 14 miles upstream of the ISFSI site. An artificial flood caused by the eruption of Mt. St. Helens and the domino-type failure of Swift, Merwin, and Yale dams on the Lewis River would result in a maximum discharge of 3,300,000 cfs at the site. Considering dam failure permutations and routing of the artificial flood, the peak water at the site corresponding to the peak flow (3,300,000 cfs) will be from 39 to 41 feet MSL. Wave run up base on an overland windspeed of 20 mph would be 1.75 feet, resulting in a peak water level at the site of 42.75 feet, which is 2.25 feet below site grade. Further details and assumptions of this analysis are contained in the Trojan Nuclear Plant FSAR.

Subsequent to the May 18, 1980 eruption of Mt. St. Helens, the Northwest Forecast Center of the National Weather Service performed a failure analysis of the Swift Dam using their dam break and wave models. Their results indicate that the generated flood wave would reach Woodland, Washington, in about 1 hour and inundate the areas to a height of 35 feet MSL. This wave would be expected to reach Rainier, Oregon, in approximately 3 hours with a peak elevation of less than 30 feet MSL. This is well below the site elevation of the ISFSI (Reference 5).

2.4.4.3 Spirit Lake Blockage Failure

Spirit Lake is located on the North Fork of the Toutle River within a few miles of Mount St. Helens. When Mount St. Helens erupted on May 18, 1980, mud and debris caused a blockage at Spirit Lake and raised the lake's surface elevation. The North Fork of the Toutle River flows into the Cowlitz River, which in turn flows into the Columbia River about 4 1/2 miles downstream of the Trojan site. Therefore, if the blockage were to suddenly fail, the released water and sediment presented a potential flooding hazard to the Trojan site.

The blockage at Spirit Lake is not a potential flooding hazard to the ISFSI site. Spirit Lake has been drained to a level where the blockage no longer confines lake water, thus eliminating the potential for flooding should the blockage be seismically dislodged.

2.4.5 PROBABLE MAXIMUM SURGE FLOODING

The ISFSI is located a considerable distance from the outlet of the Columbia River to the Pacific Ocean and is well protected by the terrain. Furthermore, although storms with winds of up to about 70 mph occur off the coast, hurricane-force winds are rare in the Pacific Northwest (Reference 6). Storm wave heights, as reported by the Columbia River Lightship off the mouth of the Columbia River, rarely exceed 30 feet. The Columbia River estuary acts as a dampener to wave action, such that ocean-bred wind waves are indistinguishable from normal river wind-wave action a few miles upstream from the mouth. Surge flooding of the site (ocean-bred), therefore, is considered unlikely.

2.4.6 PROBABLE MAXIMUM TSUNAMI FLOODING

Historically, the evidence demonstrates that the mouth of the Columbia River is relatively insensitive to tsunamis when compared to Crescent City, California, 310 miles south of the Columbia River entrance.

The tsunami effects at the mouth of the Columbia River are further dissipated inside the river due to the characteristics of the estuary, as was demonstrated during the tsunami generated by the Alaskan earthquake of March 28, 1964.

The grade elevation of the ISFSI site is 45 feet MSL. Because of the large margin between the ISFSI grade elevation and the river surface and because of the insensitivity of the river to tsunami effects, tsunamis are not considered in the design criteria for the ISFSI.

2.4.7 ICE EFFECTS

The general climate in the lower Columbia River Basin is not conducive to ice formation. In addition, the flow of the river during periods of freezing temperatures is sufficiently large (200,000 to 400,000 cfs) that ice formation is impossible in the main streamflow. During extended periods of freezing temperatures, some icing is experienced along the banks of sloughs and inlets where the water is slow moving or stagnant. The lowest recorded river temperature at the site was 34.1°F on February 6, 1971 (period of record 1967 to 1972) (Reference 7). Any surface ice formation would not affect the ISFSI because the ISFSI does not require water for operation.

2.4.8 FLOODING PROTECTION REQUIREMENTS

Facilities/equipment that are important to safety are located at or above elevation 45 feet MSL and none of the postulated floods exceeds that level. Therefore, there are no requirements for flood protection.

2.4.9 ENVIRONMENTAL ACCEPTANCE OF EFFLUENTS

The spent nuclear fuel and GTCC waste at the ISFSI is maintained in dry storage casks. There are no routine effluent releases and no credible off normal events or accidents that result in liquid effluents. Therefore, the ISFSI will have no effect on surface or ground waters.

2.5 SUBSURFACE HYDROLOGY

2.5.1 REGIONAL AND SITE CHARACTERISTICS

The ISFSI site is located on an extremely impervious rocky ridge that is bounded on one side and end by the Columbia River and on the other side and end by an old river channel that has been filled with alluvial sediments.

The old channel is bow-shaped, about 2,000 feet wide and 2 miles long. It is carved in the Eocene Goble Volcanics, which borders the old channel on the west side, and forms a rock knob on the east side that separates the old channel from the Columbia River. The ground elevation in the old channel is between 14 feet and 17 feet. The Goble Volcanics have poor aquifer characteristics.

The slough is filled with fine grained alluvium of Quaternary age. Previous seismic surveys done in the area indicate the bedrock to be up to 340-foot depth. None of the borings drilled in the deep section of the slough penetrated the full thickness of the alluvium. One of the deep borings - DH-7 - terminated about 9 feet in a gravel bed at the 269-foot depth, which is overlain by 154 feet of fine sand, beneath a 114-foot-thick layer of organic silt with a trace of clay. A layer of volcanic ash occurs near 70-foot depth within the silt layer.

The entire alluvial section appears to be hydraulically connected to the river on both ends of the slough. Water levels in the two domestic water wells located near the south end of the alluvial channel respond to tidal fluctuations. A gravel bed below the fine sand is the aquifer for the two wells which supply the Trojan site. These wells are 8 inches in diameter and were drilled through the 158-foot-thick impermeable gray silt bed and 117 feet of fine sand into about 28 feet of gravel at the bottom of the alluvial channel. The gravel bed appears to be bimodal or strongly gap graded. It is composed of medium-to-coarse gravel with a very fine sand matrix. This fine sand controls the aquifer characteristics. These wells are each capable of producing approximately 250 gpm of high-quality domestic water, but the demand is anticipated to be much less than this capacity.

A survey of existing wells and natural springs was made in the area between Goble and the south edge of Rainier to determine the extent to which groundwater is utilized, and to determine the elevation of the permanent water table in the area of the site. The survey showed that bedrock supplies most of the groundwater to existing wells in the area. Approximate water levels in the existing wells were determined, and piezometers were installed in six of the drill holes at the Trojan site to indicate water levels in the alluvium and in the bedrock.

Static water levels in wells, and the elevation of springs emanating from the ridge west of U.S. Highway 30, show that water levels in the ridge are considerably higher than are the water levels in the alluvium, even during periods of very high flow in the river. Consequently, it is apparent that the water table in the alluvium does not feed the water table in the ridge. Thus, the precise local direction of movement of the water is not as important as is the fact that the water in the rock and in the alluvium moves toward the Columbia River and not toward existing offsite wells or springs. The hydraulic gradient of the water table precludes contamination of the portion of the bedrock which now supplies groundwater to offsite wells or springs. It is therefore concluded that there is virtually no possibility of contamination of existing or future offsite groundwater supplies by accidental release of radioactive materials onto the alluvium or rock at the site.

2.5.2 CONTAMINANT TRANSPORT ANALYSIS

Four permeability tests of the alluvial material in Drill Holes 9 and 10 showed permeability ranging from 10 feet to 20,000 feet per year. If accidental discharge of contaminated water onto the alluvium should occur, the water would move through the upper portion of the alluvium and toward the Columbia River at a rate of approximately 15 feet per year. If accidental discharge of contaminated water onto the foundation rock would occur the water would also move toward the Columbia River. If it moved through the fractures and pores in the rock, it would move at a much slower rate than the rate in the alluvium.

2.6 GEOLOGY, SEISMOLOGY AND GEOTECHNICAL ENGINEERING

This section describes and evaluates the geologic and seismic conditions for the region around the ISFSI site. Foundation conditions are evaluated. The seismic history of the region is examined, and the earthquake design criteria are developed and described. These discussions have been summarized from the Trojan Nuclear Plant Final Safety Analysis Report (FSAR) with slight modifications to address ISFSI specifics. Further details and figures, such as geologic profiles, may be found in the FSAR.

2.6.1 BASIC GEOLOGIC AND SEISMIC INFORMATION

The ISFSI site is 31 miles north of the city limits of Portland, Oregon on the Oregon bank of the Columbia River. A portion of the PGE property is underlain by a north-south trending steep-sided ridge of volcanic rock that borders the river and rises to a maximum elevation of 134 feet above MSL. The remainder of the PGE property is underlain by a flat alluvian plain with elevation ranging between 5 and 18 feet. Approximately ½ mile west of the site, a north-south trending range of hills rises steeply above the alluvial plain to elevation in excess of 1000 feet MSL. The Columbia River flows in a northerly direction at the site, but turns to the west several miles downstream.

The ISFSI is located on the east side of the PGE property in a flat, yard area at an elevation of 45 feet MSL. The reinforced concrete slab on which storage casks sit is located on competent rock.

Upon original siting for nuclear plant operation, investigations were performed that may be used in determining the suitability of the PGE property for the storage of spent nuclear fuel and GTCC waste. The investigations were conducted to determine the characteristics of the foundation material, especially in regard to their suitability for supporting the structures, to determine the depth and configuration of the groundwater table, to determine the characteristics of the soil and rock materials with respect to their effect on the migration of radioactive solutions if such solutions come in contact with them, and to evaluate the seismicity of the area so that appropriate parameters for seismic design could be selected. Consultants in geology and seismology were retained to evaluate independently the results of the field investigations.

A river bottom survey ("Boomer" survey) was made by EG&G of Goleta, California, using continuous seismic profiling, to define the shape of the river bottom adjacent to the PGE property. A geophysical survey was performed by Geo-Recon Inc., of Seattle, Washington, across the alluvial valley to the west of the PGE property. A geophysical survey was made at the reactor site, by P.C. Exploration of Carmichael, California, to measure P-wave and S-wave velocities and to calculate the dynamic modulus of elasticity of the foundation rock. Drilling and sampling was done by Lynch Bros. of Seattle, Washington, under the direction of Bechtel Corporation. Soil tests were designated by Bechtel and done by Shannon and Wilson, Inc., Seattle, Washington, in their Portland laboratory. Selected rock core samples from the drill holes were tested by Bechtel in their laboratory in San Francisco.

A comprehensive geophysical survey was made to investigate geologic conditions in the Columbia River channel adjacent to the PGE property. Studies included seismic refraction, resistivity, aeromagnetic, and gravity surveys. The results of these investigations were evaluated by a special advisory board and the results are presented in a geophysical survey report dated August 1, 1972 (Reference 1).

2.6.1.1 Regional Geology

The ISFSI site is located in the Oregon Coast Range section of the Pacific border physiographic province. The Coast Range is farther divided, and the site is on the southeastern margin of the Willapa Hills subsection. The Coast Range section is bordered on the north by the Olympic Range and on the south by the Klamath Mountains. In the area near the site and along the northern two-thirds of the Coast Range, the Puget Trough forms the eastern boundary. The southern third is bounded on the east by the Sierra-Cascade Province.

The Cascade Range east of the site is marked by a chain of volcanic cones whose activity spans most of Tertiary time. Lava flows and pyroclastic deposits range from Eocene to Recent in age. Due to the proximity of the site to these Tertiary features, a detailed study was made to determine the possible effect on the site of lava flows, ash release, or mud flows related to volcanic activity in the region. Special emphasis was placed on studies of Mt. St. Helens, the volcanic cone closest to the site, and one that was active during Recent time.

The rocks exposed in the area are Cenozoic in age. They include marine and terrestrial sediments, and volcanic rocks. The volcanic rocks predominate in quantity. The oldest rocks are a thick sequence (over 5000 feet) of Upper Eocene basaltic flows, pyroclastics, and associated sediments called the Goble series. The foundation rock on the PGE property is part of the Goble series. The unit is widespread in parts of northwestern Oregon and southwestern

Washington. Marine tuffaceous sandstones and other sediments, which were derived in part from the erosion of the Goble series, were later deposited in an advancing Oligocene sea. Accompanying or following the retreat of the sea, the rocks were folded and then eroded to form an area of moderately low relief. During Miocene time, intermittent flows of basaltic lavas poured over this eroded Oligocene surface and buried it to depths of as much as 700 feet. After a period of weathering and erosion followed by some folding, the Troutdale sediments were deposited during the Pliocene period by the ancestral Columbia River. Later tectonic activity folded both the Columbia River basalt and the Troutdale sediments. Changes in sea level during and after Pleistocene contributed to considerable erosion which in places has removed the younger geologic units and exposes the older Goble series, as it has on the PGE property. During Pleistocene, the sea level was 300 to 500 feet lower than its present level and the Columbia River channel near the site was eroded to depths of at least 340 feet below the present sea level. Alluvium has partially filled in the channel since that time.

Folding generally conforms to the northwest Coast Range structural trend. The Eocene formations north of the Columbia River have been folded into a syncline which dips as high as 45 degrees but generally about 10 to 20 degrees. The Goble series underlying the PGE property dips gently to the south or southwest, usually at less than 10 degrees. Pleistocene and recent deposits are apparently flat-lying. No rift-type faults or extensive, continuous faults are in existence in southwest Washington and northwest Oregon. Earthquake activity during the period 1858 to 1965 does not indicate any major active fault near the site. Berg and Baker state in the Bulletin of the Seismological Society of America, January 1963, that the grouping of earthquake epicenters in the Portland area and in other parts of Oregon is associated with the local faulting in those areas. They also state that the probable extension of the San Andreas Fault is clearly exhibited by the alignment of offshore epicenters trending northwest off the coast of Oregon. These offshore epicenters are over 200 miles west of the site

The presence of some ancient faulting in the area is suggested by topography, but the faulting is apparently minor since mapped faulting is generally of small displacement. No evidence of post-Pleistocene faulting has been found. It was therefore concluded that the evidence indicates there are no active faults in the area.

2.6.1.2 Site Geology

Geologic mapping was performed to locate the various geologic units and their contacts and to determine the geologic structure and the characteristics of the geologic units at and near the site. Seismic lines totaling 4350 feet in length were run across the alluvial-filled valley to the west of the rock ridge to obtain a profile of the subsurface materials. River bottom soundings near the site were obtained by continuous seismic profiling to define the shape of the river bottom. A geophysical survey was made to measure the dynamic modulus of elasticity of the foundation rock for the reactor. A drilling program was conducted, consisting of 59 diamond drill and soil sampling holes totaling about 5200 feet, and three piezometer holes totaling 153 feet. Samples were not taken in the piezometer holes. Piezometers were installed in three of the holes. Existing water wells and natural springs near the site were located, and information obtained on the occurrence, present utilization, and movement of the groundwater in the area.

Diamond drill holes are located in the area in order to define the characteristics and configuration of the bedrock which provides the foundation for the structures. Soil borings were located in the alluvium to check the foundation conditions for the main access road and the railroad track.

Groundwater hydrology and seismology are discussed in Sections 2.5.1 and 2.6.2, respectively.

The PGE property is underlain by bedrock, which is a part of the Goble series of Upper Eocene age, and by recent alluvium. Outcrops of bedrock have not been distinguished or separated from areas where bedrock is obscured by a relatively thin cover of residual soil. The bedrock is exposed on the ground surface along a narrow, elongated ridge bordering the left bank of the Columbia River. The ISFSI reinforced concrete storage pad is founded on the rock which forms the ridge. This ridge was formerly an island in the river, but alluvium has since filled in the old river channel west of the ridge to elevations of 5 to 18 feet.

Drill hole DH-4 penetrates the rock in the ridge to an elevation of -240 feet or approximately 285 feet below the general foundation grade. Forty-two other holes also penetrate the rock which forms the ridge. Three holes are located at the intake structure to define foundation conditions. Approximately five holes are situated along the access road and railroad alignment.

The bedrock is volcanic in origin and consists principally of tuffs with lesser amounts of flow breccias, tuff breccias, agglomerates, and basalt flows. Basalt and agglomerate often are exposed on the ground surface since they are more resistant to erosion than is the tuff; however, tuffs and flow breccias are the predominant rock type in the ridge. There are numerous interbedded thin basalt flows, and based on DH-4 and other borings, basalt may be the predominant rock type below river level. Since the basalt often flowed onto eroded or uneven surfaces, it varies greatly in thickness, but in the ridge the basalt flows are generally thin. Some of the individual flows are not continuous across the ridge, probably because at the time they occurred they only filled in depressions in the ground surface.

The tuffs, tuff breccias, and flow breccias are soft to moderately hard, gray with some white spots and veinlets, commonly quite porous, and bedding is commonly not distinguishable. They are usually lightly to moderately fractured but occasionally highly fractured. The fractures are commonly rehealed and tight.

The agglomerate varies in color from reddish blue to blue grey. It is moderately hard, moderately well cemented, and occasionally vesicular. It commonly is more fractured than the other rocks but not as fractured in place as a superficial inspection of the core recovered from drilling might indicate. Nearly all fractures are at least partially rehealed by the deposition of secondary minerals, and the rock is commonly essentially impervious.

The predominant mineral that has rehealed fractures in the rock is calcite, but chlorites and zeolites also occur. The presence of these minerals increases the overall strength of the rock by increasing the cohesion and consequently the shear strength along fractures in which they occur. The presence of these secondary minerals is not, however, required in order for the rock to have sufficient strength to provide an adequate foundation. Materials with no cohesion, such as clean sands and gravels, have high bearing capacities provided they are dense. Fractures in the rock are irregular, and the rock in the foundation is confined, thus the fractures have considerable shear strength.

The basalt varies from vesicular to dense but is generally vesicular. It is blue to blue-black, usually fine grained but occasionally slightly porphyritic. When unweathered it is very hard. Both core and outcrops of basalt commonly exhibit a high degree of fracturing, due to cooling stresses rather than tectonic forces.

The rock in the ridge is often broken by closely spaced fractures and often contains weathered zones, some of which are at considerable depths. Soil cover on the ridge is usually thin. Rock frequently crops out on the ground surface. There are, however, some depressions in the top of the bedrock, and several old stream channels have been eroded across the ridge. An old channel containing potholes filled with sand occurred under the southeast edge of the cooling tower.

Thick alluvial deposits occur in the valley to the west of the ISFSI location. The geophysical survey indicated that the alluvium has a maximum thickness of approximately 340 feet in the area between the ridge and the hills to the west of the site. The accuracy of the geophysical survey was partially verified by DH-7, which was drilled at a point where the survey indicated the alluvium to be approximately 280 feet thick. The hole penetrated alluvium to a depth of 278 feet but it had to be abandoned at that depth. However, the total thickness of the alluvium at the hole is considered to be close to 280 feet since the hole terminated in gravels and boulders which probably occur near the top of bedrock.

The upper approximately 80 to 100 feet of the alluvium usually consists of soft to very soft clayey silt to silty clay with varying amounts of intermixed fine sand and layers of silty fine sand. It also contains considerable amounts of decomposed wood fragments and vegetation, particularly in the 50-foot depth range. In DH-5, -9, and -10, and P-1, -2, and -3, the upper 25 to 35 feet of the alluvium is predominantly silty fine sand, but contains significant quantities of silty clay and clayey silt. DH-7 contains less sand in the upper 35 feet than do the above holes. Holes in the alluvium encountered principally soft clayey silt between approximately 30 to 90 feet in depth.

The seismic survey distinguished a denser material, as indicated by a higher velocity, below a depth of approximately 80 feet, and DH-7 confirmed the existence of the denser material, which is a thick layer of fine and very fine-grained sand. In DH-7, progressively more sand was encountered below a depth of about 100 feet, and between depths of 115 feet and 270 feet the material was essentially fine sand.

2.6.2 VIBRATORY GROUND MOTION

2.6.2.1 Seismicity

The ISFSI is located in an area that experiences moderate seismic activity. Most of the seismic activity has been concentrated in three areas - one about 40 miles east of the site, another approximately 25 miles south of the site, near Portland, Oregon, and a third approximately 65 to 120 miles north of the site, along a belt between Olympia and Seattle, Washington. It is important to note that there is no alignment of epicenters to suggest the existence of any active fault near the site (References 2 - 5).

2.6.2.2 Geologic Structures and Tectonic Activity

Bedrock at the site consists chiefly of basaltic flows and associated pyroclastics included in the Goble series. This unit contains a thick section of widespread volcanics that crop out along both sides of the Columbia River in this area. The unit is Upper Eocene in age. Generally, the rock ranges from dense to vesicular basalt with interbedded agglomerates. As is common with the volcanics of this area, the rocks show closely spaced fractures and locally contain weathered zones.

The data obtained from the geologic study, core drilling and geophysical surveys indicate that the foundation is composed of moderately hard, competent rock that is suitable for an ISFSI. Rock types include tuffs, tuff breccias, flow breccias, basalts, and agglomerates. Of these, tuffs are the most prevalent. Unconfined compressive strengths of the 41 samples of tuffs, tuff breccias, and flow breccias (200 tons/sq. foot) with an average of 1225 psi (88 tons/sq. foot). The specific gravity of the tuffs range from about 1.84 to 2.33 with about 2.10 the average. Absorption ranges from 5 to 17 percent with an average of around 10 percent. The ISFSI reinforced concrete storage pad is founded on rock. Therefore, studies for liquefaction, thixotropy, or differential consolidation of soils were not required.

Geophysical surveys showed compression wave velocities to be 8200 to 10,600 fps, which indicates adequate foundation conditions. Shear wave velocities of foundation rock ranged from 4500 to 5000 fps. A value of 1.9×10^6 psi was used for the dynamic modulus of elasticity, based on the geophysical measurements. Values for static modulus of elasticity were obtained by numerous laboratory tests on representative core samples. A conservative value of 0.8×10^6 was used for the design. No uphole velocity measurements were made.

2.6.2.3 Maximum Earthquake Potential

The largest historically recorded shock center within 50 miles of the site occurred on November 5, 1962. It had an epicentral intensity of VII about 35 miles south of the site near Vancouver, Washington. At Longview, Washington, and at Rainier, Oregon, the intensity was reported as VI; however, the damage was confined to cracked plaster.

On October 12, 1877, an intensity VII earthquake was felt in Portland, Marshfield (now called Clackamas), and Cascades (now called Cascade Locks), Oregon. The location of the epicenter of this shock is uncertain. It is plotted 45 miles from the site in the southern part of Portland, near Clackamas. The epicenter may have been farther east toward Cascade Locks, Oregon. The original reference to this earthquake is by Rockwood (Reference 6). Unfortunately, he does not state exactly what his source is. His account follows:

> "October 12, 1877. Quite severe shocks were felt in Oregon occurring in Portland at 1:53 p.m. (Two shocks being noticed; at Marshfield, Clackamas Co., at
>
> 1:45 p.m.; and at Cascades at 9:00 a.m.) The vibrations were in each case from north to south and were sufficiently violent to overthrow chimneys."

A second reference by Holden (Reference 7) assigned an intensity VIII on the Rossi-Forel (R-F) scale to this earthquake. Townley and Allen (Reference 8) also showed intensity VIII (R-F scale). Rasmussen (Reference 9) and Berg (Reference 2) both used intensity VIII but changed to the Modified Mercalli (MM) scale. The U.S. Coast and Geodetic Survey (Reference 4) shows intensity VII. It is generally accepted that VII (MM) is the equivalent of VIII (R-F).

In correspondence with Bechtel, Mr. Don Tocher, Director of the Earthquake Mechanism Laboratory of the U.S. Coast and Geodetic Survey in San Francisco, stated that he believes an intensity VII (MM) to be correct for the October 12, 1877, earthquake, and states that the intensity VIII (MM) referred to by Berg (Reference 2) and Rasmussen (Reference 5) is probably a result of carelessness in designated scales when changing from the Rossi-Forel scale to the Modified Mercalli scale.

Seven earthquakes of maximum intensity VI were centered within 50 miles of the site. On February 3, 1892, a shock of intensity VI occurred at Portland, and strong vibrations were felt at Astoria, Salem, and Lake Harney, 235 miles southeast of Portland. On December 29, 1941, a shock of intensity VI was felt near Portland, Oregon, about 38 miles south of the site. Another shock, centered near Portland on December 15, 1953, was felt with intensity IV at Kalama, Washington. On September 15, 1961, an earthquake was centered about 33 miles east of the site. The maximum intensity of VI was reported at Swift Dam on the Lewis River, which was designed for 0.10 g, but no damage was done to the dam. At Rainier, the shock on September 15, 1961, was also of intensity IV, and at Carrolls, the intensity was I to III. An after shock on September 17, 1961, was also intensity VI, but was not felt at Rainier or Carrolls. On November 6, 1961, a shock was centered about 30 miles south of the site. At Rainier, the intensity was reported as V. On December 26, 1963, an earthquake of intensity VI occurred about 30 miles southwest of the site. The intensity was reported as V in Longview, and III at Goble. It was not felt in Rainier.

The largest earthquakes within 150 miles of the site were two shocks of epicentral intensity VIII which occurred on April 13, 1949, and April 29, 1965. The epicenters were in the Puget Sound area, approximately 70 miles and 95 miles, respectively, northeast of the site. Heavy damage, deaths, and injuries were reported in the epicentral areas. The accelerograms indicate a maximum resultant horizontal acceleration of about 0.10 g at Seattle, where the intensity was VIII. At Rainier, which is about the same distance from the epicenter as is Seattle, the intensity was also given as VIII, which is the greatest intensity reported historically at Rainier from any earthquake. However, the intensity at Rainier was based on damage to only one building, and that building was founded on marshy ground. A study of the damage reported at Rainier indicates that a lower intensity might reasonably be assigned. At Goble, the intensity due to the 1949 earthquake was only VI, apparently because Goble is founded on rock. The ISFSI is founded on rock belonging to the same unit as the rock that underlies Goble.

The April 29, 1965, earthquake caused lower intensities in the site area than the 1949 earthquake. The intensity at Kelso and Longview was VI; at Rainier it was V; and at Goble only IV.

2.6.2.4 Seismic Margin Earthquake

The maximum intensity that has been reported at Rainier is VIII. Since this intensity occurred on overburden, it is probable that on rock at the site the intensity for this same shock was not over VII. Intensity VII correlates with a horizontal acceleration of 0.12 g according to Hershberger (Reference 10). This historical data formed the bases for assigning the Safe Shutdown Earthquake (SSE).

The SSE was determined such that any probable earthquake experienced at the site would not exceed the intensity selected. An intensity of VIII was selected since it was probable that an earthquake of that magnitude had never been experienced at the site. An intensity VIII is equivalent to an acceleration of 0.25 g.

There have been significant changes in the perception of earthquake hazards in the Pacific Northwest since the time of the initial design and licensing of the Trojan Plant. It is now commonly believed among the geoscience community that large subduction zone earthquakes likely occurred along the Oregon-Washington-Vancouver Island coast (known as the Cascadia margin, or Cascadia Subduction Zone) within the recent past (Holocene), and that the potential for such events to occur in the future should be considered in any evaluation of safety and reliability of critical facilities during earthquake loading.

In 1987, in response to the emerging issue of potential subduction zone earthquakes, PGE initiated a program of close monitoring of earthquake hazard research conducted along the Cascadia margin. The results of these studies, together with studies initiated by PGE, have been use to characterize the maximum events that could be expected to occur in the region and the resulting free-field ground motions that may occur at the site. This maximum potential earthquake that could affect the site is called the Seismic Margin Earthquake (SME).

These studies determined a value for the SME peak horizontal ground acceleration of 0.38 g (Reference 11). A 1994 earthquake in Northridge, California slightly changed the conclusions of these studies in that the controlling earthquake varies from the intraslab source for peak ground acceleration, to the crustal earthquake for periods between 0.1 and 0.6 seconds, to the interface source for longer periods, whereas in the original study, only the intraslab and interface sources were controlling (Reference 12). Nonetheless, the response spectra are bounded by the Regulatory Guide 1.60 spectrum shape when anchored at the 0.38 g peak acceleration. Therefore, input from recent earthquakes shows that the SME is the appropriate design basis event for the ISFSI, as required by OAR 345-26-390, and the ISFSI design considers the SME peak horizontal acceleration of 0.38 g.

2.6.3 SURFACE FAULTING

The site is in the Willapa Hills geomorphic province, a part of the Coast Range. Most of the region is below 2000 feet in elevation. The descent from the hills to the Columbia River is rather precipitous, but elsewhere the hills merge gradually into the surrounding lowlands.

The bedrock in the area is comprised of a series of moderately folded tertiary formations of both sedimentary and volcanic origin. The folding in the area conforms generally to the northwest Coast Range structural trend. The Eocene formations north of the Columbia River are folded as part of a syncline. Pliocene sediments in the vicinity of the site are only slightly warped and the Pleistocene and Recent deposits appear to be flat lying and undisturbed. A detailed discussion of the regional geology is presented in Section 2.6.1.1.

Faulting is minor in the structural development of the area, and is generally of small displacement. Many of the mapped faults in the area are based on topographic lineations in the pre-Pleistocene strata. No evidence of post-Pleistocene surface displacement has been found in the area.

An extensive investigation to locate faults which might be significant to the site was made as part of the geologic evaluation for siting the Trojan Nuclear Plant. A special effort was made to detect any lineations or indications of offset in the alluvium, terrace deposits, or the Pleistocene alluvial deposits. The details of the investigation are contained in the Trojan Nuclear Plant FSAR. As a result of the investigation, the following conclusions were reached:

1. The "Kelso Fault", as indicated in Bulletin 54 of the Washington Division of Mines and Geology (Reference 13), does not exist.

2. The available geologic evidence indicates that there is not a fault in the old stream channel west of and adjacent to the site.

3. The field evidence indicates that the Clatskanie fault does not extend farther east than indicated on the Oregon State Geologic Map.

4. The available geologic evidence does not indicate that a fault exists along the Columbia River adjacent to the site.

5. The fault zone exposed in the road cut southeast of Kelso and 4.7 miles from the site is apparently not extensive and probably has not experienced movement since deposition of the Troutdale formation during lower Pliocene time. This fault zone is not significant to the site.

6. There is no fault within 5 miles of the site which has experienced movement since Pleistocene time.

The size of faults within a 200-mile radius of the site, together with the known historical activity and distance from the site, suggest that ground accelerations reaching the site from these mapped faults would fall well below those for which the ISFSI was designed.

2.6.4 STABILITY OF SUBSURFACE MATERIALS AND FOUNDATIONS

The existing gravel fill will be removed during excavation for the ISFSI reinforced concrete storage pad. No soluble or cavernous rocks underlie the site area, and no poorly consolidated or mineralogically unstable rocks occur at the site. No oil, gas, or other mineral extraction, or subsurface mining occurs or has occurred in the vicinity of the site. It is therefore concluded that future subsurface subsidence is not a problem at the site.

No evidence of recent regional warping was encountered, and USGS Bulletin 1119, 1963, by Donald E. Trimble (Reference 14), states:

> "...deformations of the region apparently ended in early Pliocene time as the Troutdale formation is the youngest one involved in the warping. The undeformed boring lava overlies an erosional surface of considerable relief cut on the Troutdale. Post-Troutdale crustal movements, if any, have consisted only of vertical movement of regional extent."

The geologic mapping near the site, and the study of aerial photographs, did not disclose any indications of recent regional warping.

Because the ISFSI reinforced concrete storage pad is founded on the crest of a rock ridge which shows no evidence of deformation since Pliocene time, no unrelieved residual stresses should be expected to exist in the foundation rock. No evidence of unrelieved residual stress was observed during previous excavations for the nuclear plant foundations.

2.6.4.1 Geological Foundation Evaluation

Fifty-five samples of rock core from the drill holes were tested in the Bechtel Geology Laboratory in San Francisco for the original siting of Trojan Nuclear Plant. Standard testing procedures were used in the determination of the physical properties of the rocks.

Rock types include tuffs, flow breccias, basalts, and agglomerates. Since contacts between rock types are not always horizontal, and the rock units are often lenticular, the ISFSI reinforced concrete pad may rest on one rock type in one portion of the pad and a different rock type in another part of the pad. This does not complicate the design, however, since the pad is designed for the strength of the weakest rock types, which tests have consistently shown to be the tuffs.

Since the tuff is assumed to form the foundation for the reinforced concrete storage pad, its strength will generally determine the allowable bearing capacity of the foundation rocks. The lowest and highest unconfined compressive strength of the 41 samples of tuff which were tested are 360 psi (26 tons/ft^2) and 2790 psi (200 tons/ft^2). The average is 1225 psi or 88 tons/ft^2.

Testing showed the specific gravity of the tuff to be in the range of about 1.84 to 2.33 with about 2.10 the average. Porosity varies from 10.3 to 32.4 percent with 22 percent begin about the average. Absorption ranges from 5.0 to 17 percent with an average of around 10 percent.

Soil test data such as grain size, Atterberg limits, water content, soil density, and shear strength were not required because the ISFSI reinforced concrete storage pad is founded on rock. Tests were made in the flat alluvial area where access roads are found.

The geophysical survey showed the compression wave velocities in the bedrock to be 8200 to 10,600 fps, which indicates adequate rock for good foundation conditions. Shear-wave velocities of the foundation rock ranged from 4500 to 5000 fps. The flat alluvial area west of the rock ridge had compression wave velocities of 2000 to 2500 fps for the overburden and velocities of 4700 to 5100 fps for the older compacted overburden. No equipment important to safety is founded on this alluvial material.

In determining the bearing capacity of these rocks, it was noted that the rock which will form the foundation has been preloaded by the weight of overlying material, much of which has been removed by erosion. This, as well as the jointing and weathering, were considered in determining the allowable bearing capacity of the rock.

Values for the static modulus of elasticity were obtained by numerous laboratory tests of representative core samples. A conservative value of 0.8 x 10^6 psi for the static modulus of elasticity was used for design. For dynamic modulus of elasticity, a value of 1.9 x 10^6 psi was used in design. The value was determined by geophysical measurements on in-site foundation rock. Values for Poisson's ratio (dynamic) range from 0.28 to 0.36. From these values, the bulk modulus was computed to be 1.8×10^6, and a value of 0.7×10^6 psi was calculated for the shear modulus. For design purposes, soil structure and rock foundation interaction was assumed to be negligible, thus no damping was used for the rock.

Observations of groundwater levels at the site indicated that the ridge supports a local groundwater mound, probably maintained by rainfall trapped in depressions on the top of the ridge. Previous excavation for the nuclear plant showed the water to be trapped in joints and fractures in the rock and the water drained off rapidly. No artesian pressures were encountered during the previous excavation.

2.6.5 STABILITY OF SLOPES

Permanent excavated slopes to the north and east of the ISFSI are through rock and no problem with long-term stability of such slopes should be anticipated. Sloughing of small amounts of loose weathered surface material would likely not reach the ISFSI, and would not represent a hazard to the ISFSI in any event.

2.6.6 VOLCANOLOGY

Due to the proximity of existing inactive volcanoes in the Cascade Range east of the site, the significance of renewed activity of these volcanoes was considered with regard to the possible effects on the site. The historical seismicity of the Cascade volcanoes was considered as well as the type of volcanic activity that might conceivably occur in the future. The volcanoes that were considered and their distances and direction from the site are:

1. Mt. St. Helens, Washington - 34 miles - ENE

2. Mt. Adams, Washington - 67 miles - E

3. Mt. Hood, Oregon - 74 miles - SE

4. Mt. Rainier, Washington - 77 miles - NE

The details of volcanology are described in the Trojan Nuclear Plant FSAR. In summary, the conclusions of the FSAR were that while predictions related to future volcanic activity are impossible to make, the possibility of volcanic activity significant to the site is considered very remote. In addition, even if activity did occur, it is extremely unlikely that it would occur without warning.

The series of Mount St. Helens eruptions in 1980 resulted in tephra accumulations at the Trojan site of no more than 1/8 inch. If Mount St. Helens were to have another tephra eruption similar to the May 18, 1980 eruption, only directed towards the ISFSI with winds blowing towards the ISFSI, then the expected ash fall accumulation would be about 1.8 inches.

For these reasons, the risk to the ISFSI from volcanic cones in the Cascade Range is considered minimal.

2.7 SUMMARY OF SITE CONDITIONS AFFECTING CONSTRUCTION AND OPERATING REQUIREMENTS

The site-specific phenomena and characteristics described in this chapter have been used to define appropriate design criteria, as described in Chapter 3. Table 2.7-1 is a summary of site-specific information for the ISFSI.

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REFERENCES FOR SECTION 2.5

None

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Table 2.1-1

Public Facilities and Institutions

	Enrollment	Distance (mi)	Direction
WASHINGTON SCHOOLS:			
Longview			
Broadway	114	7	NNW
Carrolls	131	2-1/4	NNE
Cascade Middle School	910	8-1/2	NNW
Columbia Heights	329	8-1/2	NNW
Columbia Valley Gardens	496	8-1/2	NNW
Kalama High School	413	3	SE
Kalama Grade School	410	3	SE
Kessler	535	7	NNW
Lower Columbia Jr. College	3869	7-1/2	NNW
Mark Morris High School	1189	7-3/4	NNW
Mint Valley	688	8-3/4	NNW
Monticello Middle School	866	7-2/3	NNW
Natural High School	24	8-3/4	NNW
Olympic	562	7-2/3	NNW
R. A. Long High School	952	7-2/3	NNW
Robert Gray	582	10-1/2	NNW
Rose Valley	173	5	NNE
St. Helens	421	7	NNW
Kelso			
Barnes	552	8-1/4	N
Beacon Hill	447	9-1/4	N
Butler Acres	528	8-2/3	N
Catlin	426	7-2/3	NNW
Coweeman Middle School	662	7	N
Huntington Middle School	583	8	N
Kelso High School	1222	7	N
Wallace	467	6-2/3	N

Table 2.1-1

Public Facilities and Institutions

	Enrollment	Distance (mi)	Direction
Private			
Columbia Heights	175	8-3/4	NNW
Kelso Christian Academy	200	7-1/4	NNW
Longview Christian School and Day Care	700	8	NNW
St. Rose	237	7	NNW
Seventh Day Adventist	31	7	NNW
OREGON SCHOOLS:			
Columbia City Elementary	151	10-1/4	SE
Goble Elementary	119	2-1/4	S
Hudson Park	349	6-1/4	WNW
Rainier Elementary	307	4-1/4	NW
Rainier High School	441	6	NW
Rainier Middle School	244	3-3/4	NW
HOSPITALS			
St. Johns Medical Center	340	6-2/3	NNW
NURSING HOMES			
American Convalescent & Retirement Home	74	7	NNW
Campus Towers	110	7-2/3	NNW
Canterbury Retirement Inn	104	7	NNW
Cowlitz Convalescent Center	46	7-1/4	NNW
Delaware Plaza	87	6-1/2	NNW
Frontier Extended Care Facility	136	6-3/4	NNW
Manor Nursing Home	51	7	NNW
Northwest Continuum Care Center	66	9-1/2	NNW
Park Royal	49	7	NNW
Woodland Care Center	58	10	SSE

Table 2.1-1

Public Facilities and Institutions

	Enrollment	Distance (mi)	Direction
PARKS			
Bailey Park		8-3/4	NNW
Clearview Park		6-1/2	NNW
Cloney Park		7	NNW
Gerhart Gardens Park		5	N
Highlands Park		6	NNW
Hudson Parcher Park		4	NW
John Null Park		8-1/3	NNW
Kellogg Park		8	NNW
Kelso Rotary Park		8	N
Lake Sacajawea Park		7	NNW
Prescott Beach Park		1	NNW
R. A. Long Square		7-1/4	NNW
Riverside Park		10	N
Roy Morse Park		10-3/4	NNW
Scott Hollow Park		7-1/4	N
Seventh Avenue Park		6-1/2	NNW
Tam O'Shanter Park		6-3/4	N
Vandercook Park		7-1/4	NNW
Windemere Park		10-3/4	NNW
COUNTY FAIR GROUNDS			
Cowlitz County	56,000 during fair - 54,000 total during rest of year	7	NNW

Table 2.1-2

1994 LAND USE CENSUS

NEAREST LOCATION TO TROJAN WITHIN A FIVE-MILE RADIUS

Directional Sector	Residence	Garden	Milk Cow	Milk Goat	Meat Animal
	Radial Mileage for Nearest Location				
N	0.70	0.80	None	None	None
NNE	2.00	2.80	4.00	2.00	None
NE	1.60	2.00	None	None	2.00
ENE	2.30	None	None	None	4.00
E	1.30	1.30	None	None	1.30
ESE	0.80	2.40	None	None	2.40
SE	2.30	2.80	None	None	2.70
SSE	1.40	3.00	None	None	3.00
S	1.20	1.40	None	None	2.00
SSW	0.90	2.60	None	2.60	1.00
SW	1.50	3.00	None	3.00	2.10
WSW	1.40	1.60	None	None	3.20
W	1.70	2.10	None	None	2.20
WNW	1.70	1.70	None	None	1.70
NW	1.20	1.20	None	None	2.00
NNW	0.60	0.60	None	None	None

Table 2.2-1

Nearby Industrial Facilities

Map Ref No.	Company	Distance from site (mi)	Description of Products or Services	Annual Production	Product Storage	Shipments: In	Shipments: Out	Hazardous Chemicals Used	Storage
1	Port of Longview	5.6	Dock facilities loading, unloading, storage	180-200 ships per year	3.3 million sq ft	Ship Rail Truck	Ship Rail Truck		
2	International Paper Company	5.5	Logs, woodchip export	[b]	logs 20 million board ft chips 20,000 bone dry units	Truck Rail	Ship Truck	Diesel Fuel	5,000 gal [c]
3	Longview Fibre Company	5.0	Paper products and paper board	1.2 million tons	[b]	Rail Truck Ship Barge	Rail Truck Ship	Sulfuric Acid Caustic Soda Chlorine Fuel Oil Liquid Propane Ammonia Phosphoric Acid Nitric Acid Hydrogen Peroxide	740,000 lb 500,000 lb 28,000 lb 9,800,000 gal 10,000 gal 30,000 lb 40,000 lb 60,000 lb 70,000 lb
4	Cytec Industries	6.0	Paper, water, and mining chemicals	35 million lbs	400,000 gal	Rail Truck	Truck Rail	Acrylic Acid Acrylamide Dimethylamine Parafin solvent Caustic Soda Sulfuric Acid Hydrochloric Acid Formaldehyde Sulfur Dioxide	10,000 gal 30,000 gal 30,000 gal 30,000 gal 8,000 gal 3,500 gal 6,000 gal 15,000 gal 250 lbs
5	Specialty Minerals	7.0	Chemicals and products for paper industry [b]	[b]	[b]	Truck	Truck	Calcium Carbonate Calcium Hydroxide Calcium Oxide Acid	500,000 lb [c] 50,000 lb [c] 500,000 lb [c] 50,000 lb [c]

Table 2.2-1

Nearby Industrial Facilities

Map Ref No.	Company	Distance from site (mi)	Description of Products or Services	Annual Production	Product Storage	Shipments: In	Shipments: Out	Hazardous Chemicals Used	Storage
6	Weyerhaeuser	7.0	Lumber, wood pulp, paper board, kraft pulp, container board, printing and writing materials	4.5 million tons	[b]	Rail Truck Ship Barge	Rail Truck Ship Barge	Sulfuric Acid Alum Chlorine Caustic Starch Clay	300,000 lb [c] 200 tons [b] [b] 400 tons [c] 200 tons [c]
7	North Pacific Paper Company	7.75	Newsprint	675,000 tons	21,600 tons	Rail Truck	Rail Truck	[b]	[b]
8	Reynolds Metal Company	8.7	Aluminum billet	220,000 tons	Little	Ship Rail	Ship Rail Truck	Alumina Coke Bituminous Pitch Propane Chlorine	70,000 tons 10,000 tons 100,000 gal 80,000 gal 1 tank car
9	Wayron	6.5	Industrial machine shop, custom machine building, welding, and painting	2,000 tons	500,000 lbs	Truck	Truck		
10	Peavy Grain	3.5	Grain handling and storage	200 million bushels	2 million bushels	Ship Rail	Ship Barge Truck		
11	Kalama Chemical Company	2.0	Phenols, aldehydes, amines, benzoates, and K-flex plasticizers	750 million lbs	[b]	Ship	Rail Truck	Toluene	5,000,000 gal
12	Northwest Natural Gas Pipeline	1.6	16 inch O.D. natural gas pipeline	3 million cu ft per hour capacity					
13	Kelso-Longview Airport	6.5	Municipal Airport	35,000 operations	60 planes	-	-	Aviation Fuel	10,000 gal
14	Gram Lumber Company RSG Forest Products	3.0	Cedar lumber, Dimensional lumber	180 million board ft	[b]	Truck	Truck Rail Barge	Natural Gas	pipeline

Table 2.2-1

Nearby Industrial Facilities

Map Ref No.	Company	Distance from site (mi)	Description of Products or Services	Annual Production	Product Storage	Shipments: In	Shipments: Out	Hazardous Chemicals Used	Storage
15	Chevron Chemical Company	9.25	Fertilizer	200,000 tons	20,000 tons	Pipeline	Rail Barge Truck	Natural Gas	Pipeline
16	Olympic Pipeline Company	1.9	14 inch O.D. refined petroleum product pipeline	~8,000 bbl per hour					
17	Solvay Interox	8.5	Manufacture hydrogen peroxide	143 million lbs	8 million lbs	Truck	Rail Truck	Natural gas	pipeline
18	All Pure Chemical Company	2.0	Repackage chlorine gas, sodium hypochlorite, household ammonia, products for water treatment	~70 gal/min	15,000 gal	Rail	Truck	Chlorine Caustic Soda Sodium Hypochlorite	180,000 lbs 180,000 lbs 15,000 gal
19	Hoechst Celanese Corporation	3.0	Bleaching agents for pulp and paper industry, zinc oxide, sodium hydrosulfite	12 million lbs	9,500 lbs	Rail Truck	Rail Truck	Sulfur Dioxide Sodium Hydrosulfide	300,000 lbs 200,000 lbs
20	BHP Steel	0.5	Zinc/aluminum coated and painted steel strip in coils (est. 1997)	400,000 tons	[b]	Truck Ship Rail	Truck Rail	Liquid Hydrogen Hydraulic Oils Paint Caustic Soda Diesel Fuel Propane Gas Hydrochloric Acid Natural Gas Passivation Chemical Paint Solvents	5,000 gal 10,000 gal 15,000 gal 20,000 lbs 10,000 gal 20,000 gal 20,000 gal pipeline 500 gal 500 gal
21	Wilson Oil (dba Wilcox & Flegel)	5.6	Petroleum distribution	7 million gal	26,190 bbl	Truck	Truck	Petroleum	26,190 bbl
22	Northwest Pipeline	3.5	26 inch and 30 inch O.D. natural gas pipelines	20 million cu ft per hour capacity					

[a] Shipping is noted in order of importance

[b] Proprietary data or not available.

[c] Estimated.

Table 2.7-1

Summary of Site Conditions Affecting Construction and Operating Requirements

Parameter	Extreme Value Measured or Estimated	Design Value
Ambient Temperature	-20°F minimum, 105°F maximum (Longview/Kelso, Washington) -3°F minimum, 107°F maximum (Portland, Oregon)	-40°F minimum 100°F maximum 125°F short term extreme
Tornado Winds/Pressure Drop	240 mph, maximum 190 mph, rotational 50 mph, translational 1.5 psi, pressure drop 0.6 psi/sec, rate of pressure drop (Regulatory Guide 1.76, Region III)	360 mph, maximum 290 mph, rotational 70 mph, translational 3.0 psi, pressure drop 2.0 psi/sec, rate of pressure drop
Maximum Flood Level	42.75 ft MSL (estimated - seismically induced dam failure with coincident wave runup)	None required - nominal ISFSI grade elevation is 45 ft MSL
Snow and Ice Loading	10 inches of snow in 24 hours (Longview/Kelso) 16 inches of snow in 24 hours (Portland)	100 psf ground load (12 inches of snow exerts from 1 to 6 psf)
Atmospheric Dilution Factor	7.10 10^{-5}sec/m^3 (measured at Trojan 1976-1993)	4.08 10^{-3}sec/m^3
Seismic Margin Earthquake	0.38g, horizontal acceleration (estimated) 0.25g, vertical acceleration (estimated)	0.38g, horizontal acceleration 0.25g, vertical acceleration

3.0 PRINCIPAL DESIGN CRITERIA

The following sections provide a discussion of the principal design criteria for the ISFSI. The design criteria have been derived from Title 10, Code of Federal Regulations, Part 72 (10 CFR 72), and applicable industry codes and standards.

3.1 PURPOSES OF INSTALLATION

The Trojan Independent Spent Fuel Storage Installation (ISFSI) is designed for dry, above ground storage and ultimate offsite transport of intact and failed spent nuclear fuel assemblies, fuel debris, and Greater Than Class C (GTCC) waste. The material will be sealed in baskets and stored within ventilated concrete storage casks arranged on a reinforced concrete pad. The stand alone ISFSI, when operational, allows deactivation of the Trojan Nuclear Plant Spent Fuel Pool and decommissioning of the reactor site.

3.1.1 MATERIALS TO BE STORED

The physical, thermal, and radiological characteristics of the material to be stored in the ISFSI are provided below.

3.1.1.1 Intact Fuel Assemblies

A total of 780 intact fuel assemblies will be loaded. Of these, 732 are Westinghouse 17x17 fuel assemblies and 48 are B&W Fuel Company 17x17 fuel assemblies. Fuel assemblies may contain inserts which consist of rod cluster control assemblies (RCCA) or burnable poison rod assemblies (BPRA), thimble plugs and sources. Not all fuel assemblies contain inserts.

Physical design parameters of the fuel assemblies are shown in Table 3.1-1. Limiting radiological and thermal characteristics are shown in Tables 3.1-2 and 3.1-3.

For 17x17 fuel assemblies, 264 grid locations contain fuel pins, 24 grid locations contain thimble guides to allow for fuel assembly inserts, and the center grid location contains an instrument sheath. RCCAs consist of 24 absorber rods which can be inserted into the thimble guides. A total of 61 RCCAs will be stored in the ISFSI. BPRAs are similar to RCCAs but consist of fewer absorber rods (9 to 20). Thimble plugs were used to "plug" thimble guides which did not contain absorber rods or sources during reactor operation. Sources are similar in shape to absorber rods but a portion of the length contains a secondary neutron source. The primary design concern associated with these components is weight. Table 3.1-4 summarizes the physical characteristics of the inserts.

The main physical parameters of concern are the fuel assembly dimensions and weight, and envelope (cross-sectional dimension). These parameters establish the mechanical and structural design aspects of the Concrete Cask and basket. The thermal and radiological characteristics establish the shielding and thermal aspects of the design.

3.1.1.2 Failed and Partial Fuel Assemblies

Ten (10) partial fuel assemblies, which contain intact, suspect, or failed fuel rods will require storage.

3.1.1.3 Fuel Debris

Fuel debris consists of loose fuel pellets, fuel pellet fragments, and fuel assembly fragments (portions of fuel rods, portions of grid assemblies, etc.). The quantity of fissile material contained as fuel debris will not exceed 10 kg per basket. This limit is imposed to satisfy the license conditions of the TranStor™ Shipping Cask (Reference 1). An additional limit for fuel debris of no more than 20 curies of plutonium is imposed to meet the offsite transportation requirements of 10 CFR 71.63.

3.1.1.4 GTCC Waste

GTCC waste consists of activated core components consisting mainly of segmented reactor internals. GTCC waste characteristics such as weight and curie content are addressed in Table 3.1-2. GTCC waste is not stored in the same basket with spent fuel.

3.1.2 GENERAL OPERATING FUNCTIONS

The Trojan ISFSI design accounts for a maximum of 36 Concrete Casks, each containing a PWR Basket or GTCC Basket. It is anticipated that only 33 PWR Baskets will be required to store spent nuclear fuel, however, the ISFSI design accounts for 34 PWR Baskets and two (2) GTCC Baskets.

The Concrete Casks and associated baskets are oriented vertically and arranged above ground on a reinforced concrete Storage Pad.

Basket loading operations and preparations for storage are discussed in Section 5.1.1. After preparations for storage are complete, the basket is transferred into a Concrete Cask within the Fuel Building. The loaded Concrete Cask is then transferred to the ISFSI storage location on an air pad system.

Due to the passive design of the ISFSI, operations primarily consist of inspecting the Concrete Cask air vents for blockage and monitoring outlet vent air temperature. Handling operations are anticipated to be limited to transferring the PWR Baskets and GTCC Baskets to a shipping cask for off-site transport for disposal or storage. In the unlikely event of a basket leak, the basket may be repaired or transferred to a Basket Overpack.

3.2 STRUCTURAL AND MECHANICAL SAFETY CRITERIA

The storage system is designed to be stored outdoors without additional weather protection. The system is designed to withstand the daily and seasonal temperature and environmental fluctuations as well as tornado, flood, seismic and handling loads. Various loads are considered per the recommendations of ANSI 57.9 (1984). Trojan ISFSI site specific environmental and geological features are addressed in Chapter 2.

Average summer and winter air temperatures for the region are 65°F and 40°F, respectively. Short term extreme temperatures for the site are not anticipated to be less than -20°F or greater than 107°F based on historical data. Steady state temperature limits for the site were assumed to be bounded by ambient air temperatures between -40°F with no solar loads and 100°F with maximum solar loads. Since short term operation with ambient air temperature in excess of

100°F is credible, an analysis was also performed assuming 125°F steady state air temperature with maximum solar loads for comparison to ISFSI short term operating limits. Thermal evaluation of ISFSI performance is provided in Section 4.2.6.

3.2.1 TORNADO AND WIND LOADINGS

The ISFSI storage structures are designed to withstand loads associated with the most severe meteorological conditions, including extreme wind and tornado, which are postulated to occur at the storage site. Tornado design parameters used to evaluate the suitability of the Concrete Cask include high winds, wind generated pressure differentials and tornado generated missiles. NUREG 0800 (1987), Regulatory Guide (R.G.) 1.76 (1974), ANSI 57.9 (1984), ANSI A58.1 (1982), and National Defense Research Committee (NDRC) methodologies were used to guide the tornado analyses.

3.2.1.1 Applicable Design Parameters

The design basis tornado characteristics (consistent with R.G. 1.76 -1974) are presented in Table 3.2-1. The design values shown are for generic storage system design and bound the postulated wind and tornado conditions for the Trojan ISFSI site. The generic values are for Region I, as defined in R.G. 1.76, and are more conservative than those defined for Region III, in which the Trojan site is located.

3.2.1.2 Determination of Forces on Structures

Wind and tornado loadings are applied to the Concrete Cask in storage only. The Concrete Cask is designed to withstand the effects of wind loading, wind generated pressure differentials, and tornado generated missiles without toppling, or significant impact damage. The methods used to convert wind and tornado loadings into forces on the cask are based on NUREG-0800 (1987), Section 3.3.1 (wind loadings), Section 3.3.2 (tornado loadings and combined loadings), and Section 3.5.3 (barrier design procedures).

3.2.1.3 Ability of Structures to Perform Despite Failure of Structures Not Designed for Tornado Loads

Each Concrete Cask is designed to operate independently without reliance on support structures. The Concrete Cask is designed to withstand tornado loadings and maintain its required safety function.

During storage, each PWR Basket, GTCC Basket, or Basket Overpack is enclosed and protected by the Concrete Cask from tornado missiles. In addition, the baskets can withstand the atmospheric pressure drop (-3 psid) associated with tornadoes.

The Transfer Cask and Transfer Station are not designed for tornado loads. Administrative controls will be implemented to prevent transfer operations in the event of a tornado watch or tornado warning.

3.2.1.4 Tornado Missiles

The Concrete Cask is designed to withstand the effects of impacts associated with postulated tornado generated missiles as identified in NUREG-0800 (1987), Section 3.5.1.4.III.4. Spectrum I missiles are used and assumed to impact in a manner that produces the maximum damage to the cask. The design basis tornado generated missiles for the Trojan ISFSI are shown in Table 3.2-2. The effect of postulated impacts on the Concrete Cask is discussed in Section 8.2.4.2.2.

3.2.2 WATER LEVEL (FLOOD) DESIGN

The floods postulated for the Trojan ISFSI do not exceed the site elevation, therefore, flooding of the ISFSI is not a credible event. Trojan ISFSI site hydrology is discussed in Section 2.4.

3.2.2.1 Flood Elevations

The nominal ISFSI elevation is 45 ft. MSL. Section 2.4 discusses the potential flood levels for the ISFSI location. A summary of potential floods is as follows:

Standard Project Flood (SPF, 1000 yr.):	21 ft. MSL
Probable Maximum Flood (PMF):	39.2 ft. MSL
Seismically Induced Dam Failure:	42.75 ft. MSL

3.2.2.2 Flood Protection

There are no ISFSI storage system components classified as important to safety located at an elevation below the most limiting credible flood level (42.75 ft. MSL).

3.2.3 SEISMIC DESIGN

3.2.3.1 Input Criteria

ISFSI design criteria were evaluated using the Trojan Nuclear Plant Seismic Margin Earthquake (SME) as the design basis seismic event (Reference 2). The SME represents the maximum potential earthquake for the ISFSI site.

3.2.3.1.1 Design Response Spectra

The Trojan Seismic Margin Earthquake has the following peak conditions:

Horizontal ground acceleration:	0.38g
Vertical ground acceleration:	0.25g

The design earthquake response spectra are in accordance with Regulatory Guide 1.60 (Rev. 1, December 1973), with a maximum horizontal ground acceleration equal to the Trojan SME value. Damping values of Regulatory Guide 1.61 (Rev. 1, October 1973) were used.

A minimum factor of safety for overturning of 1.1 was used based on guidance provided in NUREG-0800 (1987), Section 3.8.5. This document provides criteria used in determining the acceptability in the design of Seismic Category I foundations at nuclear facilities.

3.2.3.1.2 Design Response Spectra Derivation

The response spectral shapes of Regulatory Guide 1.60 (Rev. 1, December 1973) are used for the design of structures important to safety. Earthquake time functions or other data, therefore, are not required for derivation in accordance with Regulatory Guide 3.48 (Rev. 1), Section 3.2.3.1.2.

3.2.3.1.3 Design Time History

The earthquake is defined by the response spectra, therefore time histories are not used.

3.2.3.1.4 Use of Equivalent Static Loads

The equivalent static loading is used for the storage system evaluation since the Concrete Cask is a very rigid body with natural frequencies in excess of the zero period acceleration cut-off. No dynamic amplification by the cask is expected. Refer to Section 3.2.3.2.1 for a discussion of seismic analysis methods.

3.2.3.1.5 Critical Damping Values

Damping values are developed in accordance with Regulatory Guide 1.61 (Rev 1, October 1973) for a Safe Shutdown Earthquake (SSE).

3.2.3.1.6 Bases for Site-Dependent Analysis

The SME was developed as part of the U.S. Nuclear Regulatory Commission's Individual Plant Examination for External Events (IPEEE) program. The results of this study focused on determining the maximum potential earthquake that could affect the site. The SME for the Trojan Site conservatively bounds the Design Basis Earthquake (DBE). In addition, the SME is a specified design criterion of Oregon Administrative Rule (OAR) 345-26-0390.

3.2.3.1.7 Soil-Supported Structures

The reinforced concrete pad on which the Concrete Cask rests is the only at-grade ISFSI structure. The pad is located directly on competent rock or shallow compacted gravel founded on competent rock.

3.2.3.1.8 Soil-Structure Interaction

The foundation rock contains joints and fractures, as do essentially all rocks exposed to the earth's surface. However, none of these features should be expected to affect the stability of the foundation rock during vibratory motion. The foundation rock is confined by natural, in situ materials, and foundation loads are small in comparison to the foundation rock's ultimate bearing capacity. There will be no loss of strength or stability of the foundation rock during vibratory motions. Since the reinforced concrete pad is located directly on competent rock or shallow compacted gravel, founded on competent rock, soil-structure interaction is negligible.

3.2.3.2 <u>Seismic-System Analyses</u>

3.2.3.2.1 Seismic Analysis Methods

The Concrete Cask is a very stiff structure. Its lowest natural frequencies exceed the zero period acceleration threshold. No dynamic amplification of the ground motion is expected from the cask. For the purpose of calculating seismic loads, the cask is treated as a rigid body, and equivalent static analysis methods were used to calculate loads, stresses, and overturning moments. Although free-standing, it is conservatively analyzed for stresses as a cantilever, fixed at the base.

The storage system is evaluated statically for overturning by conservatively applying static loads to the cask equivalent to the peak horizontal component acting simultaneously with the peak vertical component applied upward.

3.2.3.2.2 Natural Frequencies and Response Loads

The fundamental natural frequency of vibration for the cask was determined as shown below:

$$f_n = [(K_n)/2\pi]\ [(E)(I)(g)/(w)(L^4)]^{0.5}$$

where:

f_n = Frequency of the n-th mode

K_n = 3.52 for first mode of vibration

E = Modulus of Elasticity

I = Moment of Inertia

g = Gravitational acceleration

L = Height of cask

w = Uniform weight per unit length of cantilever

3.2.3.2.3 Procedure Used to Lump Masses

The storage system components are very rigid and no dynamic modal analysis needs to be performed as explained above. Thus, no mass lumping procedure was used.

3.2.3.2.4 Rocking and Translational Response Summary

The cask restoring moment is always higher than tipover moments from seismic loads. Therefore, no cask rocking is expected.

3.2.3.2.5 Methods to Determine Overturning Moments

The storage system has been evaluated conservatively by applying the static loads to the cask equivalent to the horizontal component acting simultaneously with the vertical component applied upward. For the SME ground accelerations listed in Section 3.2.3.1.1 the margin of safety against overturning was determined and shows that the cask will not overturn during the SME.

3.2.3.2.6 Analysis Procedure for Damping

No damping was assumed in the storage system.

3.2.3.2.7 Seismic Analysis of Overhead Cranes

There are no overhead cranes included in the ISFSI design. The use of mobile cranes is discussed in Section 4.7.3.6.

3.2.3.2.8 Seismic Analysis of Specific Safety Features

The ISFSI must be designed to meet the requirements of 10 CFR 72.122(b)(2) and OAR 345-26-0390. The ISFSI shall be capable of withstanding a SME without overturning the Concrete Cask or compromising the PWR Basket or GTCC Basket confinement boundary. A SME will not result in uncontrolled release of radioactive material or increased radiation exposure to workers or members of the general public.

3.2.4 SNOW AND ICE

The criterion for determining design snow loads is based on ANSI A58.1 (1982), Section 7.0. Flat roof snow loads apply and are calculated from the following formula:

$$p_f = 0.7 C_e C_t I p_g$$

where:

p_f = Flat roof snow load (psf)

C_e = Exposure factor = 0.8

C_t = Thermal factor = 1.0

I = Importance factor = 1.2

p_g = ground snow load, pounds per square foot = 100 psf

The numerical values of C_e, C_t, and I, are obtained from Tables 18, 19, and 20 respectively, of ANSI A58.1 (1982). A conservative value of 100 pounds per square foot was assumed for snow load (p_g).

The exposure factor accounts for wind effects. The Trojan site is assumed to have siting category A which is defined to be a "windy area with roof exposed on all sides with no shelter afforded by terrain, higher structures, or trees." The thermal factor accounts for the thermal condition of the structure. Due to the presence of substantial heat load in the storage system, it is classified as a heated structure. The storage system cask is conservatively classified as Category III which is the highest category in the ANSI standard. Ground snow loads for the contiguous United States are given in Figures 5, 6, and 7 of ANSI A58.1 (1982).

Based on the above, the design criterion for snow and ice loads is:

Flat Roof Snow Load $\quad p_f = (0.7)(0.8)(1.0)(1.2)(100)$

$= 67.2$ psf

3.2.5 COMBINED LOAD CRITERIA

The storage system components are subjected to normal, off-normal, and accident loads. These loads are defined as follows:

Normal Loads -	Dead weight, pressure, handling, thermal, winds, snow, rain
Off-Normal Loads-	Severe environmental conditions, interference during basket lowering from Transfer Cask to Concrete Cask, blockage of one-half of air inlets, off-normal handling
Accident Loads -	Complete blockage of air inlets, maximum heat load, basket drop into shipping cask or Concrete Cask, tornado (wind and missiles), flood, seismic, fuel pin rupture (for the basket)

Normal loads due to pressure, temperature, and dead weight act in combination with other loads. No two unrelated accident events are postulated to occur simultaneously. However, loads due to one event, such as tornado wind and tornado missile loads, are assumed to act in direct combination.

3.2.5.1 Load Combinations and Design Strength - Concrete Cask

The load combinations specified in ANSI 57.9 (1984) and ACI-349 (1985) for concrete structures were used and are shown in Table 3.2-3. The steel liner, air ducts and bottom plate of the Concrete Cask are stay-in-place forms. They are designed to ACI 349 (1985) requirements for steel forms and reinforcement.

3.2.5.2 Load Combinations and Design Strength - PWR Basket, GTCC Basket, and Basket Overpack

The PWR Basket, GTCC Basket and Basket Overpack are designed to the 1992 edition of the ASME Boiler and Pressure Vessel Code, Section III. The pressure retaining components (shell and lids) are designed to Section III, Subsection NC. The PWR Basket internals are designed to Section III, Subsection NG. Buckling/instability of the internal members are evaluated according to the requirements of NUREG/CR-6322. The load combinations for normal, off-normal, and accident conditions and corresponding Service Levels are shown in Table 3.2-4.

Service Levels A and C, as specified by the ASME Code, are used for normal and off-normal conditions, respectively. The analyses methods allowed by the ASME Code are employed. Stress intensities caused by pressure, temperature and mechanical loads are combined before comparison to ASME code allowables.

Service Level D, as specified by the ASME Code, is used for accident conditions. The design strength criteria for the evaluation of Service Level D events are that any damage to the basket will not prevent the basket from performing the following safety functions: heat dissipation, criticality prevention, confinement, and shielding. Limited plastic deformation is allowed. Stresses caused by normal condition loads are combined with the stresses caused by accident or off-normal loads. These stresses are then compared to the stress limits defined in Appendix F of the ASME Code.

The PWR Basket design weight is based on 24 intact fuel assemblies, each containing a RCCA (149 lbs.). The maximum GTCC waste loading is limited to 29,000 lbs. per GTCC Basket. The maximum heat loading per GTCC Basket is bounded by the fuel analysis. The structural design criteria are summarized in Table 3.2-5.

3.2.5.3 Load Combinations and Design Strength - Transfer Cask

The Transfer Cask is a special lifting device designed and fabricated to the requirements of ANSI 14.6 (1993) and NUREG 0612 (1980). The criteria for its load-bearing components are:

Maximum principal stress during the lift (with 10% dynamic load factor) will be less than $S_y/3$ or $S_u/5$.

Load bearing members of the Transfer Cask shall be subject to drop weight test (ASTM E208) or charpy impact test (ASTM A370) per ANSI 14.6 (1993) paragraph 4.2.6.

3.2.5.4 Load Combinations and Design Strength - Failed Fuel Can

The Failed Fuel Can is designed to be placed into one of the four corner storage locations of the PWR Basket. The Failed Fuel Can is designed to allow for water draining and vacuum drying during PWR Basket closure. Once placed into its storage location, the Failed Fuel Can is not subjected to external loadings applicable to ASME Service Level A (normal). Specific structural design criteria and load combinations are not applicable.

3.2.5.5 Load Combinations and Design Strengths - Fuel Debris Can

The Fuel Debris Can is designed to ASME Section III, Subsection NG (1992) for Service Level A (normal) conditions. The Fuel Debris Can must withstand stresses caused by external pressure as well as internal hydrostatic test pressure. Structural design criteria are summarized in Table 3.2-5.

3.3 SAFETY PROTECTION SYSTEMS

3.3.1 GENERAL

The ISFSI is designed for safe, long-term storage of the radioactive material described in Section 3.1.1. The ISFSI withstands normal, off-normal and credible accident conditions without release of radioactive material to workers or members of the general public.

The primary functions of the PWR Basket (including internals) are:

1. To provide containment and confinement for the spent nuclear fuel during normal storage, off-normal events and postulated accidents,
2. To provide criticality control in the absence of a moderator under design conditions and postulated accidents,
3. To provide adequate heat transfer so that the fuel clad temperature does not exceed allowables under design conditions and postulated accident conditions, and
4. To provide adequate shielding (together with a Concrete Cask or Transfer Cask) to meet 10 CFR 72 requirements.

The primary functions of the GTCC Basket are:

1. To provide containment and confinement of the GTCC waste under design conditions and postulated accidents, and
2. To provide adequate shielding (together with a Concrete Cask or Transfer Cask) to meet 10 CFR 72 requirements.

The primary functions of the Basket Overpack are:

1. To provide the same containment and confinement function as a PWR Basket or GTCC Basket in the highly unlikely case that either basket develops a leak, and

2. To provide adequate heat transfer so that the fuel clad temperature does not exceed allowables under design conditions.

The primary functions of the Concrete Cask are:

1. To protect the basket from weather and postulated environmental events such as earthquakes and tornado missiles,
2. To provide adequate heat transfer for the PWR Basket and GTCC Basket, and
3. To provide adequate shielding (together with a PWR Basket or GTCC Basket) to meet 10 CFR 72 requirements.

The primary functions of the Transfer Cask are:

1. To serve as a special lifting device meeting the requirements of NUREG-0612 (1980)/ANSI 14.6 (1993) for movement of a PWR Basket or GTCC Basket, and
2. To provide radiation shielding to minimize exposure rates during transfer operations.

The primary function of the Transfer Station is to prevent the Transfer Cask from falling or overturning during Basket transfer operations.

The primary function of the Failed Fuel Can and Fuel Debris Can is to provide a containment boundary for failed fuel assemblies and fuel debris such that this material will be constrained within its PWR Basket storage location. Constraining this material to fixed storage locations is required to maintain the assumptions in the criticality analysis and heat transfer modeling.

As discussed in the following sub-sections, the ISFSI design incorporates features addressing each of the above design considerations to assure safe operation during fuel loading, storage system handling, and storage.

3.3.2 PROTECTION BY MULTIPLE CONFINEMENT BARRIERS AND SYSTEMS

3.3.2.1 Confinement Barriers and Systems

Oregon Administrative Rule (OAR) 345-26-0390 prohibits the storage of spent nuclear fuel or radioactive materials other than that generated or used in the operation of the Trojan Nuclear Plant. Spent nuclear fuel and fuel related material will be confined within PWR Baskets. GTCC waste will be confined within GTCC Baskets.

The PWR Basket and GTCC Basket are designed to provide a confinement barrier for spent nuclear fuel and GTCC waste in accordance with the general design criteria requirements of 10 CFR 72, Subpart F. Each basket is a stainless steel seal welded enclosure. The basket shield lid and structural lid closures are accomplished by multi-pass welding. The PWR Basket and GTCC Basket confinement barriers are designed in accordance with ASME, Section III, Subsection NC (1992).

The PWR Basket internals which are used to constrain fuel assemblies, Failed Fuel Cans, and the Fuel Debris Can during storage are designed in accordance with ASME, Section III, Subsection NG (1992). The PWR Basket internals provides 24 storage locations. The four (4) corner locations are designed slightly larger to accommodate a Failed Fuel Can or Fuel Debris Can.

The Fuel Debris Can provides an additional confinement boundary for fuel debris within the PWR Basket. It is designed in accordance with ASME, Section III, Subsection NG (1992).

The Failed Fuel Cans and GTCC Cans do not provide a confinement boundary and are considered to function as part of the basket internals. The Failed Fuel Can is designed in accordance with applicable portions of ASME, Section III, Subsection NG (1992). The GTCC Can is designed in accordance with applicable portions of ASME, Section III, Subsection NF (1992).

In the unlikely event of a PWR Basket or GTCC Basket confinement boundary failure, the affected basket may either be repaired or sealed within a Basket Overpack. The design criteria for the Basket Overpack are the same as those specified for the PWR Basket and GTCC Basket confinement boundary.

The PWR Basket must be designed to withstand credible drop accidents without damaging the stored fuel (i.e., the storage cells do not deform such that they bind the fuel). The PWR Basket must also be designed to provide confinement in the event of a fuel clad failure.

3.3.2.2 Ventilation-Offgas

The ISFSI is designed to confine radioactive materials within a sealed enclosure for the life of the facility. There are no radioactive releases during normal operations or credible accidents. In the unlikely event a leaky basket must be placed in a Basket Overpack, evacuation of the Basket Overpack and backfilling with helium would be required. The operation is discussed in Chapter 5. A suitable filtration system such as a high efficiency particulate air (HEPA) filter would be used for the vacuum system vent path during this evolution.

3.3.3 PROTECTION BY EQUIPMENT AND INSTRUMENTATION SELECTION

3.3.3.1 Equipment

The equipment/components that have been identified as important to safety for the ISFSI are:

1. Concrete Cask,
2. PWR Basket,
3. GTCC Basket,
4. Basket Overpack,
5. Fuel Debris Can,
6. Failed Fuel Can,
7. Transfer Cask, and
8. Transfer Station

The design criteria for the PWR Basket, GTCC Basket, and Basket Overpack are summarized in Table 3.2-5. The design criteria for the Concrete Cask are summarized in Table 3.6-1.

3.3.3.2 Instrumentation

A temperature monitoring device is provided for each of the air outlet vents per storage cask (four per cask). The temperature monitoring devices are commercial grade. Additional discussion of temperature monitoring is provided in Section 5.1.3.4 and Section 5.4.1.

3.3.4 NUCLEAR CRITICALITY SAFETY

The storage system is designed to maintain subcritical conditions ($K_{eff} \leq 0.95$) under normal handling and storage conditions, off-normal handling and component functioning, and hypothetical accident conditions.

3.3.4.1 Control Methods for Prevention of Criticality

Subcritical conditions are to be maintained by PWR Basket internal geometry. The PWR Basket internals will establish fuel assembly spacing. The design will assume a fuel assembly enrichment equal to or greater than the maximum initial fuel assembly enrichment that will be stored (3.56 wt% U^{235}). No credit will be taken for burnup or fuel assembly control inserts. Although neutron absorbing material is incorporated into the PWR Basket internals design, it is not credited in the criticality analysis for dry storage conditions.

Table 3.1-1 lists the fuel characteristics. Loose pellets and fuel debris are placed in a Fuel Debris Can. Administrative controls limit the amount of fuel debris which can be placed within a Fuel Debris Can.

There are no criticality control requirements for the GTCC Baskets.

3.3.4.2 Error Contingency Criteria

The values of K_{eff} include error contingencies and calculational and modeling biases. K_{eff} equals the calculated K_{eff}, plus criticality code bias, plus two times code uncertainty. K_{eff} has been evaluated for infinite cask array and optimum assembly position within the cell.

3.3.4.3 Verification Analyses

The criticality analysis was performed using an NRC accepted code. The code was validated in accordance with SNC Quality Assurance Program (Reference 3).

3.3.5 RADIATION PROTECTION

3.3.5.1 Access Control

Personnel exposure and access to radioactive material is minimized by the ISFSI design.

Personnel exposure is minimized by incorporating shielding into the design of the PWR Baskets, GTCC Baskets, Transfer Cask, and Concrete Casks. Chapter 7 discusses the administrative procedures designed to limit personnel exposure. In addition, physical access to the ISFSI is restricted by a security fence. Access to the facility is limited to those persons who have satisfactorily completed access training or are escorted by a person who has satisfactorily completed access training. Training requirements for facility access are discussed in Section 9.3.

In addition to the security fence which restricts access to the ISFSI, access to stored radioactive material is further prevented by facility design as the radioactive materials are confined within welded steel enclosures (PWR Basket or GTCC Basket).

3.3.5.2 Shielding

The storage system in conjunction with appropriate administrative control is designed to maintain radiation exposure As Low As Reasonably Achievable (ALARA). The storage system is designed to provide an average external side surface dose (gamma and neutron) of less than 100 mrem/hr on the sides and 200 mrem/hr on the top and at the air vents. The design maximum dose rate at the top of the basket structural lid is 200 mrem/hr to allow limited personnel access during basket closure operations.

Expected dose rates associated with ISFSI operations are contained in Section 7.4.

3.3.5.3 Radiological Alarm Systems

The storage cask system does not produce routine solid, liquid, or gaseous effluents. Chapter 8 discusses an inadvertent release of surface contamination from the exterior of the basket. The consequences of this event are negligible (2.4 mrem at 100 meters). Therefore, an alarm for airborne radioactivity is not required to protect personnel or the environment.

The estimated working dose rates for the PWR cask (maximum fuel burnup) and GTCC cask are 10 mrem/hr and 31 mrem/hr, respectively, and the dose at 100 meters (based on a 2000 hour occupancy) is 24 mrem/yr as described in Chapter 7. These dose rates do not warrant a radiation alarm to protect personnel or the environment.

Based on the above, radiological alarms are not required for the Trojan ISFSI.

3.3.6 FIRE AND EXPLOSION PROTECTION

The potential for fires at the ISFSI are minimized by the use of paved open areas and minimum combustible materials within the ISFSI security fence. As discussed in Section 2.2.3.3 the facility is well protected from industrial and forest fires by natural barriers. Sections 8.2.9 and 8.2.14.2.2 provide additional discussion on fires.

Explosion analyses for the ISFSI are presented in Sections 8.2.8, 8.2.14.2.3 and 8.2.14.2.4

3.3.7 MATERIALS HANDLING AND STORAGE

3.3.7.1 Spent Fuel or GTCC Waste Handling and Storage

The loading of each PWR Basket is limited to a decay heat load of $24KW_t$ per basket with no individual fuel assembly heat generation exceeding $1.08KW_t$. The PWR Basket heat load conservatively bounds that which can be produced by a loaded GTCC Basket. The ISFSI is designed to accommodate decay heat loads of 26 KW_t and maintain fuel cladding temperature below limits established for inert dry storage (Reference 4). In addition, temperature limits for storage system components must also be maintained below design limits. The Technical Specifications establish surveillances to preclude exceeding material design temperature limits.

The fuel clad temperature limit is a function of fuel burnup, fuel pin fill gas pressure, and fuel age. For the Trojan ISFSI, the fuel clad temperature limit was determined to be 388°C (730°F). This limit was determined using Westinghouse 17x17 fuel with 42,000 Mwd/MTU and 5 years of cooling.

Concrete Cask temperature limits are based on guidance provided by ACI-349 (1985). Section 4.2.6 presents the thermal analysis of the Concrete Cask and PWR Basket for the anticipated range of operating conditions.

The design criteria for maintaining subcritical condition are presented in Section 3.3.4.

Contamination control is addressed in Section 3.3.2.

Although no credible events result in loss of confinement due to damage to a PWR Basket or GTCC Basket, provisions for this unlikely occurrence are provided in the ISFSI design. If the affected basket cannot be repaired, the affected basket can be removed from the Concrete Cask and placed in a Basket Overpack. The Basket Overpack is designed and fabricated to the same criteria as the PWR Basket and GTCC Basket. The design of the Basket Overpack must also ensure fuel clad temperature limits and Concrete Cask temperature limits are not exceeded.

Operations involving the Basket Overpack are presented in Chapter 5. Design analyses for the Basket Overpack are presented in Chapter 4.

The ISFSI shall have a minimum design life of 40 years in accordance with the requirements of Oregon Administrative Rule 345-26-0390(4)(j).

3.3.7.2 Radioactive Waste Treatment

ISFSI operations do not result in the generation of liquid or gaseous radioactive waste. Although generation of solid radioactive waste is not expected, it is possible that small quantities of low level radioactive waste could be generated as a result of routine radiological surveys (e.g., swipes). This waste would be stored in an appropriate container pending disposal.

3.3.7.3 Waste Storage Facilities

Section 6.3 discusses waste storage facilities for the ISFSI.

3.3.8 INDUSTRIAL AND CHEMICAL SAFETY

There are no required special industrial or chemical design criteria that are important to personnel or plant safety.

3.4 CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Structures, systems, and components important to safety are those features of the ISFSI whose function is:

1. To maintain the conditions required to store spent fuel or high-level radioactive waste safely,

2. To prevent damage to the spent fuel or the high-level radioactive waste container during handling and storage, or

3. To provide reasonable assurance that spent fuel or high-level radioactive waste can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

ISFSI structures, systems and components important to safety are identified in Section 3.3.3.1.

3.5 DECOMMISSIONING CONSIDERATIONS

Decommissioning activities consist primarily of transferring the PWR Baskets and GTCC Baskets from the Concrete Casks into a shipping cask. The PWR Baskets and GTCC Baskets are then shipped off-site for disposal or storage.

The storage system has been designed to minimize contamination of the Concrete Cask exterior during loading and unloading operations. Although no contamination of the Concrete Cask is expected, the interior steel liner can be decontaminated and the complete Concrete Cask broken up (or left whole) and shipped to a landfill.

3.6 SUMMARY OF DESIGN CRITERIA

A summary of the basket design criteria is presented in Table 3.2-5. A summary of the Concrete Cask design criteria is presented in Table 3.6-1.

3.7 REFERENCES

1. SNC-95-71 SAR, Revision 0, "Safety Analysis Report for the TranStor™ Shipping Cask System," Sierra Nuclear Corporation, December 1995.

2. "Seismic Margin Earthquake Study," letter from PGE to NRC dated May 26, 1993, Docket 50-344.

3. Sierra Nuclear Corporation Quality Assurance Program for the Ventilated Storage Cask System, Pacific Sierra Nuclear Associates and Sierra Nuclear Corporation, October 1991. Docket Number 72-1007.

4. E.R. Gilbert et al., "Control of Degradation of Spent LWR Fuel During Dry Storage in an Inert Atmosphere," PNL-6364, Pacific Northwest Laboratory, Richland, WA (1987)

Table 3.1-1

Fuel Characteristics

	Parameters	Westinghouse 17x17	B&W 17X17
Fuel Assemblies:	UO_2 rods per assembly	264	264
	Rod Pitch (in)	0.496	0.496
	Overall dimensions (in)	8.426 x 8.426	8.426 x 8.426
	Fuel Weight, (lb U/ assy)	1154	1129
	Structural Weight (excludes fuel)	313	302
	Total Fuel Assy Weight (dry)	1467	1431
Fuel Rods:	Outside diameter (in.)	0.374	0.374
	Clad thickness (in.)	0.0225	0.024
	Active fuel length (in.)	144	144
	Clad Material	Zircaloy-4	Zircaloy-4
Fuel Pellets:	Material	UO_2 sintered	UO_2 sintered
	Density (% of theoretical)	95	95
	Diameter (in)	0.3225	0.3195
	Maximum Enrichment	3.37 wt% U^{235}	3.56 wt% U^{235}

Table 3.1-2

Design Maximum Radiological Characteristics of Stored Material

Characteristic	Value
Fuel:	
Maximum burnup 5 yr. cooled[1]	40,000 MWD/MTU
Maximum burnup 6 yr cooled[2]	45,000 MWD/MTU
Initial Enrichment[3]	3.02 wt% U^{235} (for 40,000 MWD/MTU) 3.30 wt% U^{235} (for 45,000 MWD/MTU)
Gamma Source (24 fuel assemblies)	1.856E+17 γ/sec
Neutron Source (24 fuel assemblies)	1.188E+10 η/sec
GTCC Waste (Per GTCC Basket):	
Weight	29,000 lbs
Activity	1.14E+6 curies

1 The 5 year cooled fuel results in the most conservative gamma source term.

2 The 6 year cooled fuel results in the most conservative neutron source term.

3 Low initial enrichments will yield higher gamma and neutron source terms for a given burnup. The enrichment values provided bound the Trojan ISFSI spent fuel inventory.

Table 3.1-3

Design Maximum Thermal Characteristics of Stored Material

Characteristic

Fuel[1]	Value
Decay heat per assembly	1.08 KW_t
Decay Heat per PWR Basket	26 KW_t
Maximum Burnup (5 year cooled)	42,000 MWD/MTU
Clad Temperature	
Long Term Limit	388°C (730°F)
Accident Limit	570°C (1058°F)

1 The thermal characteristics for the fuel bound the values for GTCC waste.

Table 3.1-4

Physical Parameters of Fuel Assembly Inserts

RCCA	Neutron Absorber	Ag-In-Cd
	Cladding Material	304 SS
	Number of Absorber Rods per Assembly	24
	Number of Assemblies	61
	Weight of Assembly	149 lbs.
	Overall Length	161 in.
BPRA	Cladding Material	304-SS
	Number of Absorber Rods per Assembly	Varies between 9 and 20
	Weight of Assembly	Weight is bounded by Weight of RCCA
	Overall Length	161 in.
	Number of BPRA	90
Thimble Plugs	Material	Zircaloy
	Weight	Bounded by Weight of RCCA
Source	Cladding Material	304-SS
	Weight	Bounded by Weight of RCCA

Table 3.2-1

Wind and Tornado Design Specifications

Environmental Condition	Value
Rotational Wind Speed, mph	290
Translational Speed, mph	70
Maximum Wind speed, mph	360
Radius of Max. Wind Speed, ft.	150
Pressure Drop, psi	3.0
Rate of Pressure Drop, psi/sec	2.0

Table 3.2-2

Design Basis Tornado Generated Missiles

Missile Description	Weight (lbs.)	Velocity (mph)
Automobile	3960	126
Armor Piercing Shell (8 in. diameter)	275	126
Steel Sphere (1 in. diameter)	0.22	126

Table 3.2-3

Design Load Combinations

Load Comb	Dead	Live	Wind	Norm/ Acc Temp	Seismic	Tornado Missile	Tornado wind	Soil Pressure
1	1.4D	+1.7L						
2	1.4D	+1.7L						+1.7H
3	0.75(1.4D	+1.7L	+1.7W	$+1.7T_o$				+1.7H)
4	0.75(1.4D	+1.7L		$+1.7T_o$				+ 1.7H)
5	D	+ L		$+ T_o$	$+E_{ss}$			+ H
6	D	+ L		$+ T_o$		+A		+ H
7	D	+ L		$+ T_a$				+ H
8	D	+ L		$+ T_a$			$+W_t$	+ H

D	=	Dead Load	T_a	=	Accident Temperature Load
L	=	Live Load	E_{ss}	=	Earthquake Load
W	=	Wind Load	W_t	=	Tornado Missile Load
T_o	=	Normal Temperature Load	A	=	Impact/Cask Tipover Load
H	=	Soil Pressure Load			

Table 3.2-4

Summary of Load Combinations

LOAD		Normal			Off-Normal			Accident							
ASME Service Level		A			C			D							
Load Combination Number[1]		1	2	3[2]	1	2[2]	3	1[3]	2	3	4	5[2]	6	7	8
Dead Weight	Basket with fuel	X	X	X	X	X	X	X	X	X	X	X	X	X	X
Thermal	Inside Concrete Cask: 75°F	X		X		X		X	X	X	X	X			
	Inside Transfer Cask: 75°F		X		X								X		
	Inside Concrete Cask: -40°F or 100°F						X								
	Inside Concrete Cask: Max Heat Load (125°F)													X	X
Pressure	Normal	X	X	X	X	X		X	X	X	X			X	X
	Accident											X	X		
Handling Load	Normal		X	X								X	X		
	Off-Normal				X	X	X								
Drop (Vertical or Horizontal)								X							
Seismic									X						
Flood										X					
Tornado											X				

[1] Load Combination Number corresponds to load combinations provided on Table 3.2-3.

[2] Controlling load combination for the Service Level.

[3] Load combination included to address basket drop within Transfer Cask which is bounded by the 10 CFR 71 drop analysis for PWR Basket (Reference 1).

Table 3.2-5

PWR Basket, GTCC Basket, and Basket Overpack Design Criteria

Component (Applicable Code or Criteria)	Criteria
PWR Basket, GTCC Basket, or Basket Overpack Normal Operation - Service Level A ASME Section III, Subsection NC (shell) ASME Section III, Subsection NG (internals)	$P_m < 1.0\ S_m$ $P_L + P_b < 1.5\ S_m$ $P + Q < 3.0\ S_m$
PWR Basket, GTCC Basket, or Basket Overpack Off-Normal Operation - Service Level C ASME III, Subsection NC (shell) ASME III, Subsection NG (internals)	$P_m < 1.2S_m$(shell), $< 1.5S_m$ (cells) $P_L + P_b < 1.8S_m$(shell), $< 2.25S_m$ (cells)
PWR Basket, GTCC Basket, or Basket Overpack Accident Condition - Service Level D ASME III, Subsection NC (shell) ASME III, Subsection NG (internals)	$P_m < 2.4\ S_m$ or $0.7\ S_u$ (whichever is less) $P_L + P_b < 3.6\ S_m$ or $1.0\ S_u$ (whichever is less)
Fuel Debris Can Normal Operation - Service Level A ASME Section III, Subsection NG	$P_m < 1.0S_m$ $P_L + P_b < 1.5\ S_m$

Table 3.6-1

Summary of Concrete Cask Design Criteria

Design Load Type	Design Parameters	Applicable Criteria and Codes
Tornado	360 mph, maximum 290 mph, rotational 70 mph, translational 3.0 psi, pressure drop 2.0 psi/sec, rate of pressure drop	NRC Reg. Guide 1.76 (1974) ANSI A58.1 (1982)
Tornado Missile	At 126 mph: Automobile 8 in dia. shell 1 in solid sphere	NUREG O800 (1987) Section 3.5.1.4
Flood	Not Applicable -- ISFSI elevation above credible flood levels	10 CFR 72.122
Seismic	0.38g horizontal acceleration 0.25g vertical acceleration	10 CFR 72, NRC R.G. 1.60 Rev. 1, NRC R.G. 1.61 Rev. 1
Dead Loads	Dead weight, including basket weight (concrete density at 145 lb/ft^3)	ANSI 57.9 (1984)
Air Temperature	-40°F minimum 100°F maximum 125°F short term extreme	10 CFR 72.122
Concrete Temperature	250°F- normal 350°F - accident	ACI-349 (1986)[1]
Snow and Ice Loads	67.2 psf included in live loads	ANSI A58.1 (1982) and ANSI 57.9 (1984)

1 ACI-349 (1986) limits maximum long term concrete temperature to 200°F. Refer to Section 4.2.4.2.4 for discussion of elevated limit.

4.0 INSTALLATION DESIGN

This chapter provides a description of the ISFSI including the installation layout, major components, handling equipment, and auxiliary systems. It also provides a summary of the analysis performed to demonstrate compliance with design requirements presented in Chapter 3. Installation design, analysis, and fabrication are covered by the SNC Quality Assurance Program and the PGE Nuclear Quality Assurance Program as addressed in Chapter 11.

4.1 SUMMARY DESCRIPTION

4.1.1 LOCATION AND LAYOUT OF INSTALLATION

The location of the ISFSI site is in Columbia County in Northwest Oregon by the Columbia River and is shown in Figure 2.1-1. Figures 1.1-2 and 2.1-2 show the location of nearby structures, roadways, railways, and rivers. The ISFSI layout is shown in Figure 2.1-3.

4.1.2 PRINCIPAL FEATURES

4.1.2.1 Site Boundary

The PGE owned property area is shown in Figure 2.1-2.

4.1.2.2 Controlled Area

The controlled area established by the criterion in 10 CFR 72.106 is shown in Figure 2.1-2.

4.1.2.3 Site Utility Supplies and Systems

The ISFSI design relies on the natural circulation of air to provide cooling of the spent nuclear fuel. Heat generated by the spent nuclear fuel is transferred to the air located in the Concrete Cask annulus. This heated air rises and exits via air outlets (4) located near the top of the Concrete Cask. Ambient air enters via air inlets (4) located at the bottom of the Concrete Cask. This passive design eliminates the need for utilities to support storage conditions. Electrical power is provided for security requirements which are addressed in the ISFSI Physical Security Plan. Electrical power is also available to support instrumentation such as temperature monitoring. Water and sewage utilities are not required or provided for the ISFSI.

4.1.2.4 Storage Facilities

Section 2.2 discusses the location of nearby storage facilities. Figures 2.2-1 and 2.2-2 show the location of these facilities.

4.1.2.5 Stacks

There are no stacks required for the operation of the ISFSI.

4.2 STORAGE STRUCTURES

This section provides a description of the ISFSI installation and major components, selected design criteria, materials of construction, fabrication summary, and quality assurance activities. Thermal, and criticality evaluations under normal and off-normal storage conditions are summarized. The shielding analysis is presented in Chapter 7, and the accident analysis is presented in Chapter 8. ISFSI handling structures and components are addressed in Section 4.7.

4.2.1 STRUCTURAL SPECIFICATION

The design criteria of the storage structures and components account for both normal and off-normal conditions, including a range of credible and postulated accidents. The principal design criteria for the ISFSI is in accordance with Title 10, Code of Federal Regulations, Part 72 (10 CFR 72), and ANSI/ANS 57.9. The design criteria for the ISFSI is presented in Chapter 3. The design codes for the major ISFSI storage structures and components are summarized in the following table.

Component	Governing Code/Standard[1]
Storage Pad	ACI 318 (1983)
PWR Basket	Confinement boundary - ASME, Section III, Subsection NC
	Internal assembly - ASME, Section III, Subsection NG
GTCC Basket	Confinement boundary - ASME, Section III, Subsection NC
Basket Overpack	ASME, Section III, Subsection NC
Fuel Debris Can	Confinement boundary - ASME, Section III, Subsection NG
Failed Fuel Can	ASME, Section III, Subsection NG
Concrete Cask	ACI 349 and ANSI 57.9

Section 3.4 provides the criteria used to classify structures, systems and components, important to safety.

The ISFSI Storage Pad meets the requirements of ACI 318 and is capable of supporting the loads associated with the array of Concrete Casks and transfer equipment. The ISFSI Storage Pad is not classified as important to safety. Its function is to provide a slab-on-grade supporting surface for the Concrete Casks, Transfer Station and shipping cask. It also provides a smooth level surface to allow operation of the air pad system.

[1] Applicable revision of governing code/standard is provided in Chapter 3.

The storage pad is conservatively designed as a beam system on elastic foundation for bounding case loading combinations per ACI-318, including consideration of seismic components associated with a Seismic Margin Earthquake (SME).

The GTCC Cans are not important to safety and are only used to facilitate loading GTCC waste into the GTCC Baskets.

The remaining structures listed above are considered important to safety. Structural evaluations presented in Section 4.2.5 demonstrate compliance with the above codes and standards. The PWR Basket, GTCC Basket and Basket Overpack are fabricated and inspected to the requirements summarized in Table 4.2-1. The Concrete Cask is fabricated and constructed to the requirements summarized in Section 4.2.4.2.4 and Table 4.2-2.

4.2.2 INSTALLATION LAYOUT

4.2.2.1 Building Plans and Sections

The Trojan ISFSI is an open-air facility. The installation layout is presented in Figure 2.1-3. The higher dose rate Concrete Casks are located toward the center of the Storage Pad in order to minimize dose rates at the protected area boundary. Chapter 7 discusses anticipated exposures associated with ISFSI operations.

The Storage Pad is designed and constructed in accordance with ACI-318 (1983). The Storage Pad consists of a reinforced concrete pad approximately 170 feet long by 100 feet wide. The Storage Pad will be constructed on competent rock or shallow compacted fill on competent rock and will have an approximate thickness of 18 inches.

4.2.3 CONFINEMENT FEATURES

The PWR Basket and GTCC Basket provide the primary confinement boundary for spent nuclear fuel and GTCC waste respectively. In the unlikely event of a PWR Basket or GTCC Basket confinement boundary failure, the affected basket may either be repaired or sealed within a

Basket Overpack. Section 3.3.2 discusses the design criteria applicable to these ISFSI components.

The cladding of intact fuel assemblies provides an additional confinement boundary. The Fuel Debris Can provides a confinement boundary for fuel fragments and is stored within a PWR Basket. The Failed Fuel Cans provide a containment boundary for failed fuel assemblies to constrain these assemblies and associated components within its PWR Basket storage location. Constraining this material to fixed storage locations is required to maintain the assumptions in the criticality analysis and heat transfer modeling.

The design requirements for confinement barriers and systems are further discussed in Section 3.3.2.

4.2.4 INDIVIDUAL UNIT DESCRIPTION

The ISFSI is comprised of up to 36 individual storage systems. Each storage system consists of a Concrete Cask containing either a PWR Basket or GTCC Basket. In the unlikely event a PWR Basket or GTCC Basket fails to maintain a confinement boundary and can not be repaired, a Basket Overpack is available. The Concrete Casks are arranged on the Storage Pad as discussed in Section 4.2.2.1.

4.2.4.1 Functional Description

The primary functions of the ISFSI storage system components are discussed in Section 3.3.1.

4.2.4.2 Component Descriptions

4.2.4.2.1 Description of the PWR Basket

The PWR Basket is a transportable cylindrical container consisting of an outer shell assembly, a shield lid, a structural lid, and an internal basket assembly. The basket shell provides the confinement boundary and is designed to withstand credible accidents without loss of integrity.

The shell exterior is coated with a gloss epoxy coating for ease of decontamination following loading operations.

The PWR Basket internal assembly is fabricated from steel plates formed into an array of 24 square storage cells. Four (4) of the outer corner cells are slightly larger to allow accommodation of a Failed Fuel Can or Fuel Debris Can. Intact fuel assemblies, with or without inserts, may be stored in any of the storage locations. The internal assembly uses structural tubes to provide support for the storage cells during a postulated drop accident. Neutron absorbing poison sheets are also used in the construction of the PWR Basket internal assembly, however they are not credited in the criticality analysis for dry storage conditions.

Section 5.1.1 discusses the operations associated with basket loading and installation of the shield lid and structural lid. The steel shield lid contains two penetrations to allow for vacuum drying and helium backfilling of the basket internal atmosphere. Prior to lowering the shield lid onto the basket after loading is complete, a pipe is threaded into one of the two penetrations. When the shield lid is in place, the pipe length is such that it extends to the bottom of the basket to facilitate water removal. Upon completion of water removal, a pipe plug is threaded into the drain pipe penetration. The other penetration utilizes a quick disconnect fitting to allow connection to a vacuum drying and helium backfilling system. The shield lid is seal welded to the basket shell. The first and final shield lid weld passes are inspected by dye penetrant testing. The basket is then hydrostatically tested at approximately 7.3 psig.

Following completion of hydrostatic testing, a steel structural lid containing a penetration allowing access to the shield lid penetrations is placed on top of the shield lid and seal welded to the basket shell and to the shield lid (where exposed by the structural lid penetration). The first and final weld passes are dye penetrant checked.

Upon completion of the vacuum drying and helium inerting of the internal basket atmosphere, the shield lid and structural lid penetrations must be sealed. The quick disconnect fitting is relied upon to maintain the helium atmosphere until the penetration closure plates are installed. The shield lid penetrations are isolated by two steel plates inserted into the structural lid access penetration. The steel plates are inserted individually and seal welded to the sides of the structural lid penetration. The first and final weld passes for each of these closure plates are dye penetrant checked.

The vent port lid welds which form part of the confinement barrier, are not hydrostatically tested. This is considered acceptable based on the specified methods of construction and intended application. The bases for this conclusion are summarized as follows:

1. The welds in question employ a double weld design providing redundant barriers,
2. The calculated internal basket pressure during normal service is approximately atmospheric resulting in negligible pressure stresses,
3. The welds in question are dye penetrant tested thereby providing assurance that the weldment is free of unacceptable imperfections, and
4. Neither operating nor environmental conditions are expected to subject the welds to cyclic loading.

Additional testing of the shield welds and structural welds is performed prior to installation of the shield lid penetration closure plates. The internal atmosphere is pressurized with helium and the welds are leak checked with a helium leak detector. This method of leak detection is very sensitive providing assurance that the shield and structural lid welds are providing a gas tight seal.

The PWR Baskets are designed, fabricated, and tested to provide a confinement barrier for spent nuclear fuel and GTCC waste in accordance with the general design criteria requirements of 10 CFR 72 Subpart F. Although the baskets will not be N-stamped in accordance with ASME Section III, NC-8100 (1992), basket design, fabrication and testing controls must be approved by the NRC prior to commencing ISFSI operations. Design, fabrication, and testing of baskets will be performed in accordance with a Quality Assurance Program meeting the applicable requirements of 10 CFR 72 Subpart G as presented in Chapter 11.

Table 4.2-1 presents a summary of fabrication requirements. Figure 4.2-1 provides a description of the PWR Basket.

4.2.4.2.2 Description of the GTCC Basket

The GTCC basket shell is similar to that of the PWR Basket shell and is built to the same requirements. The internal structure, however, consists of a steel shell that sits freely in the main confinement boundary shell. This steel shell is designed and fabricated to selected portions of ASME Section III, Subsection NF requirements This additional shell provides stability for the inserted GTCC cans and additional shielding needed because the gamma flux spectrum from the activated metal is higher than that of the fuel. The shield lid for the GTCC basket consist of lead held between steel plates. The GTCC Basket structural lid is similar to that used for the PWR Basket.

The GTCC Basket does not use an internal assembly to constrain the GTCC Cans. During loading, a 28 slot grating is placed on top of the internal shell and used to guide the GTCC cans in place. After the loading is complete, this grating is removed and the shield lid is placed on the basket. Other operations are the same as those for a PWR Basket.

Table 4.2-1 presents a summary of fabrication requirements. Figure 4.2-2 provides a description of the GTCC Basket.

4.2.4.2.3 Description of the Basket Overpack

In the unlikely event of a leak in the confinement boundary of a PWR Basket or GTCC Basket that cannot be repaired, a Basket Overpack would be used.

The Basket Overpack is a cylindrical shell with sufficient inside diameter to accommodate a PWR Basket or GTCC Basket. The Basket Overpack is designed to the same code requirements as the PWR and GTCC Basket confinement boundary.

The confinement boundary is not hydrostatically tested. This is considered acceptable based on the specified methods of construction and intended application. The bases for this conclusion are summarized as follows:

1. The calculated internal basket pressure during normal service is approximately atmospheric resulting in negligible pressure stresses,
2. The welds in question are dye penetrant tested thereby providing assurance that the weldment is free of unacceptable imperfections,
3. Neither operating nor environmental conditions are expected to subject the welds to cyclic loading, and

4. The internal atmosphere of the Basket Overpack is pressurized with helium and the structural lid welds are checked with a helium leak detector.

The Basket Overpack is designed, fabricated, and tested to provide a confinement barrier for spent nuclear fuel and GTCC waste in accordance with the general design criteria requirements of 10 CFR 72 Subpart F. Although the Basket Overpack will not be N-stamped in accordance with ASME Section III, NC-8100, basket design, fabrication and testing controls must be approved by the NRC prior to commencing ISFSI operations. Design, fabrication, and testing of Basket Overpacks will be performed in accordance with a Quality Assurance Program meeting the applicable requirements of 10 CFR 72 Subpart G as presented in Chapter 11.

The description of operations involving the Basket Overpack is presented in Chapter 5. The fabrication summary is the same as for the basket as presented in Table 4.2-1. Figure 4.2-3 provides a description of the Basket Overpack.

4.2.4.2.4 Description of the Concrete Cask

The Concrete Cask is a reinforced concrete cylinder designed to the requirements of ACI-349 and constructed to ACI-318. The concrete is Type II Portland Cement, 145 pcf, 4000 psi concrete. Outer and inner re-bar cages are formed by vertical hook bars and horizontal ring bars. The internal cavity of the Concrete Cask is formed by a steel liner and bottom plate. The steel and concrete walls of the cask are designed to minimize side surface radiation dose rates.

The thermal evaluations discussed in Section 4.2.6 demonstrate that the concrete temperature limits provided in ACI-349 may be exceeded under credible environmental conditions for the ISFSI site. ACI-349 Section A.4 establishes a normal operating temperature limit of 150°F except for local areas which may not exceed 200°F. Short term or accident temperature limits shall not exceed 350°F. Higher temperatures than those specified above may be allowed if tests are provided to evaluate the reduction in strength and this reduction is applied to design allowables.

The concrete mix used to fabricate the Concrete Casks is intended to allow satisfactory long term concrete temperatures as high as 250°F. Studies have shown that there is no reduction in strength for bulk concrete temperatures up to 250°F (Reference 11).

An air flow path is formed by the openings at the bottom (air entrance), the air inlet ducts, the gap between the basket exterior and the Concrete Cask interior, and the air outlet ducts at the top. The air inlet and outlet vents are steel-lined penetrations that take non-planar paths to minimize radiation streaming. A shield ring is provided over the basket-liner annulus to reduce the dose rate at the top of the cask.

The cask lid is fabricated from a steel plate which provides additional shielding to reduce the skyshine radiation. The cask lid also provides a cover and seal to protect the basket from the environment and postulated tornado missiles. The lid is bolted in place and is provided with a locking wire with a lead seal.

The bottom of the Concrete Cask is covered with a steel plate which minimizes loss of cask concrete during a bottom drop accident. The Concrete Cask has reinforced chamfered corners at the top and bottom to minimize damage during handling.

The cask is constructed by pouring concrete between a re-usable form and the inner metal liner. The reinforcing bars and air flow embedments are installed and tied prior to pouring.

A summary of fabrication requirements is presented in Table 4.2-2. Figure 4.2-4 provides a description of the Concrete Cask.

4.2.4.2.5 Failed Fuel Can

The Failed Fuel Can is designed to contain partial or complete fuel assemblies with failed or suspect rods. The internal square opening accommodates a fuel assembly without inserts. The outside dimensions allow the Failed Fuel Can to fit in one of the four oversized storage locations within a PWR Basket.

The shell of the Failed Fuel Can is fabricated from carbon steel. Near the bottom of each side of the shell assembly are two screened vent holes. These vent holes enable vacuum drying of the canister. The vent holes also expose the contents of the Failed Fuel Can to the helium atmosphere of the PWR Basket.

The lid is bolted in place and is designed to be lifted using a fuel handling tool. The lid bottom also has vent holes to facilitate draining.

Carbon steel components of the Failed Fuel Can are coated with radiation resistant, high temperature, hard surface inorganic zinc coating. Figure 4.2-5 provides a description of the Failed Fuel Can.

4.2.4.2.6 Fuel Debris Can

The Fuel Debris Can is designed to contain loose fuel pellets, fuel pellet fragments and fuel assembly fragments (portions of fuel rods, or other portions of a fuel assembly). The overall length of the Fuel Debris Can is less than that of a fuel assembly. The Fuel Debris Can is designed to fit in one of the four oversized storage locations within a PWR Basket. Lifting lugs are provided for handling operations.

The Fuel Debris Can consists of a square tube made from carbon steel. Internal filters are designed to retain the fuel debris. The top plate of the Fuel Debris Can has a drain and helium backfill connection. During vacuum drying a high efficiency particulate air (HEPA) filter will be connected to the vacuum drying system to prevent release of fuel particles which may not be retained within the Fuel Debris Can.

Carbon steel components on the Fuel Debris Can are coated after welding with radiation resistant, high temperature, hard surface inorganic zinc coating. The canister design is shown on Figure 4.2-6.

4.2.4.2.7 GTCC Can

The GTCC Can is designed to contain GTCC waste for placement within a GTCC Basket. Up to 29,000 lbs. contained within 28 GTCC Cans can be placed in a GTCC Basket.

The shell of the canister is fabricated from steel. Two vent holes are located near the bottom of each side of the shell assembly and on the bottom plate allowing draining and vacuum drying of the container.

The lid assembly consists of a bolted steel plate and has provisions to accommodate a lifting tool for handling operations.

Components on this canister are coated after welding with radiation resistant, high temperature, hard surface inorganic zinc coating. The GTCC Can design is shown on Figure 4.2-7.

4.2.4.3 Design Bases and Safety Assurance

The design codes for the individual storage structures and components are provided in Section 4.2.1. The storage structures and components are designed for safe long-term storage of spent nuclear fuel and GTCC waste. They are designed to survive normal, off-normal, and postulated accident conditions without an unacceptable release of radioactive material or excessive radiation exposure to workers or members of the general public. Storage systems and components are designed and fabricated in accordance with recognized codes and standards that provide ample safety margin.

Design features that have been incorporated in the ISFSI to provide safe long-term fuel storage include:

1. Leak-tight/multi-pass seal welds on each basket structural lid, shield lid, shell, and bottom plate,
2. Thick lids and walls to minimize radiation exposure to public and site personnel,
3. Design of basket body and internals to withstand a postulated drop accident during storage or transportation, and
4. Design of Concrete Cask to protect baskets from postulated environmental events.

Methods used to minimize personnel radiation exposure during ISFSI operations are discussed in Chapter 5 and Chapter 7.

Design features to maintain subcritical conditions for normal operations and credible accident scenarios are discussed in Section 4.2.7.

10 CFR 72.126(a)(3) requires access to areas of potential contamination or high radiation within an ISFSI to be controlled. During normal storage conditions no high radiation areas are expected within the ISFSI. Increased radiation levels are possible during component handling evolutions. Although not anticipated, any contamination associated with ISFSI operations should be limited to the Storage Pad. The Storage Pad is located in the protected area which is surrounded by a security fence. Access to this area is controlled by security and is discussed in Section 3.3.5.1. The Radiation Protection Program is discussed in Chapter 7.

The ISFSI is designed to provide safe storage of spent nuclear fuel and GTCC waste for 40 years in accordance with the requirements of Oregon Administrative Rule (OAR) 345-26-390(4)(j).

Major design requirements are summarized in Table 4.2-3.

4.2.5 STRUCTURAL EVALUATION

This section describes the design and analyses of the principal structural components of the storage system and components under normal operating conditions. Basket weight calculation was performed assuming a PWR Basket containing 24 intact fuel assemblies each containing a RCCA. This weight configuration is considered to conservatively bound actual loading configurations. This section describes the methodology and analysis techniques used and presents the results. The GTCC Basket is bounded by the PWR Basket because it is lighter and has lower temperatures and pressures.

The storage system structural design criteria are specified in Chapter 3. The combinations of normal, off-normal, and accident loadings have been evaluated per ANSI 57.9 for the Concrete Cask and per the ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NC for Class 2 components for the basket confinement boundary.

The following components, utilized for normal spent fuel storage operations, are addressed in this section:

1. PWR Basket containment boundary (shell, structural lid);
2. PWR Basket internal assembly cells and structural tubes;
3. PWR Basket shield lid support ring weld;

4. Concrete Cask concrete body; and

5. Concrete Cask steel components (reinforcement, liner, cover lid).

In addition, the handling devices analysis is presented in Section 4.7.

The following sections discuss individual loads and load combinations. The structural evaluations demonstrate that components meet their structural design criteria and are capable of safely storing spent nuclear fuel or GTCC.

4.2.5.1 Weights and Centers of Gravity

The component weights and centers of gravity for the storage system are summarized in Table 4.2-4 and Figure 4.2-8.

4.2.5.2 Mechanical Properties of Materials

The mechanical properties of steels and concrete used in the structural evaluation of the storage system are presented in Tables 4.2-5 and 4.2-6.

4.2.5.3 Basket Analysis Under Normal Loads

4.2.5.3.1 Basket Thermal Stress Analysis

The storage system was evaluated for thermal stresses by using separate and distinct models for the PWR Basket and Concrete Cask. This approach is valid since these components are not structurally coupled. The basket is free to thermally expand or contract relative to the Concrete Cask. In addition, the sleeves are not connected to the basket shell so that these components can also be evaluated separately.

For the overall evaluation of the thermal stresses, the temperature distribution for the -40°F ambient condition was used because it causes the highest thermal gradients in the basket structure. The temperature distribution was obtained from the thermal analysis described in Section 4.2.6.

The PWR Basket internal structure is designed to minimize restrictions of thermal expansion. Existing gaps allow independent expansion of the internal assembly cells, structural tubes, and the shell. As a result, thermal stresses in the basket remain low. The thermal stresses are calculated using the results presented for the PWR Basket during transport (Reference 1) which were based on the detailed finite element modeling of the structure. The stresses during storage were found by scaling of the transportation stresses. The results are summarized in Table 4.2-7

4.2.5.3.2 Basket Dead Weight Load Analyses

The dead weight loads are calculated by a ratio of the vertical drop stresses with respective accelerations. The acceleration for dead weight is 1 g. Dead weight stresses are presented in Table 4.2-8.

4.2.5.3.3 Basket Pressure Analysis

The maximum operating differential pressure is provided in Table 4.2-9. The same approach as the basket thermal stress analysis was used, i.e., the stresses were ratioed from the transportation condition. The Basket Overpack pressure stresses are calculated using classical shell formulas. The results are evaluated in combination with other loadings in Table 4.2-8. The allowable external pressure has been calculated per ASME Section III, NC-3133.3. The resulting allowable pressure is much higher than the calculated maximum; therefore, basket buckling will not occur.

4.2.5.3.4 Basket Handling Analysis

The basket normal handling load has been defined as $\pm 0.5g$ applied in all directions simultaneously. This produces the sum of $(0.5g)\sqrt{2} = 0.71g$ in the horizontal direction and 0.5g in the vertical direction. The stresses are calculated by the appropriate scaling of stresses due to a drop accident. The analysis is presented in Section 8.2 and added to the stresses due to the other design loadings in Table 4.2-8.

4.2.5.3.5 Basket Load Combination

The basket design loadings are based on dead weight, thermal, internal pressure (not applicable to basket internals), and handling loads. The stresses due to the loadings are presented and evaluated in Table 4.2-8. The reduction factor of 0.75 has been applied to allowable stresses for the partial penetration welds (ASME Section III, NC-3264.6). It can be seen that all stresses are within allowable limits.

4.2.5.3.6 Basket Fatigue Evaluation

Fatigue effects on the basket are addressed using the criteria contained in ASME Section III, NC-3219.2. Fatigue analysis need not be performed provided the criteria of Condition A are met. A summary of the criteria and their application to the basket are presented in the following paragraphs.

According to ASME Section III, NC-3219.2, fatigue analysis is not mandatory for materials having tensile strength not exceeding 80 ksi (provided for the basket components) and the expected number of cycles (a) + (b) +(c) + (d) is less than 1,000.

(a) Full Range Pressure Cycles

The normal operating pressure for the basket is 14.7 psia (0 psig). The full range pressure cycles are due to: vacuum drying, two pressure tests, postulated failure of all fuel rods and significant ambient temperature changes (conservatively assumed to occur 10 times per year during 40 years of the cask lifetime).

Therefore, the total number of fluctuations of this type is (a) = 1 + 2 + 1 + (10)(40) = 404.

(b) Expected Number of Pressure Cycles

The expected number of pressure cycles of this type is 0 since fluctuations in weather conditions need not be considered here.

(c) Effective Number of Changes in Metal Temperature Between Adjacent Points

The distance between adjacent points as defined by ASME Section III, NC3219.2 is $2\sqrt{Rt}$. It can be seen from the thermal analysis results that although the temperature of the basket changes significantly due to weather conditions, the change in the temperature difference between two adjacent points never exceeds 50°F. Therefore, the effective number of cycles of this type is 0.

(d) Only for Vessels With Welds Between Materials With Different Coefficients of Expansion

The basket design employs welding of the carbon steel tubes (corner frames) to the stainless steel PWR Basket shell. A thermal cycle occurs if $(\alpha_1 - \alpha_2)\Delta T$ is ≥ 0.00034:

Where:

α_1 -coefficient of thermal expansion for shell material (9.00×10^{-6}in/in/°F)

α_2 -coefficient of thermal expansion for tube material (6.26×10^{-6}in/in/°F)

ΔT -shell temperature difference (°F)

Substituting values and solving for ΔT yields:

$$\Delta T < 124°F$$

Since shell temperature is related to ambient air temperature, it would take approximately a 124°F air temperature change to result in one cycle. The maximum temperature for the region is 107°F with a minimum of -20°F. This represents a maximum temperature change of 127°F. If this extreme temperature variation is assumed to occur once per year for the design life of 40 years, this would result in 40 type (d) cycles.

The discussion presented in the preceding paragraphs shows that (a) + (b) + (c) + (d) = 454 and less than 1000. Thus, all criteria of Condition A are met and the basket is exempt from the fatigue analysis.

4.2.5.3.7 Basket Pressure Test

The basket is hydrostatically tested to meet the requirements of ASME Section III, NC-6221, 6222. The stresses due to a test pressure of approximately 22 psia are acceptable based on results of the critical pressure analysis (Section 8.2) and evaluated per ASME Section III, NC-3217. For pressurized conditions, maximum stresses occur in the weld between the basket shell and the bottom plate. These stresses calculated at a pressure of 10 psig are:

P_m	= 0.2 ksi	(ASME Service Level A Limit 15.5 ksi)
$P_L + P_b$	= 4.6 ksi	(ASME Service Level A Limit 23.2 ksi)

Where:

P_m = general primary membrane stress intensity

P_L = local primary membrane stress intensity

P_b = primary bending stress intensity

Table 4.2-8 summarizes the results of maximum stress evaluations for the basket.

Since maximum stresses are less than allowable, the test pressure meets requirements of the ASME Code, Section III, Subsection NC.

4.2.5.3.8 Basket Fracture Toughness

The basket confinement boundary materials are made of austenitic stainless steel and are exempt from impact testing per ASME, Section III, NC-2311.

4.2.5.3.9 Basket Overpack Analysis

The basket overpack is designed for the same normal operation loads as the basket except for handling (a loaded Basket Overpack is not required to be handled).

The normal condition loads for basket overpack are dead weight, pressure, and thermal. Handling is not applicable because the loaded overpack is never lifted out of the cask. The dead weight and normal pressure stresses are negligible. The overpack thermal stresses are conservatively assumed to be the same as those for the basket because although the temperature gradients are similar, the member thicknesses are smaller making the overpack more flexible.

4.2.5.4 Concrete Cask Analysis under Normal Operating Loads

Three load components act on the Concrete Cask during normal operation: dead load, live load and thermal load due to differential thermal expansion. These components are analyzed below. The results of combining the loads and comparing the Concrete Cask stress levels to allowable limits are summarized in Table 4.2-10. As shown in this table, the Concrete Cask meets the structural requirements of ANSI 57.9 and ACI-349[1].

4.2.5.4.1 Concrete Cask Dead Load

The stress due to the dead load (f_D) on the Concrete Cask bottom is conservatively calculated by assuming the total weight of the fully loaded Concrete Cask (300,000 lbs which includes the

[1] Refer to Section 4.2.4.2.4 for justification for deviation from ACI-349 temperature limits.

weight of a Basket Overpack) is taken by the concrete bottom only over a 12 inch wide area of the bottom plate. The stress is calculated to be 200 psi.

4.2.5.4.2 Concrete Cask Live Load

The Concrete Cask is subject to two live loads: the snow and ice load and the weight of a Transfer Cask and fully loaded PWR Basket. Both of these live loads act on the top of the Concrete Cask. The snow load is uniformly distributed over the top of the cask and represents a negligible contribution to Concrete Cask stress levels.

To calculate the stress due to the loaded transfer cask (f_L), the following equation is used:

$$f_L = \text{Weight of loaded transfer cask / Area of Contact} \approx 320 \text{ psi.}$$

4.2.5.4.3 Concrete Cask Thermal Stresses

The Concrete Cask thermal stress is calculated based on the temperature gradient across different components. The Concrete Cask wall analysis is based on the standard approach to concrete which assumes that it resists only compression with steel reinforcement resisting tension. Stresses are calculated by balancing tension and compression in the section because thermal loading can not produce any resultant force.

The maximum thermal stresses for each of the Concrete Cask structural components are listed in Table 4.2-11. The acceptability of these thermal stress levels is included in the Concrete Cask load combination evaluated in Table 4.2-10.

4.2.5.4.4 Concrete Cask Load Combination

The evaluation of Concrete Cask load combinations in accordance with ACI 349 and ANSI 57.9 is presented in Table 4.2-10. Load combinations 5, 6, and 8 include results of the accident analysis discussed in Chapter 8. For load combination 8, the thermal loads in the critical sections are zero due to the self-balancing nature of the thermal stresses across the entire cask section which resists the tornado missile impact. For load combination 6, the thermal stresses from Table 4.2-11 are recalculated into a moment using the standard technique for concrete analysis.

4.2.6 THERMAL EVALUATION

This section presents the thermal analysis of the storage system for normal operation. The significant thermal design feature of the storage system is the air flow path used to remove the maximum of 26 kW of decay heat. This natural circulation of air inside the Concrete Cask allows the concrete temperatures to be maintained below the design limits and keeps the long term fuel cladding temperatures below limits where degradation might occur.

The base calculation was performed assuming 75°F ambient conditions to model the average long term temperatures expected over the life of the cask. No solar load was used because the amount of time required for the massive cask to heat up noticeably is substantially longer than the daylight time. Even if solar loads are assumed to affect the cask for 12 to 14 day-light hours they will only affect the outer concrete temperatures for the period the sun is shining. The remaining 10 to 12 hours in which solar load is not present allows the outer concrete to return to the temperature that would have established without solar load. For normal ambient conditions, the basket and concrete temperature are not affected by solar loads. Cask tests (Reference 2) demonstrate that little or no impact on fuel temperature is experienced as a result of solar loads.

To bound the expected temperature ranges in which the storage system might operate, two off-normal severe environmental temperature conditions were evaluated. These calculations are presented in Section 8.1.2. The cases considered are -40°F with no solar loads and 100°F with maximum solar loads. The maximum solar load was calculated to be the 24-hour average solar load to model the steady state temperature expected from long term (four to five days) exposure to 100°F air.

The 75°F ambient conditions are utilized to determine long term storage temperatures and -40°F and 100°F ambient temperatures are used to model extreme environmental conditions. In addition to these three cases, three off-normal and accident conditions are analyzed. The first off-normal case considers a 125°F ambient condition with maximum solar loads and a maximum decay heat generation. The next off-normal condition considers blockage of the air inlets on one side of the cask (one-half of the inlets). These two cases are addressed in Section 8.1.2. The final analysis, an accident condition, considers the complete blockage of all air inlets. This analysis is addressed in Section 8.2.7.

Table 4.2-12 summarizes the results of the thermal calculations.

4.2.6.1 Summary of Thermal Properties of Materials

The thermal properties used in the thermal hydraulic analyses are shown in Table 4.2-13. Temperature invariant properties were used for the concrete, steel, fuel, and helium. Low values derived from the open literature and conservative calculations were used. If temperature-dependent properties (especially for concrete and fuel) were used, the maximum temperatures reported in this section would be slightly reduced.

Temperature limits were established for the materials used in the storage system. Specifically, these limits are for concrete, steel, fuel cladding, and coatings. The limits were established in accordance with the following:

Source	Component
PNL-6364 Report and SNC analysis	Fuel
ASME Section III (1992)	Steel
ACI 349[1]	Concrete
Manufacturers Recommendations	Coatings

Based upon evaluation of these limits it was determined that the fuel cladding and concrete temperature limits were the limiting conditions (steel temperatures well above 1000°F and coating temperatures up to 750°F being acceptable).

[1] Refer to Section 4.2.4.2.4 for discussion of deviation from ACI-349 generic limits.

Table 4.2-12 presents more details on the long-term and short-term temperature limits for the concrete. While the concrete limit is based on ACI-349 Appendix A, the fuel cladding temperature limit is actually a complex function of temperature versus time, and internal rod pressurization (Reference 2). The limit is established to keep the probability of cladding breach less than 0.5% per fuel rod over a 40 year storage term. Using the methodology presented in Reference 2, the fuel cladding allowable temperature limit was determined to be 388°C (730°F) for a Westinghouse 17 x 17 fuel assembly and a minimum cooling time of 5 years. The 388°C (730°F) limit was determined to bound the B & W 17 x 17 fuel assemblies which will also be stored in the ISFSI. A short term temperature limit of 570°C (1058°F) is established for off-normal and accident limits.

PWR Basket drying operations could possibly expose the spent fuel cladding to stresses in excess of those established for the 40 year dry storage period. The vacuum drying time will be administratively controlled to minimize the strain that fuel cladding will be subjected to during this operation.

4.2.6.2 Thermal Models for Normal Storage Conditions

Three basic models were utilized for the thermal evaluation of the storage system. These are:

1. Air flow rate and temperature;

2. Concrete Cask body and basket exterior heat transfer; and

3. Basket interior heat transfer;

These models are summarized in this section.

4.2.6.3 Air Flow and Temperature Calculation

The air flow up the annulus formed by the basket and Concrete Cask is calculated by determining the sum of the flow pressure losses ($\Sigma\ k/A^2$) due to all entrances, bends, straight sections, expansions, contractions, and exits and equating the resulting friction and form flow pressure losses to the pressure differential caused by the heating of the air (i.e., the stack or furnace effect).

The basic procedure for this calculation is to apply the macroscopic energy equation from the midpoint of the air inlet to the midpoint of the air outlet. This equation is:

$$-\Delta P+\frac{g\rho_0 h}{g_c}-\frac{\rho g\beta h(\Delta T)}{g_c}+\frac{\dot{m}^2}{2g_c\rho}\sum_i \frac{k^i}{A_i^2}=0 \qquad (4.1)$$

where:

ΔP	-	Pressure differential, inlet to outlet vents
$g\rho_0 h/g_c$	-	Elevation pressure head where ρ_o is the ambient air density, h is the height, g the acceleration of gravity and g_c the gravitational conversion factor
$\frac{\rho g\beta h(\Delta T)}{g_c}$		Pressure change due to air heating where ΔT is the air temperature difference between inlet and outlet air (°F), β is the compressibility factor, and ρ is the air density.
$\frac{\dot{m}^2}{2g_c\rho}\sum_i \frac{k_i}{A_i^2}$	-	Pressure loss due to friction and from loss of all flow segments I where m is the mass flow rate, A_i the flow area of the I-th segment and k_i the loss coefficient of the I-th segment.

Since the pressure difference between the inlet and outlet is equal to the elevation pressure head of the ambient air column, the first two terms in the above equation cancel out. Also, for the region of interest, β (the compressibility factor) can be approximated by 1/T (°R). Hence the equation reduces to:

$$\frac{\bar{\rho} g h \Delta T}{g_2 \bar{T}} = \frac{\dot{m}^2}{2 g_c \bar{\rho}} \sum \frac{k_i}{A_i^2} \tag{4.2}$$

where the bar over the density and temperature terms denotes average values.

Using the above equation and the formula for steady state heat balance, i.e.,

$$\Delta T = \frac{Q_T}{(\dot{m} c_p)}$$

Where:

Q_T - Total heat transfer to air

C_p - Specific heat of air

An iterative solution was derived for calculating the exit temperature and air mass flow rate. A spreadsheet program was used to perform this calculation. The axial heat source distribution is in direct proportion to the relative fuel burnup shown on Figure 7.2-1 and was used to calculate air temperature as a function of elevation as it flows through the Concrete Cask. Table 4.2-14 summarizes the results for the various ambient conditions. These results were then used in ANSYS finite element models for calculation of the Concrete Cask temperature distributions.

4.2.6.4 Concrete Cask Body and PWR Basket Exterior Thermal Model

Heat is generated in the fuel that is located in the PWR Basket. This heat is conducted through the PWR Basket and then convected to the air (that is maintained at a temperature established by the air flow calculation) and radiated to the Concrete Cask internal liner. Heat on the Concrete Cask liner is also convected to the air and a small amount is conducted through the concrete. The concrete surface dissipates heat by convecting and radiating into the ambient temperature. On a sunny day, additional heat enters the Concrete Cask through the exterior surface as solar insolation.

Radiation heat from all surfaces is addressed by:

$$q = \sigma \epsilon F A \left(T_1^4 - T_2^4\right) \qquad (4.3)$$

where:

q = Heat flow rate, BTU/hr

σ = Stefan-Boltzman constant, 1.714×10^{-9} BTU/hr-ft^2-°R^4

ϵ = emissivity

F = Radiative geometry view (form) factor

A = Radiating surface area, ft^2

T = Absolute source (1) and target (2) temperatures, °R

Surface emissivities vary with the radiating material and are provided in Table 4.2-13. View factors vary with surface and target geometry.

Convection from all surfaces is addressed by:

$$q \quad = hA(T_1 - T_2) \qquad (4.4)$$

where:

h = Natural convection heat transfer coefficient, BTU/hr-ft^2-°F

Heat conduction is expressed by the following differential equation:

$$\rho c_p \,(dT/dt) = d\,(k\; dT/dx)/dx + q''' \qquad (4.5)$$

where:

k = Thermal conductivity, BTU/hr-ft-°F

ρ = Density, lbm/ft^3

c_p = Specific heat, BTU/lbm-°F

q''' = Heat generation rate, BTU/hr-ft^3

These heat transfer modes are addressed by the ANSYS/THERMAL computer program. The model is presented in Figure 4.2-9. All units used in the calculations and in the programs are consistent: BTU, ft, hr, °F, °R, lbm. Two element types were used, the 3-D solid element (STIF 70) and a radiation link element (STIF 31). Thermal properties are specific to the materials (see Table 4.2-13 for thermal properties).

4.2.6.4.1 Concrete Cask Modeling Techniques and Assumptions

The model uses a 10° slice to model the entire Concrete Cask. The Concrete Cask geometry and temperatures are uniform with angular direction so that the two dimensional portrait is adequate. The 10° slice is small so as to minimize the complexity of the nodalization while still accurately representing the radial volume distribution.

The air vents were not included in the model. Because of the low thermal conductivity of concrete, the air vents affect only a local region. The model included low incoming and high exiting air temperatures below and above the heated region, respectively, to assure that temperature extremes are represented. The air in the heated region was modeled as a heat sink at the temperatures shown in Table 4.2-14.

The solar radiation heat input, used for the 100°F case, is per the requirements of 10 CFR 71.71(c). The solar load for the top surface was 2950 BTU/ft^2 per day while 1475 BTU/ft^2 was used for the curved side surfaces. This is extremely conservative because the vertical Concrete Cask sides will never have this heat load on both sides simultaneously and much of the cask side will be shaded by the adjacent casks. These thermal loads were converted to average rates by assuming the sun shines for 12 of the 24 hours per day. The resulting heat fluxes on the top and side surfaces are 123 and 61 BTU/hr-ft^2, respectively. Solar insolation was treated in the model as a volumetric heat generation. Although actual solar insolation appears as a uniformly distributed heat flux on the Concrete Cask surface, the ANSYS/THERMAL program allows heat fluxes only at nodes and that causes hot spot nodes on the Concrete Cask surface. The use of a heated region as a thin (0.25 in) shell assures uniform application of the solar flux in the elements of the thermal model. The generation rate was calculated as the above heat rate divided by the volume of the thin shell of concrete.

The basket portion of the Concrete Cask model treats only the basket shell in detail. The interior was simply modeled as a heat generating region with an effective thermal conductivity. This effective thermal conductivity (2.4 BTU/hr-ft-°F) was estimated from the cask test data (References 3, 4, 5, 6, and 7) only to accommodate the solution methods employed by ANSYS. The only interest in the basket in this model is the surface heat flux and the basket shell temperatures. The details of the basket interior were evaluated in a separate model.

4.2.6.4.2 Radiation

Radiation is included at surfaces radiating to the atmosphere and between the annular air spaces in the Concrete Cask. View factors of unity were assumed for the inner surfaces.

The view factor for the Concrete Cask exterior is calculated as 0.14 between the side of the cask and its surroundings (i.e., cask array on 15 ft centers). Casks and the ground are considered at equilibrium. The side view factor is calculated based on an average distance (11.87 ft) to the other cask (averaged on five different side locations). This is conservative as the solar insolation is assumed to load the entire vertical surface of the cask while only 14% of the surface is assumed to re-radiate heat. Indeed, if the solar load is actually present on the entire cask surface (i.e., a single cask), then the entire cask's surface could radiate back to the atmosphere. In reality, moving the casks closer together will decrease the solar load on the cask because of shading of adjacent casks. The top of the Concrete Cask was assumed to have a view factor of unity with respect to the sky.

4.2.6.4.3 Convection

Natural convection heat transfer coefficients were taken as 2.0 BTU/hr-ft^2-°F on surfaces as discussed in Section 4.2.6.4. This is a conservative value compared to full-scale experimental data from other casks.

4.2.6.5 <u>PWR Basket Thermal Hydraulics</u>

The PWR Basket analysis is conservatively based on the two-dimensional model of the hottest cross-section. Heat is generated in the fuel assemblies and transferred to the surrounding inert atmosphere and the basket sleeves by free convection and radiation. In turn, the heat is conducted through the storage sleeves towards the exterior of the sleeve assembly. It then conducts, convects, and radiates through the cover gas to the basket shell wall. Convection inside the basket is natural convection.

4.2.6.5.1 Fuel heat generation

Fuel heat generation rates were calculated by assuming 1.08 KW_t of heat generation per assembly (26 KW_t/ 24 assemblies). The hottest horizontal slice of the fuel region was converted to a volumetric heat generation as follows:

$$q''' = (P) \times (Q/\underline{v}) \quad (4.6)$$
$$= 591 \text{ (BTU/hr-ft}^3\text{)}$$

where:

P = Maximum axial peaking factor = 1.1

Q = Heat/assembly = 3,697 Btu/hr

$\underline{v}$ = Storage Sleeve assembly volume = 6.88 ft^3

4.2.6.5.2 Radiation

Radiation is included at all surfaces radiating to the basket shell. For simplicity, shape factors were assumed to be 1.0 with direct radiation to the basket shell. Internal fuel assembly radiation is included in the effective fuel conductivity calculation.

4.2.6.5.3 Fuel Equivalent Conductivity

Heat transfer in the fuel region is a complex combination of conduction, convection, and radiation. This heat transfer was modeled as conduction only by assuming the fuel region to be a solid with an equivalent fuel conductivity k_f.

4.2.6.5.4 Helium Equivalent Conductivity

Heat transfer in the helium region is a combination of conduction and free convection throughout the basket. For the wide flow regions shown in the basket model, the conduction coefficient of 2.8 BTU/hr-°F-ft was derived in Reference 8 based on experimental results for other casks. In the narrow areas, no credit is taken for convection and a helium conductivity value of 0.11 BTU/hr-°F-ft is used.

Heat transfer modes are addressed by the ANSYS/THERMAL computer code. The geometry of the basket interior was converted to the finite element model shown in Figure 4.2-10. The model represents a horizontal slice of unit thickness through the hottest section. The symmetrical nature of the storage system allowed use of a 45° sector model of the basket. This symmetry was utilized by imposing zero heat flux boundary conditions along the 0° and 45° model boundaries. Two element types were used, the two dimensional solid element and a radiation link element. The highest shell temperature calculated from the Concrete Cask model is used as a boundary condition. Thermal properties are specific to the materials and are presented in Table 4.2-13. The basket temperature distribution at 100°F ambient air conditions is shown in Figure 4.2-11.

4.2.6.6 Maximum Temperatures

Temperature distributions for the normal, off-normal, severe environmental, and the accident conditions are shown in Table 4.2-12. It can be seen that the components of the storage system are below their corresponding limits.

4.2.6.7 Minimum Temperatures

The possibility of brittle fracture was considered for minimum temperatures. As stated in Section 4.2.5.3.8, brittle fracture is not of concern for the basket.

4.2.6.8 Maximum Internal Pressure

The basket is backfilled with helium so that at the conditions present during normal operations the internal pressure is at approximately atmospheric pressure. The pressure calculated for different ambient conditions is presented in Table 4.2-9 and stresses are included in the structural analysis in Section 4.2.5. The worst case internal pressure occurs during a postulated accident where fuel rods inside the basket are breached and release their fission gases. This case and the resulting pressure and stresses are described in Section 8.2.6.

4.2.6.9 Evaluation of Cask Lifetime Performance under Normal Conditions of Storage

As shown in the preceding sections, the storage system operates within the thermal design limits. Therefore, no degradation due to temperature effects on materials or components is expected during the lifetime of the cask.

4.2.7 CRITICALITY EVALUATION

The criticality evaluation was performed using the KENO-Va module of the SCALE-4.1 code package (Reference 9).The model analysis was based on Westinghouse 17x17 standard fuel. The ISFSI storage system will also contain B&W 17x17 fuel which is considered to be bounded by the Westinghouse fuel. The only significant difference in these two types of fuel assemblies is the B&W assembly has a slightly smaller fuel pellet diameter. The smaller pellet size makes the B&W assembly slightly less reactive and should therefore be bounded by Westinghouse analysis.

The four corner cells of a PWR Basket may contain a Fuel Debris Can. A fuel mass limit of 10 kg per PWR Basket will be administratively controlled. A limit of 10 kg of fuel debris is significantly less than the fuel mass of an intact fuel assembly (~460 kg uranium). The 10 kg of fuel mass will not be nearly as reactive as an intact fuel assembly no matter how the fuel debris is arranged within the Fuel Debris Can. The 10 kg of fuel mass will not cause thermal, structural or shielding problems no matter how it is distributed within the Fuel Debris Can.

The parameters of concern for criticality evaluations are initial enrichment, burnup, moderation, poisons, and geometry. These parameters combined produce the reactivity of the system which is measured as K_{eff}. Neutron poison plates are included in the basket design to meet the transportation requirements of 10 CFR Part 71. Although neutron poison plates are included in the design of the basket, no credit was taken for these plates in the criticality analysis.

The analysis relies on basket geometry, and conservatively assumes an initial fuel enrichment of 4.2 wt% U^{235} with no credit taken for burnup. Fuel enrichment assumptions conservatively bound the fuel to be stored which has a maximum initial enrichment of 3.56 wt% U^{235} and burnup.

The PWR Basket atmosphere was assumed to be inerted with helium. Water moderation was not considered since it would require significant basket in-leakage coincident with an incredible flood (the ISFSI is located above the credible flood plane).

The calculated K_{eff} is 0.3880, with a one sigma statistical error of 0.0010. The addition of code bias and uncertainty effects results in calculated K_{eff} of 0.4073 for the dry storage condition.

4.3 AUXILIARY SYSTEMS

The storage system is self-contained and uses a passive design that does not require auxiliary process or cooling systems for operation. It is designed for safe interim storage of spent nuclear fuel, failed fuel, fuel debris, and GTCC waste by transferring heat to the ambient air.

4.3.1 VENTILATION AND OFF-GAS SYSTEMS

The spent fuel and other radioactive materials are confined within the basket which is stored within the Concrete Cask. There are no radioactive releases during normal and off-normal operations. In the unlikely event a leaky basket must be placed in a Basket Overpack, evacuation of the Basket Overpack and backfilling with helium would be required. This evolution is discussed in Chapter 5. A suitable filtration system would be required for the vacuum system vent path during this evolution. Initial basket loading and vacuum drying is performed in the Trojan Nuclear Plant Fuel Building under the controls of the 10 CFR 50 license.

4.3.2 ELECTRICAL SYSTEMS

The ISFSI design relies on the natural circulation of air to provide cooling of the spent nuclear fuel. Heat generated by the spent nuclear fuel is transferred to the air located in the Concrete Cask annulus. This heated air rises and exits via air outlets (4) located near the top of the Concrete Cask. Ambient air enters via air inlets (4) located at the bottom of the Concrete Cask. This passive design eliminates the need for utilities to support storage conditions. Electrical power is provided for security requirements which are addressed in the ISFSI Physical Security Plan. Electrical power is also available to support instrumentation such as temperature monitoring.

4.3.3 AIR SUPPLY SYSTEMS

An air supply is not required at the ISFSI during storage. The air pad system is anticipated to be the only requirement for compressed air usage at the ISFSI site. A permanent compressed air supply is not provided for the ISFSI since usage is anticipated to occur only during initial loading and final offsite transport of the baskets. Portable air compressors can be utilized when required.

4.3.4 STEAM SUPPLY AND DISTRIBUTION SYSTEM

A steam supply is not required for storage system operations.

4.3.5 WATER SUPPLY SYSTEM

A water supply is not required for the normal operation of the storage system.

During the loading of spent fuel into the basket, clean borated water or filtered fuel pool water is pumped into the basket-to-transfer cask gap. This water requirement will be met by the existing Trojan plant systems.

4.3.6 SEWAGE TREATMENT SYSTEM

There are no sewage treatment systems required for ISFSI operation. The ISFSI is a passive, at grade, system which does not require fluid systems for operation. Site drainage is accommodated by the existing TNP drainage system. Sanitary facilities for ISFSI staff are available in the existing Trojan Central Building.

4.3.7 COMMUNICATION AND ALARM SYSTEMS

A commercial telephone system is provided for communications with the corporate office and local and federal agencies. Portable radios are available for use by the operations staff and security staff as required. The existing communications system is also utilized for emergency plan notifications and training.

4.3.8 FIRE PROTECTION SYSTEM

The ISFSI location, along with the cask layout and use of noncombustible and heat resistant material, make the storage system design highly resistant to the effects of fire. Only small electrical or vehicle fires are considered credible in the ISFSI. Portable fire extinguishers are available in the unlikely event of a small fire. The transient analysis is presented in Chapter 8.0.

4.3.9 MAINTENANCE SYSTEMS

Prior to the transfer of fuel from the spent fuel pool to the Concrete Cask, the cask is inspected for damage. Once in storage, the storage system is designed to be passive and require no maintenance. Vent outlet temperature monitoring and inspections of air inlets and outlets for blockage are performed (per Technical Specifications) to assure proper operation of the storage system. Additionally, the Concrete Cask exterior is inspected annually to identify any surface defects.

4.3.10 COLD CHEMICAL SYSTEMS

There are no cold chemical systems required for the storage system.

4.3.11 AIR SAMPLING SYSTEMS

Spent fuel is stored within the sealed and inerted basket. The basket is designed to maintain a confinement boundary during all operating conditions. Since there are no operations or credible accident scenarios which are expected to result in a release of radioactive material, permanently installed air sampling equipment is not required. Portable air monitoring equipment can be utilized as conditions warrant.

4.4 DECONTAMINATION SYSTEMS

4.4.1 EQUIPMENT DECONTAMINATION

Decontamination equipment is not required at the ISFSI. Decontamination activities are performed in the Fuel Building prior to transferring the Concrete Cask to the ISFSI pad. This activity removes contamination from the outside surfaces of the transfer cask, lifting yoke, and upper end of the basket caused from immersion in the spent fuel pool.

Section 5.1.1.2 describes the procedures implemented to minimize the contamination of the baskets and transfer cask during loading operations. This section also discusses the decontamination of the transfer cask as well as the exterior surface of the basket. Surveys are performed on these components and decontamination, if required, would be performed prior to transfer to the ISFSI.

4.4.2 PERSONNEL DECONTAMINATION

Since the basket is decontaminated prior to transfer to the pad and the radioactive material is sealed within the basket, personnel decontamination facilities are not required during storage.

4.5 SHIPPING CASK REPAIR AND MAINTENANCE

There is no shipping cask repair or maintenance facility at the Trojan ISFSI site. Repair and maintenance facilities will be provided by the shipping cask 10 CFR 71 certificate holder.

4.6 CATHODIC PROTECTION

The Trojan ISFSI is a dry, above ground system so that cathodic protection in the form of impressed current is not required. The normal operating temperatures are well above ambient air temperatures so that there is no opportunity for condensation on any surfaces.

Several measures are taken to provide corrosion protection for the basket. The basket shell is constructed of corrosion resistant material. To avoid contact of dissimilar materials, the bottom of the basket is separated from the steel bottom plate of the Concrete Cask liner by ceramic tiles. The basket internals are coated with inorganic zinc which prevents corrosion. After the basket is sealed, dried, and backfilled with helium, the basket is maintained in an inert environment to further protect from corrosion. Finally, the basket is protected from the environment by the surrounding Concrete Cask and cask lid.

4.7 SPENT FUEL AND HIGH-LEVEL RADIOACTIVE WASTE HANDLING OPERATION SYSTEMS

This section addresses ISFSI components utilized in moving the PWR Baskets, GTCC Baskets, and Concrete Casks into storage and eventually off-site. Loading operations for spent nuclear fuel and GTCC waste are performed in the Fuel Building. The Fuel Building systems are operated under the plant 10 CFR 50 license. Evaluation of the Fuel Building structure, components and systems is within the scope of the Trojan Nuclear Plant SAR.

The fuel handling components that are considered to be a part of the Trojan ISFSI are:

Transfer Cask

Transfer Station

Air Pad System

Lifting Yoke

Basket Hoist Rings

4.7.1 STRUCTURAL SPECIFICATIONS

The Transfer Cask and Transfer Station are classified as important to safety and are relied upon to safely handle the PWR or GTCC baskets. Quality Assurance requirements are outlined in Chapter 11.

The remaining systems are not classified important to safety and are purchased as commercial grade items.

The design and fabrication codes for the components are as follows:

Component	Governing Code/Standard[1]
Transfer Cask	NUREG 0612 / ANSI N14.6
Transfer Station	AISC Manual of Steel Construction
Air Pad System	Commercial grade
Lifting Yoke	NUREG 0612 / ANSI N14.6
Basket Hoist Rings	NUREG 0612 / ANSI N14.6

[1] Applicable revision of governing code/standard is provided in Chapter 3

4.7.2 INSTALLATION LAYOUT

4.7.2.1 Building Plans and Sections

Loading of the baskets and Concrete Casks will be performed within the Fuel Building under the 10CFR50 license. ISFSI handling operations which are discussed in Chapter 5, are anticipated to be limited to transferring baskets from the Concrete Casks to a shipping cask for off-site storage or disposal. These operations will take place on the ISFSI Storage Pad utilizing the Transfer Station. Neither the ISFSI Storage Pad nor the Transfer Station is located within a building or structure. The design of the ISFSI Storage Pad is discussed in Section 4.2.2.1. The design of the Transfer Station is discussed in Section 4.7.3.2.

4.7.2.2 Confinement features

Confinement features relied upon during handling operations are the same as those during storage. Confinement features are discussed in Section 4.2.3.

4.7.3 INDIVIDUAL UNIT DESCRIPTION

4.7.3.1 Transfer Cask Description

The transfer cask is a special lifting device designed and fabricated to the requirements of NUREG 0612 and ANSI N14.6. The transfer cask is also designed as a shielding bell to reduce the dose to Trojan plant personnel in accordance with ALARA principles.

The Transfer Cask consists of a cylinder with moveable shield doors at the lower end and a top cover. The cylindrical wall of the Transfer Cask consists of various material layers. The inner and outer surface layers are made of steel. Sandwiched between the steel surface layers is a thickness of lead (inside) and neutron absorbing material (outside). The movable shield doors at the lower end allow lowering of the basket into the concrete cask. The doors slide in steel guides along each side of the Transfer Cask. Two steel pins per door are used to prevent inadvertent opening of the doors. Hydraulic pistons are used to open the doors for the basket transfer. The top cover of the cask extends over the basket to prevent the basket from being inadvertently lifted out of the top of the Transfer Cask.

The Transfer Cask is lifted from above by the Lifting Yoke via two trunnions located on the outer shell approximately three feet from the top of the Transfer Cask. The trunnions are solid steel and extend radially from the cask body. Each trunnion is welded to the inner and outer steel shells of the Transfer Cask wall with full penetration circumferential welds. The two trunnions are capable of accommodating the combined weight of the transfer cask and a fully loaded wet basket while meeting the requirements of NUREG 0612. The transfer cask and trunnions are fabricated in accordance with ANSI N14.6 requirements and are tested to 300% of their maximum design load.

Figure 4.7-1 provides a description of the Transfer Cask. Figure 4.7-2 provides a description of the trunnions.

4.7.3.2 Transfer Station

Since the Trojan Nuclear Plant spent fuel pool may not be available for basket transfer from the Concrete Cask to the shipping cask at the time DOE or another repository facility is available, the ISFSI is designed as a stand alone facility. The ISFSI is equipped with a permanent Transfer Station to support dry transfer operations.

The Transfer Station is important to safety and designed for a 0.5g Transfer Cask handling load applied in any direction. The structural steel Transfer Station allows a Concrete Cask and shipping cask (or another Concrete Cask) to be positioned adjacent to each other. The empty Transfer Cask is then placed on top of the Concrete Cask. A special sliding collar inside the station is clamped around the Transfer Cask approximately at the height of its center of gravity and locked in place to stabilize the Transfer Cask during handling operations. Transfer operations are discussed in Sections 5.1.1.5 and 5.1.1.6. Transfer Cask lifting above the Concrete Cask or shipping cask is administratively limited to less than 6 inches. The use of the Transfer Station restricts the potential handling accidents to those analyzed in Section 8.2.13.3.

A summary of the transfer station fabrication specifications is provided in Table 4.7-1. Figure 4.7-3 provides a description of the Transfer Station.

4.7.3.3 Air Pad System

A commercially available air pad system will be utilized for moving the Concrete Casks on the Storage Pad. The air pad system consists of four individual air pads approximately 48 inches square. In order to insert the air pads under the Concrete Cask, the inlet air screens must be removed. The air pads are positioned under the Concrete Cask in the air inlet channel area and pressurized. The effective lift height of the air pads is approximately 3 inches. The Concrete Cask can then be moved to the desired location where the air pads are depressurized and removed. The air inlet screens can then be reinstalled.

4.7.3.4 Lifting Yoke

The Lifting Yoke is designed and fabricated to mate with the Transfer Cask trunnions and provide a means to lift the loaded Transfer Cask. The Lifting Yoke is tested to 150 % of its maximum design load. Figure 4.7-4 provides a drawing of the Lifting Yoke.

4.7.3.5 Hoist Rings

The Hoist Rings are commercially available and are inserted into the threaded connections provided in the PWR Basket and GTCC Basket structural lids. Section 4.7.4.4 provides an analysis of the Hoist Rings to demonstrate they meet the requirements of NUREG-0612 and ANSI 14.6.

4.7.3.6 Mobile Cranes

The ISFSI design does not include a permanently installed crane, thereby requiring the use of a mobile crane for handling operations. The use of mobile cranes at nuclear power plants is governed in part by ANSI/ASME N45.2.15, with technical requirements specified in ANSI B30.5 (1994). Prior to handling spent fuel casks, procedures for load handling, inspection, safe loads analysis and load tests in accordance with ANSI/ASME N45.2.15 will be in place. Evaluation of the mobile crane as a potential drop or fall hazard will be performed prior to the time of transfer.

4.7.4 STRUCTURAL ANALYSIS OF THE FUEL HANDLING COMPONENTS

4.7.4.1 Transfer Cask Lift

A loaded Transfer Cask weight of 215,000 lbs is used for the lifting device analysis. This is higher than the maximum weight in Table 4.2-4 and, therefore, conservative.

4.7.4.1.1 Trunnions

The adequacy of the transfer cask trunnion design can be evaluated by considering the stress levels in the trunnion and the transfer cask wall. The transfer cask lifting trunnion was designed with a factor of safety of 5 or greater on ultimate and 3 or greater on yield and includes the dynamic load increase factor of 10%.

The shear stress (τ) of the trunnion is:

$$\tau = \frac{\text{(Weight of basket + transfer cask)} \times 1.1 \times 4/3}{(2) \times A_T}$$

$$= 1.4 \text{ ksi}$$

The maximum bending stress(σ_b) in the trunnion is calculated as:

$$\sigma_b = \frac{\text{(Weight of basket + transfer cask)} \times L \times 1.1}{(2) \times S}$$

where:

L = Distance between transfer cask outer wall and mid-point of load application

S = Trunnion section modulus

σ_b = 1.9 ksi

The maximum trunnion principal stress (S_I), is determined by combining shear stress and bending stress as follows (conservative because they occur at different points):

$$S_I = \sigma_b/2 + [(\sigma_b/2)^2 + \tau^2)]^{0.5}$$

$$= 2.7 \text{ ksi}$$

Therefore, the factors of safety, ϕ_u(ultimate) and ϕ_y(yield), for the trunnion (S_u = 70 ksi and S_y = 31.9 ksi for trunnion steel, at work temperature) are:

$$\phi_u = S_u/S_I = 25.9 > 5$$

$$\phi_y = S_y/S_I = 11.8 > 3$$

Hence, the trunnion is adequate (meets NUREG 0612-1980/ANSI 14.6-1993) to carry the weight of the transfer cask with the basket fully loaded with fuel and water.

4.7.4.1.2 Transfer Cask Wall

To evaluate the structural integrity of the transfer cask wall, an ANSYS finite element analysis was performed with the model shown in Figure 4.7-5. The model focuses on the transfer cask wall region near the trunnion because this is the most critical region. Only a quarter of the transfer cask is modeled due to symmetry. The 3-D "SOLID45" and "SHELL63" elements are used for the trunnion and transfer cask shells respectively.

NUREG 0612/ANSI N14.6 states that safety factors of 3 and 5 only apply to stresses that would not be relieved by local yielding, i.e., general stresses. The maximum relevant principal stress in the transfer cask wall caused by the lifting load applied at the trunnion was calculated to be:

Transfer Cask Wall Max. Principal Stress, S_I = 6.69 ksi

Therefore, the factor of safety on this stress are:

Transfer Cask Wall Factors of Safety $\phi_u = S_u/S_I = 70/6.69$

$= 10.5 \geq 5$

$\phi_y = S_y/S_I = 45.6/6.69$

$= 6.8 \geq 3$

In addition, the highest local stress in the trunnion-to-shell junction area was calculated to be 13.4 ksi.

Therefore, the factor of safety on this stress is:

Trunnion to Shell Factors of Safety $\phi_u = S_u/S_I = 70/13.4$

$= 5.2 \geq 5$

$\phi_y = S_y/S_I = 45.6/13.4$

$= 3.4 \geq 3$

4.7.4.1.3 Shield Door Rail and Welds

The shield door rails must support the weight of a fully loaded basket and the weight of the shield doors themselves (a total of approximately 101,500 lbs). The rail design consists of a thick steel plate welded to the bottom of a rectangular solid section of steel. The rail is welded to the bottom plate of the transfer cask wall. For the analysis, the rails were assumed to have an overall length of 48 inches (i.e., the supported length of the closed shield doors). The shield door rail design is shown on Figure 4.7-6.

The design load for the rails (considering 10% dynamic factor) is

$$W = 101{,}500 \cdot 1.1$$

$$= 111{,}650 \text{ lbs}$$

The structural integrity of the rails is evaluated by first considering the rail bottom plate and its welds. The shear stress in the rail bottom plate due to the applied load of W is:

$$\tau = \frac{W}{2 \times L \times t} = 0.8 \text{ ksi}$$

where:

W = Design load = 111,650 lbs

L = Rail length

t = Rail bottom plate thickness

The maximum bending stress in the rail bottom plate, σ_b, occurs at the section through the inner bottom weld and is calculated below.

$$\sigma_b = M/(Lt^2/6) = (W/2) \times \delta/(Lt^2/6) \qquad (4.9)$$

where:

M = Moment in bottom plate
= $(W/2) \times \delta$

δ = Applied load moment arm (1.375 inches)

σ_b = 4.3 ksi

The bottom plate maximum principal stress is then,

Bottom Plate S_1 = $\sigma_b/2 + [(\sigma_b/2)^2 + \tau^2]^{0.5}$
= 4.4 ksi

which based on material properties of the material (S_u = 58 ksi, S_y = 32.8 ksi) provides a safety factor of:

$\phi_u = 13.2 > 5$

$\phi_y = 7.5 > 3$

The rail lower welds were evaluated by first determining the reactive forces, F_o and F_i, experienced by the outer and inner welds due to the applied load. These forces are found from simple balance of forces and moments.

$$F_o = (\delta/w) \times (W/2) = 11.8 \text{ kips}$$

$$F_i = [(w + \delta)/w](W/2) = 67.7 \text{ kips}$$

where:

w = Distance between welds

Therefore, for the two lower welds the maximum shear stress ($\tau_{lower\ weld}$) occurs at the inner groove weld and is calculated as:

$$\tau_{lower\ weld} = F_i/(\delta_w \times L) = 2.3 \text{ ksi}$$

and the maximum principal stress at the weld is (for pure shear condition):

$$\text{Lower Weld } S_1 = \tau_{lower\ weld} = 2.3 \text{ ksi}$$

Which provides factors of safety of:

$$\phi_u = 25.2 > 5$$

$$\phi_y = 14.3 > 3$$

4.7.4.1.4 Welds Attaching the Rails to Transfer Cask Shell

The load on the weld between the rail and transfer cask wall includes the loaded wet basket as well as the weight of doors and rails for the total of approximately 106,500 lbs (117,150 lbs with a 1.1 load amplification factor). Evaluation of the structural integrity of the rail upper welds is affected by the curved geometry of the outer weld (see Figure 4.7-6). This effect stems from the fact that the load distribution between the two upper welds varies with position along the rail. The analysis is done using the standard methodology of treating the weld as a line. The area and section modulus of the weld are calculated to be 93.4 in. and 278.0 in^2 respectively. The center of gravity location is shown in Figure 4.2-8. Then the applied moment (M) is calculated to be 210 kips.

Weld force per unit of length (F_w) is calculated as:

$$F_w = (W/2) / A_w + M / S_w = 1.4 \text{ kips/in}$$

Conservatively assuming a 1/2" fillet weld, the stress (f_w) is calculated as:

$$f_w = F_w / (t_w / \sqrt{2}) = 4.0 \text{ ksi}$$

The ultimate and yield safety factors (ϕ_u and ϕ_y) based on material properties are:

$$\phi_u = S_u/S_1 = 14.5 > 5$$

$$\phi_y = S_y/S_1 = 8.2 > 3$$

4.7.4.1.5 Top Cover Plate

The purpose of the top cover plate is to prevent inadvertent lifting of the basket out of the transfer cask. Therefore, the cover plate must have sufficient strength to support the transfer cask (since an inadvertent basket lift would imply lifting the entire transfer cask by the cover plate). Since this would be an off-normal condition, NUREG-0612 safety factors do not apply and AISC allowable stresses are used for design.

The cover plate is a steel ring with a center opening slightly smaller than the basket diameter. The central opening allows access to the basket Hoist Rings when lowering the basket into the Concrete Cask. Sixteen bolts hold the cover plate in place.

The stresses on the inner and outer edges , σ_i and σ_o, of the cover plate can be calculated from the following equations (Reference 4.10).

$$\sigma_i = [\frac{-3W}{4\pi m t^2}][(m+1)(2\ln\frac{a}{r_0} + \frac{r_0^2}{a^2} - 1)] - \frac{6M}{t^2}\frac{[a^2(m-1) - b^2(m+1)]}{[a^2(m-1) + b^2(m-1)]} \quad (4.10)$$

$$\sigma_0 = [\frac{3W}{2\pi t^2}][1 - \frac{r_0^2}{a^2}] + \frac{6mM}{t^2}\frac{[2b^2]}{[a^2(m-1) + b^2(m+1)]} \quad (4.11)$$

where:

M = $[W/8\pi m]\,[(m+1)(2\ln(a/r_o) + r_o^2/a^2 - 1)]$

W = Weight of transfer cask (with lid)

r_o = Radius of applied load = basket outer radius

a = Cover plate bolt circle radius

b = Radius of cover plate central opening

m = 1/Poisson's ratio = 1/0.3 = 3.33

t = Cover plate thickness

Substituting numerical values into the above formulas yields,

σ_i = 1.2 ksi

σ_o = 13.2 ksi

The shear stress on the outer edge of the cover plate, τ_o, is calculated as:

τ_o $= W / 2\pi at = 0.5$ ksi

Therefore, the cover plate maximum principal stress due to the applied load is:

S_1 $= \sigma_o/2 + [(\sigma_o/2)^2 + \tau^2]^{0.5} = 13.2$ ksi

Comparing this stress level to the acceptance criteria shows the cover plate is structurally adequate, i.e.,

$$13.2 \text{ ksi} < 24.6 \text{ ksi } (0.75F_y \text{ allowable per AISC})$$

4.7.4.1.6 Cover Plate Bolts

The load on a single bolt due to the reactive force caused by an inadvertent basket lift is:

$$F_F = W/16 = 7.5 \text{ kips}$$

The load on each bolt due to the bending moment in the plate (prying action) is:

$$F_M = 2\pi a M_r/16\,L = \frac{2\pi a\, \sigma_o t^2/6}{16 \times L}$$

where:

L = Radial distance from outer edge of cover plate to bolt circle

M_r = Moment per circumferential length = $\sigma_o t^2/6$

t = Cover plate thickness

σ_o = 13.2 ksi, see cover plate calculation section

a = Bolt circle radius

which after numerical evaluation yields:

$$F_M = 21.5 \text{ kips}$$

Therefore, the tension on each bolt, F, is calculated as:

$$F = F_F + F_M = 29.0 \text{ kips}$$

which is within the acceptable range, i.e.,

F = 29.0 kips < 34.6 kips (allowable load for 1" A-325 bolt per AISC)

4.7.4.2 Air Pad System

The Concrete Cask may be lifted from below using air pads. This bottom lift is the normal lifting mode employed when moving the Concrete Cask out to the storage pad to its storage location. It should also be noted that the lift device is not considered important to safety since the Concrete Cask lift is limited to only 3 inches. The air pad system accommodates the fully loaded weight of a Concrete Cask (i.e., Concrete Cask, basket, 24 fuel assemblies with control components) which is conservatively assumed to be 300,000 lbs. The adequacy of the lift is evaluated by calculating the bearing pressure on the cask bottom and comparing it to allowable bearing pressure per ACI-349. Allowable bearing stress is:

$$P_b = \phi(0.85f_c) \quad (4.7)$$
$$= 0.7(0.85 \cdot 4000) = 2{,}380 \text{ psi}$$

The Air Pad bearing pressure is calculated based on the bearing area

p= Weight / (air pad area - inlet duct area - air pad area outside of cask envelope)

=42.5 psi

Air pad bearing stresses are negligible. No shear forces or bending moments will exist in the cask because the air pads effectively cover the whole bottom area. Hence, the concrete will not crush during a bottom lift of the Concrete Cask.

4.7.4.3 Lifting Yoke

The Lifting Yoke is designed to lift the combined weight of the Transfer Cask and a fully loaded basket. For conservatism the analysis assumed a loaded Transfer Cask weight of 215,000 lbs. The Lifting Yoke weight is assumed to be 7,000 lbs. A dynamic load amplification factor of 1.1 was assumed.

Figure 4.7-4 provides a drawing of the Lifting Yoke. The maximum principal stress was calculated using the following equation:

$$\sigma_l = \frac{\sigma}{2} + \sqrt{\left[\left(\frac{\sigma}{2}\right)^2 + \tau^2\right]}$$

Where: σ = bending stress

τ = shear stress

The maximum principal stress was calculated for the Lifting Yoke components. The components analyzed were: beams, pins, yoke arms, and hook. The maximum principal stress for each of these components was then compared to allowable yield and ultimate stress for its material of construction. These components were found to have safety factors of greater than 3 for yield stress, and greater than 5 for ultimate stress.

The lowest service temperature for the Lifting Yoke was determined to be -3°F. Use of the Lifting Yoke is limited to ambient temperatures greater than -3°F to satisfy fracture toughness requirements.

4.7.4.4 Hoist Rings

The adequacy of the basket lifting devices is demonstrated by considering each of the Hoist Rings, the basket structural lid and its weld to the shell. The design of the basket incorporates 8 lifting points to allow for two redundant load paths each using 4 Hoist Rings.

The analysis assumes a dynamic load increase factor of 10% and calculated the stresses associated with a single load path using 4 Hoist Rings. In addition, since the load is applied at a slight angle, a horizontal load component is introduced.

The hoist ring evaluation is based on the manufacturer's rated capacity and provided load factors. The safety factor for the Hoist Rings (single load path of 4 lift points) was determined to be 5.33. This assumed a maximum lift angle of 12° (this equated to a height between the basket hoist rings and lifting point of not less than 11 feet). Administrative controls will be implemented to ensure the lift angle is maintained below this limit.

The minimum thread engagement for the Hoist Rings was determined based on the strengths of materials to ensure stripping of thread material in the structural lid would not occur (since the structural lid is made of softer material than the hoist ring bolts). The minimum thread engagement was determined to be 1.95 inches and will be ensured by administrative procedures.

The structural lid and basket shell are evaluated based on the finite element analysis results of the Sierra Nuclear Corporations VSC-24 calculations (Reference 8). The lid and wall thickness for the ISFSI PWR Baskets and GTCC Baskets are the same as used for the VSC-24. The PWR Baskets and GTCC Baskets have a slightly increased diameter. Adjustments are made for the new loads as well as for the slight increase in diameter. The load change is incorporated by scaling. Geometry changes are ratioed, since stress is proportional to $(1-r^2/a^2)$, where r is the radius of the load application and a the plate radius.

The maximum principal stress was determined to be 5.0 ksi which results in a safety factor of 13.2 for ultimate stress and 4.3 for yield stress.

The above analysis was based on 4 Hoist Rings supporting the basket load. The baskets will be lifted using 8 Hoist Rings. Section 8.2.13.1.2 analyzes the accident condition in which a basket lift results in lifting the Transfer Cask.

4.7.5 THERMAL EVALUATION DURING FUEL TRANSFER

The transfer cask model and calculations are presented below. It only addresses transfer of fuel since the heat load produced by GTCC canisters is substantially lower.

4.7.5.1 Transfer Cask Heat Transfer Modes

Heat is generated in the fuel assemblies and transferred to the basket transfer cask (transfer cask) from the basket surface by radiation, conduction and convection through the air annulus between the basket and transfer cask. The heat is then conducted through the transfer cask wall and convected and radiated from the transfer cask outer surface. Heat transfer modes inside the basket are the same as discussed in Section 4.2.6.4.

The ANSYS/THERMAL finite element model similar to that of the Concrete Cask is used for the analysis and presented in Figure 4.7-5. Thermal conductivity of the neutron shield region was based only on the steel fins; no credit was taken for the neutron shield conductivity. An ambient temperature of 75°F was used in the analysis.

As described in Section 4.2.6.4, the basket internals are not modeled in detail since the only interest for the basket in this model is the shell temperature. This analysis determines temperature distribution in the Transfer Cask and basket shell. Figure 4.7-7 presents the profile through the cask wall at the hottest section. The highest shell temperature is used as a boundary condition in the basket analysis described below. The results are summarized in Table 4.2-12.

4.7.5.2 Basket Thermal Hydraulic Model

The basket was modeled using the ANSYS/THERMAL finite element code as discussed in Section 4.2.6. The basket hot-slice model was used to estimate the basket components and fuel temperatures. For the basket drying case, the basket model was modified to represent vacuum conditions. The inner helium elements were removed. The resulting model only includes radiation from the guide sleeves to the basket wall. Based on the benchmarks performed for vacuum cases of other cask tests and the higher temperatures, the fuel effective thermal conductivities were left unchanged. The shell temperature from the transfer cask was used as a bounding condition for the basket analysis. The results are presented in Table 4.2-12.

4.8 REFERENCES

1. SNC-95-71 SAR, Revision 0, "Safety Analysis Report for the TranStor™ Shipping Cask System," Sierra Nuclear Corporation, December 1995.

2. E. R. Gilbert et al., "Control of Degradation of Spent LWR Fuel During Dry Storage in an Inert Atmosphere," PNL-6364, Pacific Northwest Laboratory, Richland, WA (1987).

3. J. M. Creer et al., "The TN-24P PWR Spent Fuel Storage Cask: Testing and Analysis," EPRI-NP-5128, Electric Power Research Institute, Palo Alto, CA (1987).

4. J. M. Creer et al., "BWR Spent fuel Storage cask Performance Test," PNL-577, Pacific Northwest Laboratory, Richland, WA (1986).

5. J.M. Creer et al., "The Castor V/21 PWR Spent Fuel Storage Cask: Testing and Analyses, EPRI NP-4887, Electric Power Research Institute, Palo Alto, CA (1986).

6. J. M. Creer et al, "The MC-10 PWR Spent Fuel Storage Cask: Testing and Analysis," EPRI NP-5268, Electric Power Research Institute, Palo Alto, CA (1987).

7. R. A. McCann, "Comparison of HYDRA Predictions to Temperature Data From Two Single-Assembly Spent Fuel Heat Transfer Test," PNL-6074, Pacific Northwest Laboratory, Richland, WA (1986).

8. Sierra Nuclear Corporation, "Safety Analysis Report for the Ventilated Storage Cask System", PSN-91-001, Revision 0A, December 1993.

9. "SCALE-PC Modular Code System for Performing Criticality Safety Analysis for Licensing Evaluation, Version 4.1," Oak Ridge National Laboratory; CCC-619.

10. R. J. Roark, Formulas for Stress and Strain, McGraw-Hill Book Co., New York, NY (1965).

11. Bechtel Power Corporation letter to PGE, Document Control No. T040898 dated March 4, 1996 titled,"Concrete Cask Temperature Review, GSA #618"

Table 4.2-1

Fabrication Specification Summary
PWR Basket, GTCC Basket and Basket Overpack

MATERIALS

Material in accordance with design drawings. References to ASME Section III are 1992 revision.

FABRICATION[1]

Cutting, welding, and forming for the confinement boundary in accordance with ASME, Section III, NC-4000. Stamping is not required.
Cutting, welding, and forming for the PWR Basket internals in accordance with ASME, Section III, NG-4000. Stamping is not required.
Cutting, welding, and forming for the GTCC Basket internals in accordance with ASME, Section III, NF-4000. Stamping is not required.
Filler metals are ASME, Section II material.
Welders and welding operators are qualified in accordance with ASME Section IX.
Welding procedures are written and qualified in accordance with ASME Section IX
Visual examinations of welds as specified in ASME, Section V, Article 9.
Welds are liquid penetrant or magnetic particle examined in accordance with the requirements of ASME Section V, Article 6 or Article 7, respectively.
Welds to be radiographed examined in accordance with the requirements of ASME Section V, Article 2 and Section III, NC-5300.
Personnel performing examinations qualified in accordance with the quality assurance program, and SNT-TC-1A (December 1988).
Surfaces cleaned to surface classification C or better as defined in ANSI N45.2.1, Section 2.

PACKAGING AND SHIPPING

Packaging and shipping are in accordance with ANSI N45.2.2.

QUALITY ASSURANCE

The basket is fabricated under a quality assurance program that meets the applicable requirements of 10 CFR 72 Subpart G

1 Deviations from specified code and justification are provided in Table 4.2-1a.

Table 4.2-1a
ASME Code Deviations
ASME Section III Subsection NC

Section	Requirement	Exception/Justification
3211.1	Establishes scope for vessel design	Requirement of NC-5250 (referenced in this subparagraph) to radiograph all Category C joints is not met for the basket closure welds. Radiographic examination of these field welds is not feasible; redundant leakage barriers used in lieu of this requirement. Lugs and corner support tubes are not attached to the shell with continuous welds (per NC-4267) due to lack of accessibility. Lugs are not loaded while the shell is serving as a pressure boundary. Detailed drop analysis includes the actual weld configuration and considers influence on the shell as applicable. Set of calculations, drawings, and specifications is used in lieu of the Design Report per Appendix C. However, the information is provided and the intent of the Code is met.
3223.2	Requires Design Report in Appendix C format	A set of calculations, drawings, and specifications is used in lieu of the Design Report per Appendix C. However, the information is provided and the intent of the Code is met.
3252	Describes permissible types of welded joints	As stated above, basket closure welds (Category C) can not be radiographed per NC-2553.
3254	Refers to NC-4267 for structural attachment welds.	Lugs and corner support tubes are not attached to the shell with continuous welds (per NC-4267) due to lack of accessibility. Lugs are not loaded while the shell is serving as a pressure boundary. Detailed drop analysis includes the actual weld configuration and considers influence on the shell as applicable.
4266	Requirements for category D weld joints in vessels designed to NC-3200	Valve covers (Cat. D welds) are fillet welds and do not meet Figure 4266. Small diameter and weld size provide large factors of safety. Lid reinforcement is provided and redundant valve covers are installed for the PWR basket [see NC-3252].
4267	Types of attachment welds allowed in vessels designed to NC-3200	Lugs and corner support tubes are not attached to the shell with continuous welds due to lack of accessibility. Lugs are not loaded while the shell is serving as a pressure boundary. Detailed drop analysis includes the actual weld configuration and considers influence on the shell as applicable [see NC-3211.1].
5253	Category C welded joints for vessels designed to NC-3200 require RT examination.	Structural lid welds are not radiographed due to limited accessibility and redundant PWR basket closure. Welds are tested using liquid penetrant or magnetic particle examination and helium leak testing. [See NC-3211.1]
6113	Requires pressure test in presence of inspector.	The inspection is performed but not by a Code certified inspector. Acceptable because the vessel is not N-stamped.
8100	References NCA-8000 for nameplate, stamping, and report requirements.	Code Stamping is not provided. The basket is not a part of a nuclear power system as defined by NCA-1110.

ASME Section III Subsection NG

Section	Requirement	Exception/Justification
2121	Requires use of ASME (SA) materials.	The basket internals, failed fuel and fuel debris cans do not use SA materials. Materials used meet ASTM specifications that are identical to corresponding ASME specifications. All materials are supplied as important to safety, CMTRs are provided.
4110	General requirements specify use of materials per NG-2000.	NG-2000 is not entirely met (see NG-2121).
4121	Means of certification of materials.	ASTM materials are utilized for NG components (see NG-2121).
8100	References NCA-8000 for nameplate, stamping, and report requirements.	Code Stamping is not provided. The basket is not a part of a nuclear power system as defined by NCA-1110.

Table 4.2-2
Concrete Cask Fabrication Summary

MATERIALS

Concrete mix in accordance with requirements of ACI 318.
Type II Portland Cement, ASTM C150.
Fine aggregate ASTM C33.
- Thermal expansion coefficient of $6.0 x 10^{-6}$in/in-°F or less
- Composed of quartz sand, sand stone sands, or sand of the following minerals:
  - limestone, dolomite, marble, basalt, granite, gabbro, and/or rhyolite.

Coarse aggregate ASTM C33.
- Thermal expansion coefficient of $6.0 x 10^{-6}$in/in-°F or less
- Composed of the following minerals:
  - limestone, dolomite, marble, basalt, granite, gabbro, and/or rhyolite.
- Minimum bulk specific gravity of 2.60
- Shall not exceed 1.5 in. nominal size and the amount of flat and elongated particles shall be less than 15% by weight

Admixtures:
- Water Reducing ASTM C494.
- Pozzolanic Admixture ASTM C618.
- Air Entraining ASTM C260

Compressive Strength 4000 psi.
Air Entrainment: 3% - 6%
Steel components are ASTM A-36
Reinforcement are ASTM A-615

WELDING

Visual inspection of girth and longitudinal welds is performed as specified in ASME. Section III, Subsection NF
Visual inspection of other welds to assure no concrete leakage during pouring.

CONSTRUCTION

Strength tests are performed for each truckload of concrete
- Test specimens are cured per ASTM C31 and tested in accordance with ASTM C39.
- Formwork in accordance with ACI 318.
- Grade, type, and details of reinforcing steel in accordance with the referenced drawings.
- Embedded items conform with ACI 301 and the referenced drawings.
- The placement of concrete in accordance with ACI 318 and ACI-301
- Surface finish in accordance with ACI 301.

QUALITY ASSURANCE

Construction shall be under a quality assurance program that meets the applicable requirements of 10 CFR 72 subpart G.[1]

1 Deviations from specified code and justification are provided in Table 4.2-2a.

Table 4.2-2a

Concrete Cask Code Deviations

No.	Requirement	Exception/Justification
1.2	Specifies how drawings and calculations must be handled	The loads used in the design are covered in the calculations rather than the drawings and specifications.
A.4	The limits for bulk, (150°F) & local area (200°) concrete temperature.	A long term temperature limitation of 250°F is used. This increased limit is based on test data from several research efforts which show that concrete of similar composition to that used in the casks does not suffer loss of strength when exposed to temperatures in the range of 250°F.

Table 4.2-3

Conformity to Requirements

Requirement	Requirement Summary	Basis for Conformance
10 CFR 72.122(a) Quality Standards	Structures , systems, and components important to safety must be designed tested and fabricated to quality standards commensurate with their function	Quality assurance program in accordance with 10 CFR 72.104(d) implemented for ISFSI activities. Refer to SAR Chapter 11.
10 CFR 72.122(b) Protection Against Environmental Conditions and Natural Phenomena	Structures , systems, and components important to safety must be designed to accommodate the effects of and be compatible with site characteristics and to withstand postulated accidents	SAR Chapter 2 describes the site characteristics and defines credible environmental conditions. SAR Chapter 8 provides analysis to demonstrate design conformance.
OAR 345-26-390(4)(b)	The ISFSI shall be designed such that in the event of a Seismic Margin Earthquake, anticipated damage to spent nuclear fuel or containers will not preclude acceptance at federally licensed disposal facility.	SAR Section 8.2.5.2 demonstrates that Seismic Margin Earthquake does not result in damage to storage system
10 CFR 72.122(c) Protection Against Fire and Explosions	Structures , systems, and components important to safety must be designed and located so that they can continue to perform their safety function under credible fires and explosion exposure conditions	SAR Section 8.2.9 discusses impact of fire on the ISFSI, Section 8.2.14 discusses explosions
10 CFR 72.122(d) Sharing of Structures and Components	Structures , systems, and components important to safety must not be shared between the ISFSI or other facilities unless it is shown that such sharing will not impair the capability of either facility to perform its safety function.	The ISFSI is designed for stand alone operations and does not rely on other facilities to support performance of its safety function. The ISFSI does not share its facilities with any other facility.

Conformity to Requirements

Requirement	Requirement Summary	Basis for Conformance
10 CFR 72 122(e) Proximity of Sites	An ISFSI located near other nuclear facilities must be designed and operated to ensure that the cumulative effects of their combined operations will not constitute an unreasonable risk to health and safety of the public.	The Environmental Report, Chapter 5 discusses the impact of ISFSI operation on the environment. The Environmental Report concluded that ISFSI operations have no significant impact on the environment.
10 CFR 72.122(f) Testing and Maintenance of Systems and Components	Systems, and components important to safety must be designed to permit inspection, maintenance, and testing.	ISFSI structures , systems, and components important to safety are designed to minimize the need for testing. System performance monitoring is provided by temperature monitoring of the air outlet. Surveillances to ensure proper operations are provided in the Technical Specifications.
10 CFR 72.122(g) Emergency Capability	Structures , systems, and components important to safety must be designed for emergencies. The design must provide for accessibility to the equipment of on-site and available offsite emergency facilities and services.	Access to off-site emergency facilities and services such as hospitals, fire and police departments, ambulance services, and other emergency agencies is discussed in the Emergency Plan.
10 CFR 72.122(h) Confinement Barriers and Systems	The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined.	Section 4.2.3 discusses the confinement boundaries. The PWR Basket shell provides the confinement barrier even in the event of gross rupture of fuel clad. For the range of design basis accidents presented in Chapter 8 there is no loss of confinement boundary.
10 CFR 72.122(i) Instrumentation and Control Systems	Instrumentation and controls systems must be provided to monitor systems that are important to safety over anticipated ranges for normal and off-normal operation	The Trojan ISFSI is passive by design and requires no controls for operation. Storage system monitoring will be performed using measuring and test equipment calibrated in accordance with the Quality Assurance program.

Conformity to Requirements

Requirement	Requirement Summary	Basis for Conformance
10 CFR 72.122(j) Control Room or Control Area	A control room or control area, if appropriate for the ISFSI design, must be designed to permit occupancy and actions to be taken to monitor the ISFSI safely under normal conditions, and to provide safe control under off-normal or accident conditions.	Chapter 5 provides a discussion of ISFSI normal and emergency operations. These operations are monitored and performed at the Storage Pad. There is no requirement for remote monitoring, therefore there is no requirement for a control room or control area.
10 CFR 72.122(k) Utility or Other Services	Requires utility services that are important to safety be provided with redundant capabilities and adequate capacities.	The design of systems, structures and components important to safety does not rely on utility services. Section 4.1.2.3 provides additional discussion.
10 CFR 72.122(l) Retrievability	Storage systems must be designed to allow ready retrieval of spent fuel for further processing or disposal.	Spent nuclear fuel is stored within seal welded closure baskets. The ISFSI is designed to allow retrieval of the baskets and placement into a transportation cask. Chapter 5 discusses operations associated with retrieval and shipment of spent nuclear fuel.

Conformity to Requirements

Requirement	Requirement Summary	Basis for Conformance
10 CFR 72.124 Criteria for Nuclear Criticality Safety	The spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely independent, and concurrent or sequential changes have occurred. When practical the design must be based on favorable geometry or permanently fixed neutron absorbing material, or both.	The PWR Basket is designed to maintain subcritical conditions for credible accidents. The K_{eff} for the PWR Basket is based on design geometry and does not credit the neutron absorbing material for the dry storage condition. In order to invalidate criticality assumptions, the PWR Basket volume would have to be filled with water. In order for this condition to occur, the ISFSI location would require a flood in excess of the design basis flood scenario coincident with confinement barrier failure. Neither of these conditions are considered credible based on the accident analysis presented in Chapter 8.
10 CFR 72.126 (a) Exposure Control	Structures, systems and components for which operation, maintenance, and required inspections may involve occupational exposure, must be designed, fabricated, located, shielded, controlled, and tested so as to control internal and external exposure to personnel.	Radioactive materials are confined within a welded steel enclosure. As a result, it is not anticipated that personnel will be exposed to airborne radioactivity. Radiation exposure control is implemented by limiting access to the ISFSI. Access control will be implemented in accordance with radiation control and security procedures. Chapter 7 discusses the potential sources for radiation exposure and the design, procedures and programs utilized to implement ALARA concepts.

Table 4.2-3

Conformity to Requirements

Requirement	Requirement Summary	Basis for Conformance
10 CFR 72.126(b) Radiological Alarm Systems	Radiological alarm systems must be provided in accessible work areas as appropriate to warn operating personnel of radiation and airborne radioactive concentrations above a setpoint.	Section 7.3.1 of the SAR describes the design features of the installation and equipment that ensures that personnel exposure to radiation is ALARA. Section 7.3.4 describes the radioactive monitoring instrumentation.
10 CFR 72.126(c) Effluent and Direct Radiation Monitoring	As appropriate, effluent systems must be provided with means for measuring the amount of radionuclides in effluents. Areas containing radioactive materials must be provided with systems for measuring direct radiation.	The confinement features of the Trojan ISFSI storage system are such that radioactive releases are not considered credible, thus, effluent radiation monitoring systems are not required. Direct radiation monitoring consists of thermoluminescent detectors (TLDs) posted at the perimeter of and in the Controlled Area near the Storage Casks. Radiation protection equipment, instrumentation, and facilities are discussed in Section 7.5.2 of the ISFSI SAR.
10 CFR 72.126(d) Effluent Control	The ISFSI must be designed to provide a means to limit to levels as low as reasonably achievable.	The ISFSI is designed to ensure confinement of stored radioactive materials. The confinement design features are discussed in Section 4.2.3 of the SAR. There is no anticipated release of radioactive material during normal operations.
10 CFR 72.128(a) Spent Fuel Storage and Handling Systems	Spent fuel storage and other systems that might contain or handle radioactive materials must be designed to ensure adequate safety under normal and accident conditions.	The ISFSI is designed to provide confinement of spent nuclear fuel and related radioactive material for the spectrum of operating conditions and accidents.

Table 4.2-3

Conformity to Requirements

Requirement	Requirement Summary	Basis for Conformance
10 CFR 72.128(b) Waste Treatment	Radioactive waste treatment facilities must be provided. Provisions must be made for the packing of site-generated low-level wastes in a form suitable for storage onsite awaiting transfer to disposal sites.	Generation of radioactive waste is not anticipated at the ISFSI. Radioactive material stored at the facility is contained within welded enclosures. Site-generated waste confinement is discussed in Chapter 6 of the SAR. Since there is no anticipated generation of radioactive waste, a waste treatment facility has not been included in the design.
10 CFR 72.130 Criteria for Decommissioning	The ISFSI must be designed for decommissioning.	Decommissioning activities consist primarily of transferring the baskets for permanent off-site disposal or storage. The storage system has been designed to minimize contamination of the cask exterior during loading and unloading operations. No contamination is expected on the concrete cask and because of low neutron flux levels activation of the concrete and steel is considered insignificant.
OAR 345-26-0390(4)(j) Design Life	The ISFSI must have a minimum design life of 40 years.	The ISFSI is designed for 40 year life.

Table 4.2-4

Weights and Centers of Gravity

Item/Configuration	Weight	Center of Gravity
PWR Basket		
Empty (without lids)	27,441	81.1
Loaded Wet-(Fuel, Inserts, water and shield lid)	86,955	91.8
Loaded Dry (Fuel, Inserts, shield and structural lid)	76,379	96.0
Structural Lid	2,700	--
Shield Lid	7,357	--
GTCC Basket		
Empty (without lids or GTCC Cans)	27,931	78.2
Loaded Wet (GTCC Waste, water and shield lid)	86,450	93.4
Loaded Dry (GTCC Waste, shield and structural lids)	75,738	97.7
Structural Lid	2,700	--
Shield Lid	9,754	--
Basket Overpack		
Empty (without lid)	6,558	77.8
Structural Lid	1,006	--
Transfer Cask		
Empty (no lid)	120,042	85.1
Empty (with lid)	120,631	85.6
Lid	397	--
With PWR Basket (empty without lids)	148,072	86.7
With PWR Basket (loaded wet with shield lid)	208,317	92.6

Transfer Cask (continued)

Table 4.2-4

Weights and Centers of Gravity

Item/Configuration	Weight	Center of Gravity
With PWR Basket (loaded dry with lids)	197,010	93.7
With GTCC Basket (loaded wet with shield lid)	207,812	93.3
Concrete Cask		
Empty (without lid or shield ring)	211,152	103.9
Empty (with lid and shield ring)	213,620	105.1
With PWR Basket (loaded dry)	289,999	108.4
With GTCC Basket (loaded dry)	289,357	108.9
With PWR Basket and Basket Overpack	297,616	108.8
With GTCC Basket and Basket Overpack	296,975	109.3
Lid	1,235	--
Shield Ring	1,232	--

Table 4.2-5
Mechanical Properties of Steels Used in the Storage System

Material Specification	Type of Grade	Temp (°F)	Yield S_y (ksi)	Ultimate S_u (ksi)	Allowable S_m (ksi)	Elastic Modulus[1] (10^6 psi)	Coefficient of Thermal Expansion[2] (10^{-6} in/in°F)
ASME SA-516	70	70	38.0[3]	70.0[4]	23.3[5]	29.5	—
		100	38.0[3]	70.0[4]	23.3[5]	—	5.53
		200	34.6[3]	70.0[4]	23.1[5]	28.8	5.89
		300	33.7[3]	70.0[4]	22.5[5]	28.3	6.26
		400	32.6[3]	70.0[4]	21.7[5]	27.7	6.61
		500	30.7[3]	70.0[4]	20.5[5]	27.3	6.91
		600	28.1[3]	70.0[4]	18.7[5]	26.7	7.17
		700	27.4[3]	70.0[4]	18.3[5]	25.5	7.41
ASTM A-36		70	36.0[6]	58.0[7]	19.3[8]	Use SA-516 Grade 70 data	
		100	36.0[6]	58.0[7]	19.3[8]		
		200	32.8[6]	58.0[7]	19.3[8]		
		300	31.9[6]	58.0[7]	19.3[8]		
		400	30.8[6]	58.0[7]	19.3[8]		
		500	29.1[6]	—	19.3[8]		

[1] ASME, Sec. II, Part D, Subpart 2, Table TM-1

[2] ASME, Sec. II, Part D, Subpart 2, Table TE-1

[3] ASME, Sec. II, Part D, Subpart 1 Table Y-1

[4] ASME, Sec. II, Part D, Subpart 1, Table U

[5] ASME, Sec. II, Part D, Subpart 1, Table 2A

[6] ASME, Code Case N-71-16, Table 4

[7] ASME, Code Case N-71-16, Table 5

[8] ASME, Sec. II, Part D, App. 2

Table 4.2-5 (continued)

Mechanical Properties of Steels Used in the Storage System

Material Specification	Type or Grade	Temp (°F)	Yield[1] S_y (ksi)	Ultimate[2] S_u (ksi)	Allowable[3] S_m (ksi)	Elastic Modulus[4] (10^6 psi)	Coefficient of Thermal Expansion[5] (10^{-6} in/in °F)
ASME SA-240	304	70	30.0	75.0	20.0	28.3	—
		100	30.0	75.0	20.0	—	8.55
		200	25.0	71.0	20.0	27.6	8.79
		300	22.5	66.0	20.0	27.0	9.00
		400	20.7	64.4	18.7	26.5	9.19
		500	19.4	63.5	17.5	25.8	9.37
		600	18.2	63.5	16.4	25.3	9.53
		700	17.7	63.5	16.0	24.8	9.69
ASME-SA-240	304L	70	25.0	70.0	16.7	Use SA-240 Type 304 data	
		100	25.0	70.0	16.7		
		200	21.3	66.2	16.7		
		300	19.1	60.9	16.7		
		400	17.5	58.5	15.8		
		500	16.3	57.8	14.8		
		600	15.5	57.0	14.0		
		700	14.9	56.2	13.5		

[1] ASME, Sec. II. Part D. Subpart 1, Table Y-1
[2] ASME, Sec. II. Part D. Subpart 1. Table U
[3] ASME, Sec. II. Part D, Subpart 1, Table 2A
[4] ASME, Sec. II. Part D, Subpart 2, Table TM-1
[5] ASME, Sec. II. Part D, Subpart 2, Table TE-1

Table 4.2-5 (continued)

Mechanical Properties of Steels Used in the Storage System

Material Specification	Type or Grade	Temp. (°F)	Yield[1] S_y (ksi)	Ultimate[2] S_u (ksi)	Allowable[3] S_m (ksi)	Elastic Modulus (10^6 psi)	Coefficient of Thermal Expansion (10^{-6} in/in °F)
ASTM A -588		100	50	70.0	23.3	Use SA-516 Grade 70 data	
		200	47.5	70.0	23.3		
		300	45.6	70.0	23.3		
		400	43.0	70.0	23.3		

[1] ASME. Code Case. N-71-15. Table 3

[2] ASME. Code Case. N-71-15. Table 5

[3] ASME. Code Case. N-71-15. Table 1

Table 4.2-6

Properties of Concrete Used in Concrete Cask

Temp. °F	Density (lb/ft^3)	Thermal Conductivity (BTU/hrft °F)	Compressive Strength (psi)	Thermal Expansion (in/in/°F)	Modulus of Elasticity (psi)
					40 years
70	145	---	4,000	5.5×10^{-6}	---
100	145	0.87	4,000	5.5×10^{-6}	3.64×10^{6}
200	145	---	4,000	5.5×10^{-6}	3.38×10^{6}
300	145	0.82	3,800	5.5×10^{-6}	3.09×10^{6}
400	145	0.80	3,600	5.5×10^{-6}	2.73×10^{6}
500	145	0.78	3,400	5.5×10^{-6}	2.43×10^{6}

These properties are from Reference 8 and ACI 349.

Table 4.2-7

Summary of Maximum Basket Thermal Stresses (ksi)

Component	Maximum Thermal Stress Q (ksi)
Shell	34.1
Bottom Plate	18.4
Structural Lid	4.1
Top Weld	11.3
Shield Lid Weld	17.9
Sleeve	15.4

Table 4.2-8

Basket Maximum Stress Evaluation

Component Location	Stresses	Dead Weight	Design Pressure	Max Thermal	Normal Handling	Total	ASME Service Level A Limit
		Calculated Value, ksi[a]					
Basket Shell	P_m	0.1	0.2	-	0.6	0.9	15.5
	$P_L + P_b$	0.2	2.8	-	2.0	5.0	23.2
	$P + Q$	0.2	2.8	34.1	2.0	39.1	46.4
Bottom Plate	P_m	0.0	0.1	-	0.6	0.7	15.5
	$P_L + P_b$	0.2	1.9	-	1.3	3.4	23.2
	$P + Q$	0.2	1.9	18.4	1.3	21.8	46.4
Structural Lid	P_m	0.0	0.0	-	0.2	0.2	15.5
	$P_L + P_b$	0.1	0.6	-	0.9	1.6	23.2
	$P + Q$	0.1	0.6	4.1	0.9	5.7	46.4
Top Weld	P_m	0.1	0.4	-	0.8	1.3	15.5
	$P_L + P_b$	0.2	0.5	-	0.9	1.6	23.2
	$P + Q$	0.2	0.5	11.3	0.9	12.9	46.4
Sleeve Assembly	P_m	0.1	0.0	-	0.2	0.3	18.3
	$P_L + P_b$	0.1	0.0	-	2.2	2.3	27.5
	$P + Q$	0.1	0.0	15.4	2.2	17.7	54.9
Shield Lid Support Ring Weld	P_m	0.1	0.0	-	0.1	0.2	11.6
	$P_L + P_b$	0.1	0.0	-	0.1	0.2	17.4
	$P + Q$	0.1	0.0	0.0	0.1	0.2	34.8
Shield Lid Weld	P_m	0.1	0.3	-	0.1	0.5	15.5
	$P_L + P_b$	0.1	0.4	-	0.1	0.6	23.2
	$P + Q$	0.1	0.4	17.9	0.1	18.5	46.4

a Values shown are maximums irrespective of location

Table 4.2-9
Summary of Maximum Storage System Temperatures
(Without Overpack)

Solar heat load[1]	no	no	yes[1]	yes[1]	no	no
Special conditions	no	no	no	no	½ air inlets blocked	all air inlets blocked
Ambient Temperature, °F (average over a 24-hour period)	75	-40	100	125	75	75
Extreme Temperature, °F						
• cask outer surface	87	-31	141	166	89	94
• concrete/liner interface	204	60	239	270	228	283
• basket outer surface	287	177	313	338	305	358
Basket pressure (psig)	-2.3	-3.7	-1.9	-1.6	-2.1	-1.4

[1] Insolation is based on sun shining for 12 of 24 hours per day.

Table 4.2-10

Concrete Cask Structural Load Combination Summary

Load Combination		Stress/Load	Maximum[1] Stress/Load	Allowable[1] Stress/Capacity
1	$1.4D + 1.7L$	Shear Normal	0 0.50	- 2.8
2	$1.4D + 1.7L + 1.7H$	Same as Combination No. 1 ($H = 0$)		
3	$0.75(1.4D+1.7L+1.7H+1.7T_o+1.7W)$	Shear Normal	0.0 1.5	0.11 2.8
4	$0.75(1.4D + 1.7L + 1.7H + 1.7T_o)$	Bounded by Load Combination No. 3		
5	$D + L + H + T_o + E_{ss}$	Shear Normal	0.05 1.3	0.11 2.8
6	$D + L + H + T_o + A$	Shear Capacity Moment Capacity Moment Capacity	22.2[2] 593.6[2] 1,056.9[2]	103.5[2] 694.5[2] 1,381[2]
7	$D + L + H + T_a$	Bounded by Combination No. 3 because $T_o = T_a$		
8	$D + L + H + T_a + W_t$	Shear Capacity Moment Capacity	457.5[2] 87,820[2]	1,106[2] 94,170[2]

[1] Units are ksi unless otherwise noted.

[2] Capacities calculated per ACI-349 are used for these load combinations instead of stresses. Capacities are in kips (force) or kips·in (moment).

Table 4.2-11

Summary of Maximum Concrete Cask Thermal Stresses
75°F Ambient Air, Normal Operation

Component	Q (ksi)
Concrete	0.85
Rebar	
Vertical	28.4
Hoop	34.2
Liner	1.4
Cover Plate	1.8
Bottom Plate	10.7

Table 4.2-12

Summary of Cask Thermal Evaluation

Case	Temperatures (°F)						
	Ambient	Solar	Air Outlet	Outer Concrete	Inner Concrete	Basket Shell	Max Clad
Normal Operation							
Limits	-	-	-	**250**	**250**	-	**730**
(without Basket Overpack) Steady State Normal	75	no	180	87	204	287	609
(with Basket Overpack) Steady State Normal	75	no	182	87	204	371	685
Off-normal and Infrequent Events							
Limits	-	-	-	**350**	**350**	-	**1058**
(without Basket Overpack)							
Steady State Severe Cold	-40	no	43	-31	60	177	511
Steady State Severe Hot	100	yes	210	141	239	313	633
12 hour Max Thermal	125	yes	240	166	270	338	655
½ of Inlets Blocked	75	no	202	89	228	305	625
All Inlets Blocked	75	no	263	94	283	358	673
Basket in Transfer Cask w/ He	75	no	-	-	-	434	743
with vacuum	75	no	-	-	-	434	851
(with Basket Overpack)							
Steady State Severe Cold	-40	no	44	-31	59	278	601
Steady State Severe Hot	100	yes	212	141	239	393	705
12 hour Max Thermal	125	yes	242	166	269	414	725
½ of Inlets Blocked	75	no	203	88	221	381	694
All Inlets Blocked	75	no	266	94	284	432	741

Table 4.2-13

Thermal Properties

Material	Temperature (°F)	Specific Heat (BTU/lbm °F)	Thermal Conductivity (BTU/hr-ft-°F)	Density (lbm/ft^3)	Emissivity
Carbon Steel	32-800	0.11	26.0	490	0.8[a]
Stainless Steel	32-800	0.11	9.4	488	0.8[a]
Concrete	32-400	0.21	0.719	141	0.9
Air	-50	0.238	0.0114	0.094	---
	0	0.239	0.0130	0.086	---
	32	0.240	0.0140	0.081	---
	100	0.240	0.0154	0.071	---
	200	0.241	0.0174	0.060	---
	300	0.243	0.0193	0.052	---
	500	0.247	0.0231	0.041	---
	700	0.253	0.0263	0.037	---
Helium	32-800	---	0.1	---	---

a Coated surfaces are expected to have emissivities in excess of 0.9, but the value of 0.8 for steel was used for conservatism.

Table 4.2-14

Summary of Storage System Cooling Air Flow Analysis

Location	Temperature at 75°F Ambient (°F)	Temperature at 100°F Ambient (°F)	Temperature at -40°F Ambient (°F)
Air Inlet	75	100	-40
Air Elevation[1] (inches)			
0 - 16	84	110	-33
16-32	97	123	-23
32 - 48	110	136	-13
48 - 64	123	150	-3
64 - 80	135	163	7
80 - 96	148	177	17
96 - 112	161	190	27
112 - 128	172	202	36
128 - 144	181	211	43
Air Outlet Temperature	180	210	43
Air Flow Rate (lbm/sec)	0.97	0.93	1.24

[1] Elevation is based on height above beginning of heated fuel length.

Table 4.7-1

Transfer Station Fabrication Specification Summary

MATERIALS

Steel components shall be of material as specified on the referenced drawings.

WELDING

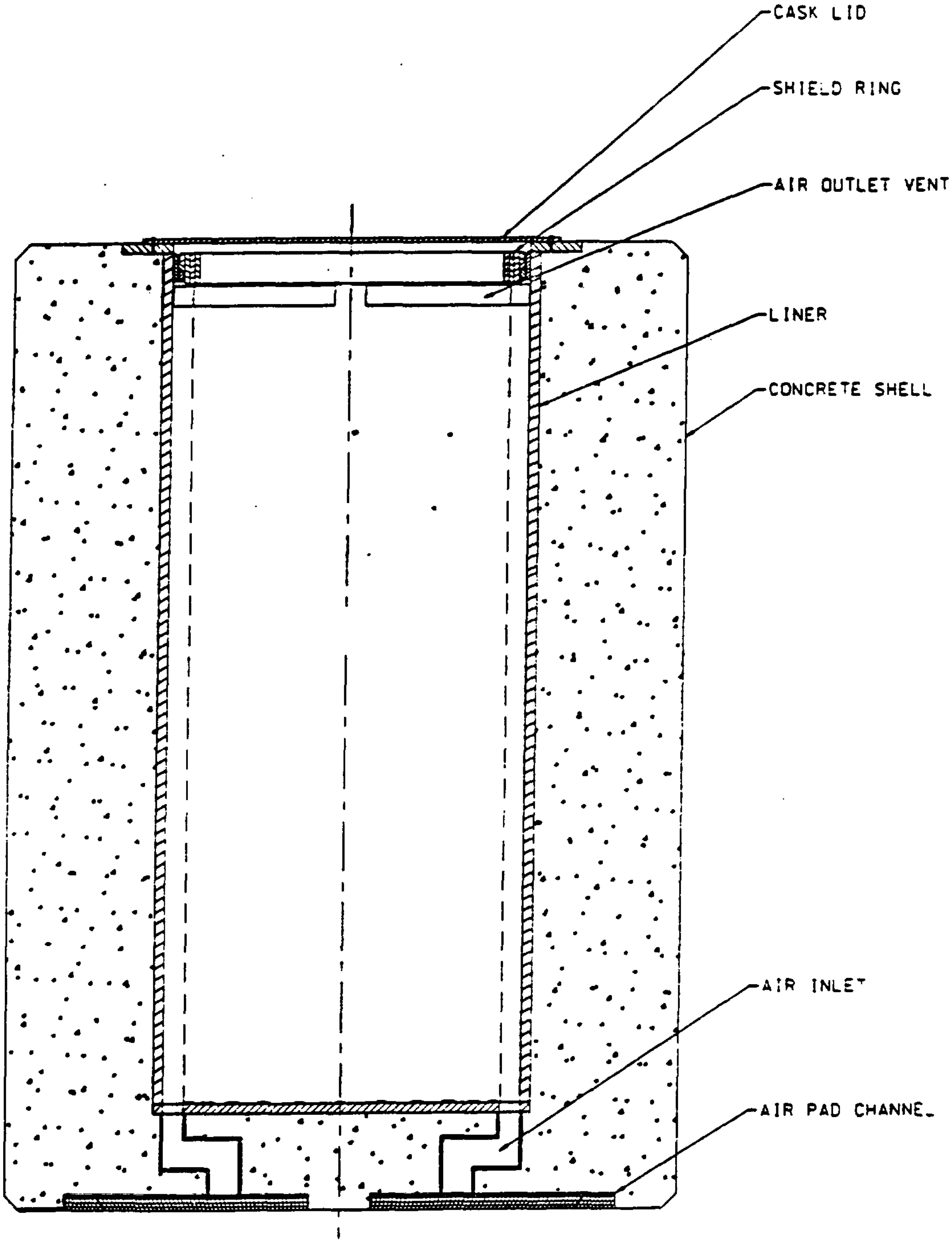
Welds shall be in accordance with the referenced drawings.
Filler metals shall be appropriate AWS D1.1 material.
Welders and welding operators shall be qualified in accordance with AWS D1.1.
Welding procedures shall be written and qualified in accordance with AWS D1.1.
Visual inspection of structural welds shall be performed to the requirements of AWS D1.1.

CONSTRUCTION

Cutting and forming shall be in accordance with AISC Manual of Steel Construction.
Attachment bolts shall be installed and torqued in accordance with referenced drawings.

Quality Assurance

The Transfer Station shall be constructed under a quality assurance program that meets the applicable requirements of 10 CFR 72 Subpart G.



TROJAN ISFSI
SAFETY ANALYSIS REPORT

FIGURE 4.2-4
CONCRETE CASK

5.0 OPERATIONS

5.1 GENERAL DESCRIPTION

The methods and sequences described below define the operational controls which personnel performing spent fuel loading and storage activities will implement to assure that operations utilize the passive safety features of the Trojan ISFSI design described in Chapter 4. Fuel loading and basket sealing operations (including non-destructive examination and pressure testing) will be performed within the Fuel Building in order to utilize the existing systems and equipment for heavy lifts, radiation monitoring and controls, decontamination and any necessary auxiliary support (i.e., electrical, crane, service air, etc.). Fuel handling and cask loading operations in the Fuel Building will be performed in accordance with Portland General Electric Company's 10 CFR 50 license for the Trojan Plant. Storage at the ISFSI will be subject to the requirements of the ISFSI license issued in accordance with 10 CFR 72. Once the loaded storage cask is placed on air pallets in the Fuel Building Bay and moved to the ISFSI concrete slab area, operational activities are essentially limited to monitoring proper decay heat removal.

5.1.1 OPERATION DESCRIPTION

The following sections describe the spent fuel handling, basket sealing, and cask loading activities relevant to the operation of the Trojan ISFSI. As previously described in Chapter 3, the Trojan ISFSI will contain intact and failed spent nuclear fuel assemblies, fuel debris, and GTCC waste. The PWR Basket is vertically loaded with fuel assemblies and/or special cannisters holding failed fuel or fuel debris. The GTCC Basket is loaded with a special cannister containing segmented GTCC waste. Section 5.1.1.1 describes the operational controls for loading the individual cannisters. Section 5.1.1.2 describes operational controls for loading the individual basket.

Specific procedures will define and control classification criteria, loading sequence and individual basket/cask inventory. Fuel/debris/GTCC waste will be visually inspected as it is loaded to verify that each assembly/item conforms to the established classification criteria. As a minimum, item identification and/or serial numbers will be verified and recorded. Fuel loading operations will be videotaped to visually record fuel assembly serial numbers and to provide an independent record of loaded inventory. Fuel will be examined to verify that pellets are structurally contained within the cladding or it will be placed in a Failed Fuel Can. The Debris Cannister will be inspected to verify that it is sealed. Additional procedures will control placement and use of impact limiters, allowable travel path inside the Fuel Building, and limit lifting heights to assure compliance with bounding analysis.

5.1.1.1 Failed Fuel, GTCC Waste, and Fuel Debris Can Loading

Special containers are used to segregate failed fuel, fuel debris, and GTCC waste within the confines of the PWR Basket or GTCC Basket. The individualized containers provide containment properties by constraining the material to fixed storage locations which maintains the assumptions in the criticality analysis and heat transfer modeling.

Failed fuel is contained in special cans designed to fit in one of four oversized peripheral storage sleeves of the basket. Failed or suspect fuel that cannot structurally contain pellets within the cladding will be placed in a Failed Fuel Can. Because the cans are open to the internal basket atmosphere, they are vacuum dried and backfilled at the same time as other basket contents.

Fuel debris particles are contained within a specially designed and fabricated canister which has appropriate seals and fittings to be independently dried and helium backfilled before being loaded in a basket. The length of the Fuel Debris Can is approximately 160 inches which necessitates placement on a spacer block in the fuel cell of the spent fuel pool to allow operation of the shutoff valves on quick disconnect fittings and to allow for disconnection of the vacuum drying and backfill hoses. Loading the fuel debris canister underwater provides adequate radiation shielding. The Fuel Debris Can is tested in accordance with the requirements specified in Section 9.2 before being used to store fuel debris.

The screw-in plug that seals the Fuel Debris Can is removed and the fuel pellets, fuel pellet fragments and small particle waste are placed inside. The screw-in plug is replaced, torqued, and the canister moved to a fuel cell location and stored until cask loading. Before placement in a basket, vacuum drying and backfill hoses are connected to the canister which is then vacuum dried and backfilled with helium. The same basic process as previously described for basket vacuum/backfill operations is implemented for the Fuel Debris Can. The water from the canister is pumped back into the spent fuel pool until pump suction is lost. Drying is initiated by compressed air, nitrogen, or helium being blown through the canister (maximum pressure will be controlled to 7.3 psig) and held until all water is removed. Using the vacuum pump, multiple pump downs are performed as required to achieve a stable vacuum pressure in the basket of 3 mm Hg for at least 30 minutes. The can is backfilled with helium and pressurized to atmospheric pressure (approximately 14.7 psia). All connections to the can are then disconnected. The can is placed in one of the peripheral storage sleeves of the basket.

The GTCC Can effectively contains GTCC waste currently stored at the Trojan Nuclear Plant. The cans are filled and placed in the GTCC Basket utilizing a 28 slot alignment grating which is removed after the GTCC Basket is filled. The cans are open to the GTCC Basket atmosphere to allow drying.

5.1.1.2 Basket Loading and Sealing Operations

This section describes the sequence of operations and controls necessary to load, seal, and test a PWR Basket or GTCC Basket in the Fuel Building and to control transfer operations to the ISFSI storage pad. The major components described in Chapter 4 are further defined with design and operating characteristics. Test and/or inspection methods demonstrate compliance with design requirements.

The basket and Transfer Cask are brought into the Fuel Building through the crane bay door. After examination and any needed cleaning, the Transfer Cask is moved by use of the Fuel Building overhead crane (independent dual hook design) and Transfer Cask Lifting Yoke to the cask wash pit area. There, a protective bottom cover (e.g., plastic sheeting, plexiglas, plywood, etc.) is installed on the Transfer Cask lower hydraulic door carrier rails, which will prevent possible contamination from the cask loading pit floor from being imbedded in the cask. The basket is then moved by the same crane and placed into the Transfer Cask. After installation of radiation shielding shims in the gap between the Transfer Cask and basket, a basket retaining ring is bolted to the Transfer Cask wall. The basket is then filled with borated water. The water is filtered, if necessary, to reduce the potential for contamination on the exterior of the basket. This filling may be done in the cask wash pit area or at the cask loading pit before submergence.

The Transfer Cask (with basket) is then moved by the Fuel Building overhead crane and suspended over the cask loading pit immediately adjacent to the spent fuel pool. Borated water is continuously flushed through the basket/Transfer Cask gap to minimize unnecessary contamination of the basket external surface while the Transfer Cask is in the cask loading pit. After the Transfer Cask is lowered to the pool bottom, the specified basket contents are loaded. Operations will be conducted in accordance with approved Trojan Nuclear Plant fuel handling procedures.

The basket shield lid is placed into the basket while in the cask load pit. The loaded Transfer Cask is lifted from the pool, the basket/Transfer Cask gap drained, standing water above the shield lid is removed, the cask is washed on the exterior to remove potential contamination, and the protective bottom cover is removed. The Transfer Cask is returned by use of the Fuel Building overhead crane to the cask wash pit area.

Decontamination of the Transfer Cask and shield lid welding may begin as soon as the Transfer Cask is in the cask wash pit. The water level is lowered in the basket by approximately 75 gallons in order to form an air gap below the bottom of the shield lid and the water surface. This ensures that the shield lid weld is not affected by percolation. The root and cover weld passes are dye penetrant checked. The basket is refilled with borated water and hydrostatically tested to 7.3 psig which exceeds 1.25 times the normal operating pressure. This pressure must be held for 10 minutes with zero leakage. The skid mounted vacuum drying system is the preferred equipment for performing gas/liquid filling or evacuation activities. After test acceptance, the structural lid is installed and welded to the basket shell and the previously installed shield lid (through the valve access ports). The radiation shielding shims are removed individually from the top of the basket area to allow for completion of decontamination activities. The exterior of the basket will be checked for loose surface contamination (to the extent possible because of its inaccessibility while in the Transfer Cask) to determine if decontamination of the basket is required. Contamination limits are described in Chapter 7.

Decay heat could eventually cause boiling in the basket after it is removed from the cask loading pit. As a precaution, lid sealing, hydrostatic testing and draining must be completed within 36 hours (beginning when the basket top is lifted from the cask loading pit) or the basket must be returned to the cask loading pit and submerged until satisfactorily cooled.

Evacuation of the basket is initiated by pumping the liquid contents back into the Spent Fuel Pool or a suitable holding tank. The structural lid weld is completed and the root and cover weld pass are dye penetrant checked. To aid in removing residual moisture, dry service air is blown through the basket via the basket drain line (maximum 25 psig pressure) and out the vent line for a minimum of 15 minutes and until no water is visible coming from the vent line. The outlet for the vent line will be connected to a suitable filtration system to minimize the possibility of gaseous or particulate airborne contamination.

The vacuum drying system is used to perform multiple pump downs to achieve a stable internal basket vacuum pressure of 3 mm Hg for a minimum of 30 minutes. The basket is then flushed with helium and the evacuation/vacuum process is repeated. The vacuum drying time will be administratively controlled to minimize the strain that fuel cladding will be subjected to during this operation.

The sealed basket is backfilled with 99% pure helium and pressurized to 7.3 psig. The vacuum drying system gauges are used to regulate internal pressure. Leak tightness is verified by use of a helium sniffer. The pressure is then released back to atmospheric pressure (approximately 14.7 psia). Procedures for leak testing will be in conformance with ANSI N14.5.

The vacuum drying system attachment couplings are disconnected and the two valve covers fitting into the structural lid are welded in place and dye penetrant checked. The Transfer Cask containing the sealed basket is transported by the Fuel Building overhead crane back to the area inside the crane bay door where a Concrete Cask has been placed and prepared for acceptance of the basket. Ceramic tiles in the bottom of the Concrete Cask prevent the stainless steel basket from resting directly upon the carbon steel liner of the Concrete Cask.

Plastic sheeting is placed on top of the Concrete Cask walls to prevent contamination from the bottom of the Transfer Cask. The Transfer Cask is placed on top of the Concrete Cask and correctly positioned by the use of one inch diameter holes located on each side of the Transfer Cask. After installation of the Transfer Cask hydraulic door system and basket lifting ring and slings, the basket is slightly elevated to eliminate weight on the Transfer Cask bottom doors. The bottom doors are opened, and the basket is lowered into the Concrete Cask. When the basket is firmly resting on the ceramic tiles at the bottom of the Concrete Cask, the basket lifting rings and slings are removed with the aid of an extension device and the Transfer Cask bottom doors are closed. After removal of the hydraulic system, the Transfer Cask is lifted from the Concrete Cask and transported back to the cask wash pit area where the interior of the Transfer Cask is checked for loose surface contamination to provide a second check of the surface contamination levels on the exterior of the basket that was just removed from the Transfer Cask. If contamination levels exceed the limits established in Chapter 7, then the need to decontaminate the basket previously placed in a Concrete Cask will be evaluated. The shield ring is installed on top of the Concrete Cask and the Concrete Cask cover plate is bolted into position. A tamper indicating wire is threaded through two of the cover bolts.

The Concrete Cask exterior will be surveyed for contamination and radiation levels will be measured before transporting and placement on the ISFSI pad.

5.1.1.3 Transfer to Storage Area Operations

The air pad system described in Section 5.2.1.1.6 is inserted under the cask in the openings provided and inflated by a standard service air compressor. The use of the air pad system results in blocking of the air inlet flow path. Administrative procedures will be implemented to control this activity. The operating height after air pad inflation is approximately 3 inches. The cask is moved by forklift, tractor, or other adequate vehicle to the storage area. The loaded Concrete Casks are placed on the ISFSI pad on approximately 15 ft. center to center spacings and/or in a configuration that supports the assumptions made in calculating the direct radiation rates of the array. Air is released from the pads so that the cask is resting on the concrete storage pad surface. A startup test to confirm proper operation of the storage system will be performed once the Concrete Cask is placed on the pad.

5.1.1.4 Maintenance Operations

The Trojan ISFSI is designed to be a passive system and does not require specified maintenance tasks. Recommended inspection and surveillance activities are required by the Technical Specifications.

5.1.1.5 Off-Normal Event Recovery Operations

The analysis of normal and off-normal events and accident design events identified by ANSI/ANS 57.9, as applicable to the Trojan ISFSI, are presented in Chapter 8. Each postulated event analyzed addresses both event detection and required corrective actions. Additionally, should an off-normal event occur, an inspection for possible damage will be completed within 24 hours. An engineering evaluation will also be required to establish that a component may safely continue to perform the required function.

Although shown by analysis not to be a credible design event, a method for recovery from a leaking basket has been developed. Recovery can be accomplished at the storage site without the benefit of a spent fuel pool by implementing one of the following actions. Method a) is limited to use where the leaking weldment is accessible on top of the Concrete Cask. If the leak is not in the top weld or the leak cannot be repaired, then method b) is used.

a) After establishing the necessary radiation shielding, remove the Concrete Cask lid and locate the leaking weld area by use of a helium sniffer. The defective weld area is removed, verified by appropriate NDE, and rewelded. The finished weld area is dye penetrant checked and the basket is then purged and refilled with helium as described in the original basket loading sequence. Access to the filling and venting valves must be gained by removing the welded valve covers in the structural lid. These valve covers will be reinstalled (or replaced) per the original requirements for welding and testing.

b) Move the cask with defective basket and another Concrete Cask into adjacent positions within the Transfer Station. After establishing the necessary radiation shielding, remove the defective basket's cask lid, install the basket lifting rings and slings, and remove the cask shield ring. Place the Transfer Cask on top of the Concrete Cask using a portable crane. Install the Transfer Station sliding collar and side members. Install the Transfer Cask door hydraulic system, and open the bottom doors. Lift the defective basket into the Transfer Cask and close the bottom doors. Relocate the loaded Transfer Cask over the adjacent Concrete Cask. Insert a Basket Overpack into the now empty Concrete Cask and install the Basket Overpack shield ring. Relocate the Transfer Cask over the Concrete Cask containing the Basket Overpack. Lift the defective basket slightly to remove weight from the bottom doors, open the bottom doors, and lower the defective basket into the Basket Overpack. Close the bottom doors and remove the Transfer Cask. Remove the valve covers from the defective basket structural lid and open the valves. Weld the Basket Overpack structural lid to the shell. Install the quick connect and perform vacuum drying and helium backfill/ leak testing per original loading requirements identified in Section 5.1.1.2. Weld the disconnect cover in place and perform dye penetrant examination. Reinstall the Concrete Cask shield ring and lid and rethread the tamper wire. The Concrete Cask containing the inserted Basket Overpack is utilized for continued storage.

5.1.1.6 Off-Site Transfer Operations

A 10CFR71 licensed shipping cask will be available to transport the baskets off-site to a DOE high level waste repository or interim storage facility in the future.

Transfer operations will utilize the Transfer Station described in Chapter 4 for basket removal from the Concrete Cask and reinstallation in a 10 CFR 71 approved shipping cask.

Any basket sealed in a Basket Overpack must be removed from the Basket Overpack before being placed in a shipping cask. The procedural methods outlined in Section 5.1.1.5 b) will be utilized to position the Concrete Cask containing a Basket Overpack in the Transfer Station. An empty shipping cask will be positioned in the adjacent space of the Transfer Station. The Basket Overpack lid will be removed and the Transfer Cask utilized to relocate the basket to the shipping cask.

5.1.2 FLOWCHARTS

Figure 5.1-1 contains an operation sequence flowchart of basket loading, sealing, testing, and storage operations including anticipated task completion times.

Figure 5.1-2 contains a flowchart detailing actions and estimated task completion times for a leaking basket recovery operation utilizing the Basket Overpack.

Figure 5.1-3 contains a flowchart detailing the actions and estimated task completion times for transferring a basket to a shipping cask for off-site transport.

The sequence of operations is the basis for the collective dose assessment discussed in detail in Section 7.4.

5.1.3 IDENTIFICATION OF SUBJECTS FOR SAFETY ANALYSIS

5.1.3.1 Criticality Prevention

Specific techniques or operational procedures are not relied upon to assure criticality prevention. Geometrical spacing of the 24 fuel assemblies maintains subcritical conditions during dry storage conditions. While control assemblies may be stored integral with the fuel assemblies, they are not credited for criticality control.

5.1.3.2 Chemical Safety

The Trojan ISFSI Concrete Cask system does not employ any hazardous chemicals that would require special precautions or procedures.

5.1.3.3 Operation Shutdown Modes

Because the Trojan ISFSI Concrete Cask system relies on natural air circulation, it does not have any shutdown modes.

5.1.3.4 Instrumentation

The Trojan ISFSI is passive by design and requires no instrumentation to operate. The following chart lists the measuring and test equipment necessary to monitor the Trojan ISFSI for compliance to design requirements. Measuring and test equipment is not classified as important to safety. The instruments are commercially available, standard products and will be calibrated in accordance with Quality Assurance requirements.

Measuring and Test Equipment

Instrument	Function
1. Hand-held survey equipment (Gamma, neutron, and surface contamination)	Measures dose rates on Concrete Cask surface and contamination levels.
2. Pressure and Vacuum Gauges	Measures helium, air, water, and vacuum pressures inside the basket.
3. Helium Leak Detector	Detects the presence of helium.
4. Temperature monitoring devices	Measures temperatures

5.1.3.5 Maintenance Techniques

The Trojan ISFSI does not require specified maintenance tasks. Recommended inspection and surveillance activities are described in the Technical Specifications. The measuring and test equipment identified in Section 5.1.3.4 will be maintained in accordance with the original equipment manufacturers recommendations.

5.1.3.6 Heavy Loads Procedures

The handling of heavy loads will be addressed in a NUREG 0612 evaluation and in heavy loads procedures. Tests and certifications (including cranes, hooks, slings, trunnions, straps, cable, evaluation and procedures will assure that the Trojan Fuel Building can withstand the loads from postulated drops and that the basket design accelerations are not exceeded. Impact limiters will be used to mitigate the effects of a drop accident. Chapter 8, Accident Analysis, also addresses drops at the ISFSI pad during handling operations.

5.2 SPENT FUEL HANDLING OPERATIONS

5.2.1 SPENT FUEL HANDLING AND TRANSFER

Spent fuel handling and transfer operations including removal from the fuel pool, basket loading and sealing, transfer to the ISFSI pad and eventual transfer to an off-site location are described in Sections 5.1.1.1 through 5.1.1.6. Chapter 4 provides a description of the components and the applicable design basis utilized for safely maintaining the fuel/debris/GTCC waste in a safe storage configuration. Specific equipment function is described in Sections 5.2.1.1.1 through 5.2.1.1.6.

5.2.1.1 Functional Description

The Transfer Cask, Lifting Yoke, vacuum drying system, welding system, hydraulic system, and air pad system are necessary to facilitate basket loading, storage and eventual off-site shipping activities.

5.2.1.1.1 Transfer Cask

The Transfer Cask is a special lifting device designed and fabricated to the requirements of NUREG 0612 and ANSI N14.6 to be used during transfer operations with the Trojan Fuel Building overhead crane or with a crane at the ISFSI storage pad.

The Transfer Cask consists of a cylinder with a steel-lead-neutron shielding-steel sandwich wall. The thick-walled cylinder reduces dose rates to an acceptable level as shown in Chapter 7. The cask lid extends over the basket to prevent it from being inadvertently lifted out of the Transfer Cask during basket transfer operations. At the bottom of the Transfer Cask are doors that slide in rails along each side of the cask that open upon hydraulic actuation to allow lowering of the basket into the storage or transportation cask. Two steel pins are used to prevent accidental opening of the doors.

lowering of the basket into the storage or transportation cask. Two steel pins are used to prevent accidental opening of the doors.

5.2.1.1.2 Cask Lifting Yoke

The Transfer Cask Lifting Yoke is used for Transfer Cask handling operations both within the Fuel Building and at the ISFSI pad. It is designed to interface with the Trojan crane hooks and is fabricated from high strength carbon steel.

5.2.1.1.3 Vacuum Drying System

Following fuel loading and welding of the shield and structural lids, a skid-mounted vacuum drying system is used to remove the water from the basket, dry the interior, and backfill it with helium. The vacuum drying system is designed to evacuate the basket to < 3 mm Hg in an iterative fashion. During evacuation, the decay heat from the fuel helps to remove any residual moisture from the basket. Valves and gauges are located on the vacuum skid and basket top to monitor the performance of the system.

5.2.1.1.4 Semi-automatic Welding System

A semi-automatic welding system is the preferred equipment used for Basket Overpack closure based on design, operation, and ALARA considerations. It includes a customized welding drive carriage and adapter for mounting the system on top of the Transfer Cask.

5.2.1.1.5 Hydraulic System for Operation of Transfer Cask Doors

Two hydraulic cylinders are bolted to the outer Transfer Cask wall after the Transfer Cask is placed in the position to perform transfer in or out of the Concrete Cask. The cylinders open the bottom doors of the Transfer Cask to allow basket transfer and closure of the doors after completion of the operation.

5.2.1.1.6 Air Pad System

The air pad system will be used to transport a loaded Concrete Cask from the Fuel Building to storage location at the ISFSI pad and to the Transfer Station for off-site shipping. The air pads are commercially available lifting devices which, when inflated, lift the cask a few inches with high pressure air. The system uses four standard air lifting pads and an air compressor. When energized, the pads and the cask float on a cushion of air and can be moved about by a forklift or truck. Air pads minimize the need for handling room around the cask, thus minimizing the size of the storage pad.

5.2.1.2 Safety Features

Spent fuel handling and transfer operations utilize both component designed features and administrative controls to assure that required actions are safely accomplished. The basket is sheltered in the cavity of the Transfer Cask during spent fuel handling. Continuous flushing of the basket/Transfer Cask gap reduces potential contamination from the cask loading pit. The Transfer Cask cover ring is used to prevent accidental lifting of the basket out of the Transfer Cask. Additional shielding is placed over the basket/Transfer Cask gap and temporary shielding is used to minimize worker exposure. The basket storage sleeves are designed to withstand a postulated drop accident in the Transfer Cask without damaging stored fuel. The Transfer Cask has steel pins in the bottom doors to prevent them from being inadvertently opened.

5.2.2 SPENT FUEL STORAGE

The major components of the Trojan ISFSI system used for spent fuel storage are the basket and Concrete Cask. Design characteristics for each are described in Chapter 4. Operational performance is described in Sections 5.1.1.1 and 5.1.1.2. Removal of stored spent nuclear fuel from the Trojan ISFSI for off-site shipment is described in Section 5.1.1.6.

5.2.2.1 Inspection and Surveillance Program

The inspections and surveillances required for the Trojan ISFSI are contained in the Technical Specifications.

5.2.2.2 Safety Features

Spent fuel storage at the Trojan ISFSI utilizes the inherent safety features of a passive dry cask design as well as additional administrative controls. Fuel assemblies with higher radiological source terms will be loaded toward the center of each basket and lower radiological source term fuel will be loaded near the outer periphery in order to minimize dose rates. Fuel assemblies/debris will not be sealed in the same basket with GTCC waste. In storage, the basket is sheltered in the cavity of the Concrete Cask that reduces the surface dose to well within allowable limits as demonstrated in Chapter 7. Additional shielding, in the form of a ring, is placed over the gap between the Concrete Cask and basket after the transfer is complete. Although no credible event can overturn the Concrete Cask, the basket and Concrete Cask are designed to withstand a postulated tipover without damaging stored fuel or breaching the confinement boundary. Other safety features (as related to off-normal events) are discussed in Chapter 8, Accident Analyses.

5.3 OTHER OPERATING SYSTEMS

5.3.1 SYSTEM OPERATIONS

The Trojan ISFSI storage system is passive and does not require any systems for its operation once it is placed into storage. The only equipment required for this storage system, besides the storage structures, is for fuel loading and movement of the casks to the ISFSI as described in Section 5.2.1.

5.3.2 COMPONENT/EQUIPMENT SPARES

The Trojan ISFSI system is designed to withstand postulated design basis events as described in Chapter 8, therefore no equipment spares are needed.

5.4 SUPPORT SYSTEM OPERATION

5.4.1 INSTRUMENTATION AND CONTROLS

The operation of the Trojan ISFSI is passive and self-contained and, therefore, does not need any control systems. Temperature monitoring devices are used for measuring cask air outlet temperature.

5.4.2 SPARES

Other than the temperature monitoring devices identified in Section 5.4.1, no instrumentation or control systems are required for operation of the Trojan ISFSI Since the devices are readily available commercial components, equipment spares are not required.

5.5 CONTROL ROOM AND CONTROL AREAS

No control room or control areas are needed for the Trojan ISFSI.

5.6 ANALYTICAL SAMPLING

No sampling or analysis is necessary to ensure safe operation of the Trojan ISFSI.

6.0 SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT

6.1 ONSITE WASTE SOURCES

Fuel loading and closure of the basket will be accomplished under the existing Trojan Nuclear Plant 10 CFR 50 license. The radioactive waste generated during fuel loading and closure of each basket would consist mainly of potentially contaminated water used for decontamination of the transfer cask and basket and a small amount of solid waste, e.g., absorbent rags, used for decontamination. These wastes would be processed using existing Trojan Plant procedures and systems. Gaseous waste is not anticipated, but the system used to pump down and vacuum dry the basket will be designed to filter or capture gaseous waste as required.

Normal operation of the Trojan ISFSI will not generate radioactive waste. The spent nuclear fuel and GTCC waste are sealed inside a leak-tight stainless steel basket and the normal surveillance and inspection activities do not affect the confinement capability of the basket.

6.2 OFFGAS TREATMENT AND VENTILATION

During the fuel loading and basket closure process, the basket that contains the spent nuclear fuel is vacuum dried and backfilled with helium while located inside the Trojan Fuel Building. Any radioactive gas that is drawn off the basket during the vacuum drying process is passed through a system designed to filter or capture gaseous waste as required.

Gaseous radioactive waste is not generated by normal operation of the Trojan ISFSI. Any radioactive gases that are released from the spent nuclear fuel during the storage period will remain inside the leak-tight basket. The basket is designed to remain sealed while stored at the ISFSI, but if opening the basket is required during the storage period, a venting system would be used that would filter or capture radioactive gases as required.

6.3 LIQUID WASTE TREATMENT AND RETENTION

During the fuel loading and basket closure process, potentially contaminated liquid will be generated while decontaminating the transfer cask and basket. This potentially contaminated liquid will be collected and processed by existing Trojan Nuclear Plant procedures and systems under the 10 CFR 50 license.

Liquid radioactive waste is not generated by normal operation of the Trojan ISFSI. The basket is vacuum dried during the closure process. Therefore, there is no liquid in the leak-tight basket during the storage period.

6.4 SOLID WASTES

During the fuel loading and basket closure process, a small amount of solid low level waste, e.g., absorbent rags, will be generated while decontaminating the transfer cask and basket. This solid waste will be collected and processed using existing Trojan Nuclear Plant procedures under the 10 CFR 50 license. Solid waste is not anticipated during the basket pump down and vacuum drying process, but the system used to pump down and vacuum dry the basket will be designed to filter or capture such wastes as required.

Solid radioactive waste is not generated by normal operation of the Trojan ISFSI because any solid waste will remain inside the leak-tight basket during the storage period. No contamination is anticipated at the ISFSI, but periodic surveys are performed during normal operation to confirm that there is no contamination. In the unlikely event that contamination is discovered, a small amount of radioactive waste, e.g., swipes and absorbent rags, would be collected and processed as low level waste.

6.5 RADIOLOGICAL IMPACT OF NORMAL OPERATIONS - SUMMARY

Gaseous, liquid or solid radioactive waste are not generated during normal operation of the Trojan ISFSI. The radioactive waste generated during the loading and closure of the baskets is collected and processed under the 10 CFR 50 license using existing Trojan Nuclear Plant procedures and systems.

7.0 RADIATION PROTECTION

7.1 ENSURING THAT OCCUPATION RADIATION EXPOSURES ARE AS LOW AS IS REASONABLY ACHIEVABLE (ALARA)

7.1.1 POLICY CONSIDERATIONS

The Radiation Protection Program used for operating the ISFSI implements the regulatory requirements of 10 CFR 20, "Standards for Protection Against Radiation," 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," and Oregon Administrative Rule (OAR) 345-26-390, through approved procedures.

The primary objective of the Radiation Protection Program is to maintain exposures to workers, visitors, and the general public As Low As Reasonably Achievable (ALARA). The dry storage system that will be used at the Trojan ISFSI has been designed with ALARA considered for the operation, inspection, maintenance, and repair of the system. PGE provides or will contract qualified staff, facilities, and equipment to ensure that exposures are ALARA during operation of the ISFSI. The ISFSI will be monitored and evaluated on a routine basis to maintain radiation exposures to unrestricted areas ALARA.

Specific design and operation oriented ALARA considerations are described in the following sections.

7.1.2 DESIGN CONSIDERATIONS

The storage cask system is designed to ensure that occupational radiation exposures are ALARA as defined in 10 CFR 20. As such, special design considerations have been taken to ensure exposure rates are ALARA. These considerations include:

1. Thick walls and lids that contain various shielding materials;

2. Totally passive system that requires minimum maintenance;

3. Multiple pass welds on all seal closures to provide redundant radioactive material containment;

4. Non-planar paths for the air inlets and outlets to minimize radiation streaming; and

5. Spacing of the casks on the concrete pad to provide self-shielding for interior casks.

In addition to the basket for storing spent nuclear fuel, a separate basket has been designed to store Greater than Class C (GTCC) Waste. Due to the GTCC waste high gamma source strength, the GTCC basket design incorporates additional features to ensure that the GTCC concrete cask external dose rates are similar to those of the casks containing spent nuclear fuel. These additional features are:

1. An extra thickness of steel inside the basket shell;

2. An extra steel bottom plate;

3. A steel and lead shield lid (instead of just steel); and

4. An extra steel shielding/support plate.

Detailed descriptions of the storage cask system components are included in Chapter 4.

7.1.3 OPERATIONAL CONSIDERATIONS

ISFSI operational details including basket loading, closure, transfer, storage, and off-site shipping are included in Chapter 5. Loading and transfer operations have been determined following ALARA guidelines. ISFSI personnel will follow site procedures consistent with Regulatory Guide 8.8 and Regulatory Guide 8.10. Personnel radiation exposure during handling and closure of the basket is minimized by the following steps.

1. Fuel loading procedures that follow accepted practice and build on existing experience;

2. Loading spent nuclear fuel in the basket within the controlled environment of the Fuel Building to prevent the spread of contamination;

3. Loading the most radioactive fuel in interior basket positions;

4. Injecting filtered, borated water into the annulus between the transfer cask and basket to minimize contamination of the basket external surface;

5. Placing the shielding lids on the basket while the transfer cask and basket remain in the cask load pit;

6. Decontaminating the exterior of the transfer cask and welding the basket lid while the basket is still filled with water;

7. Draining the basket while still housed in the transfer cask;

8. Using portable shielding as necessary;

9. Using the shielded transfer cask that is remotely operated to transfer the basket to the storage cask;

10. Placing a shielding ring over the annular gap between the storage cask and basket;

11. Swiping the storage cask exterior for contamination prior to leaving the Fuel Building;

12. Storing higher dose rate storage casks toward center of ISFSI pad to provide additional shielding; and

13. Using ALARA pre-job briefings prior to fuel movement and cask loading sequence.

7.2 RADIATION SOURCES

7.2.1 CHARACTERIZATION OF SOURCES

The general criteria for the radioactive material to be stored in the Trojan ISFSI are provided in Table 3.1-2. Shielding analyses were performed for two design basis cases, 40,000 MWd - 5 year cooled fuel, and 45,000 MWd - 6 year cooled fuel. The entire Trojan spent fuel inventory is bounded by these two cases with respect to burnup level and cooling time (Reference 4). For each case, lower bound initial enrichment levels are assumed, since this will yield the maximum gamma and neutron source terms for fuel of a given burnup level.

Five distinct radiation sources are modeled by the shielding analyses. The active fuel region of the assembly contains two sources, the fuel region gamma source and the fuel region neutron source. In addition, three non-fuel region gamma sources are modeled, the bottom nozzle region gamma source, the gas plenum region gamma source, and the top nozzle region gamma source. Each of these three gamma sources is almost entirely due to Co-60 activity from activated steel. Each of the above sources is described in the following subsections.

The radiation source data used in the shielding analyses are taken from the Office of Civilian Radioactive Waste Management (OCRWM) spent fuel computer database (Reference 6). This database gives radiation source strengths and spectra for spent fuel (on a per MTU of fuel basis) as a function of burnup level, cooling time, and initial enrichment. The database also gives sufficient additional data to calculate the gamma source terms for the non-fuel assembly regions.

7.2.1.1 Fuel Gamma Source

The fuel gamma source is comprised of 163 principle fission product radionuclides, several activation products, and actinide radionuclides present within the UO_2 fuel. The isotopes that are the major contributors to the fuel source term are listed in Table 7.2-1.

The fuel region gamma source is modeled as a homogenous volumetric source which completely fills the basket interior over the axial span that contains active fuel. The gamma source is evenly distributed throughout this defined volume. The fuel region gamma source is based upon 24 complete PWR fuel assemblies, each with the maximum (initial) uranium loading of 0.469 MTU. Using the same term for complete fuel assemblies in the model bounds the source term for partial fuel assemblies (failed fuel) and fuel debris because a complete fuel assembly has a higher source term.

The fuel region gamma sources for each design basis case are listed in Table 7.2-2. Gamma source strengths are shown for both of the bounding burnup and cooling time cases. For each of the two cases, minimum initial enrichment levels are conservatively assumed. An initial enrichment level of 3.02% U-235 is assumed for the 40,000 MWd case, and an initial enrichment of 3.3% U-235 is assumed for the 45,000 MWd case. These are conservative lower bound enrichment levels which bound the entire Trojan spent fuel inventory (i.e., Trojan fuel of a similar burnup level has a higher enrichment level).

The gamma source strength (in gammas/sec-cask) in Table 7.2-2 is given for each gamma energy line. The gamma source strengths from the OCRWM computer database for each energy line are multiplied by the fuel loading of 0.469 and the number of assemblies (24) and are added together to yield the gamma source strengths per cask shown in Table 7.2-2.

The 1.25 MeV gamma source strengths shown in Table 7.2-2 include the additional gamma source from activated control components which may be inserted into the assemblies. The additional gamma source comes from activated control component stainless steel cladding. The OCRWM database gives the active fuel region Co-60 activity (in Ci) per MTU of fuel as a function of burnup, cooling time, and enrichment. It also gives the total amount of cobalt initially present in the assembly per MTU of fuel. The active fuel region cobalt activation level (Ci of Co-60 per gram of initial cobalt) can be determined from this data. The total amount of cobalt present in the control component is then determined from the total stainless steel mass in the control component cladding and the maximum cobalt concentration in stainless steel. This cobalt mass is multiplied by the cobalt activation level (described above) to yield a total active fuel region Co-60 activity due to the control component cladding. This Co-60 gamma source is then collapsed onto the nearest fuel source gamma energy line (the 1.25 MeV line) and added to the source strength output for the spent fuel by the database. The resulting 1.25 MeV gamma source strength is that shown in Table 7.2-2.

The axial burnup profile present in the active fuel (Reference 8) is modeled in the shielding analyses. This profile, which has a peak to average ratio of 1.1, is shown in Figure 7.2-1. The peak to average ratio of the gamma source strength profile is also 1.1 because the gamma source strength is roughly directly proportional to the fuel burnup level. The gamma shielding models account for this profile by multiplying all dose rates calculated on the storage and transfer cask sides by 1.1. The dose rates calculated on the cask ends are based upon the flat source distribution, and conservatively neglect the effects of the axial burnup profile.

7.2.1.2 Fuel Neutron Source

Neutron sources are based on spontaneous fission sources from various actinides and alpha, neutron (α,n) reactions. The primary neutron source is the spontaneous fission of Cm^{244}. The total neutron source strength for each of the two burnup cases (per MTU of fuel) is taken from the OCRWM computer database. These sources are based on the same initial enrichments assumed for the active fuel region gamma source. These neutron sources are multiplied by the assembly uranium loading of 0.469 MTU and the cask capacity of 24 (assemblies) to yield the per cask total neutron source strengths shown in Table 7.2-3.

The neutron source strengths shown for each neutron energy group were obtained by multiplying the total neutron source strengths by the normalized neutron spectrum shown in the right column of Table 7.2-3. This normalized spectrum is assumed in all of the shielding analyses. The spectrum, which is similar to the spontaneous fission spectrum of Cm^{244}, is taken from the Multipurpose Canister Sub-System Design Procurement Specification (Reference 9).

The neutron source strength has a highly non-linear dependence upon fuel burnup level. Surveys of OCRWM computer database neutron source strength outputs show that the neutron source strength is roughly proportional to the burnup level raised to the 4.2 power. Therefore, this dependence is assumed in these analyses. Based on this dependence, a 10% increase in the fuel burnup level results in a ~50% increase in the neutron source strength. Thus, the neutron source strength for the peak burnup section of the fuel will be ~50% higher than the neutron source strength calculated (by the database) for the assembly average burnup level.

The neutron shielding analyses explicitly modeled the axial neutron source profile which results from the axial burnup profile. This was done by dividing the active fuel region into several small axial subsections, each with a different neutron source strength. These axial subsections are described in Table 7.2-4. The axial span (measured from the bottom of the active fuel region) is shown for each subsection, along with the relative fuel burnup level and the relative neutron source strength. These values are relative to the assembly average burnup level and the neutron source strength calculated for the assembly average burnup level.

7.2.1.3 Non-Fuel Region Gamma Sources

The gamma sources for the three non-fuel assembly regions (the bottom nozzle region, the gas plenum region, and the top nozzle region) are almost entirely due to Co-60 in activated metal components. The Co-60 activity for each of these non-fuel assembly regions can be determined, as a function of assembly burnup level, cooling time, and initial enrichment, from data available in the OCRWM spent fuel computer database (Reference 4).

As discussed in Section 7.2.1.1, the OCRWM database gives the active fuel region Co-60 activity (in Ci) and the total fuel region initial cobalt inventory (in grams) per MTU of fuel. This allows the core region cobalt activation level (Ci Co-60 per gram initial cobalt) to be determined as a function of burnup level and cooling time. During reactor operation, neutron flux levels in the non-fuel regions of the assembly (above and below the core) are much lower than those present in the core region. Thus, the Co-60 activation level is expected to be much lower for the non-fuel regions. Cobalt activation level adjustment factors have been determined for each of the three non-fuel assembly regions (Reference 10).

The OCRWM database gives the mass of each metal type present in each assembly non-fuel region. Multiplying these metal masses by the maximum cobalt concentrations for each metal type and summing the results will yield the total cobalt inventory (in grams) for each of the non-fuel assembly regions. The cobalt inventory for each non-fuel region is multiplied by the adjusted cobalt activation factor to yield the Co-60 activity for that region.

These per assembly Co-60 activities are multiplied by 24 to yield cask total Co-60 activities for each non-fuel region of the basket. These Co-60 activities are then converted into 1.173 MeV and 1.333 MeV gamma source strengths (a Co-60 decay emits one 1.173 MeV gamma and one 1.333 MeV gamma). These total (per cask) non-fuel region gamma source strengths are shown in Table 7.2-5 for each of the three non-fuel regions and for each of the two fuel burnup cases. In the shielding models, each of these non-fuel region gamma sources is evenly distributed throughout the corresponding axial subsection of the basket interior.

7.2.1.4 Greater Than Class C Waste

GTCC waste characterization is presented in detail in WMG-9418, "Preliminary Characterization, Trojan Reactor Internals". GTCC waste consists of activated core components, mainly segmented reactor internals, with an estimated total of 49,000 lbs. In WMG-9418, "Preliminary Characterization, Trojan Reactor Internals", the activated curie contents of the core baffle and baffle former plates are the most limiting. The GTCC waste gamma source activities are listed in Table 7.2-6.

7.2.1.5 Fuel Debris

The PGE fuel debris consists of individual fuel pellets and fragments from damaged fuel rods. For the shielding analysis, fuel debris source terms are conservatively assumed to be the same as for intact fuel. This assumption is conservative because the fuel debris will be stored in fuel debris cans, separate from intact fuel, and the total quantity of fuel debris is only a few kilograms, as compared to an intact fuel assembly with several hundred kilograms of fuel material.

7.2.1.6 Non-Fuel Bearing Components

In addition to failed fuel, the failed fuel cans will also be used to store fuel assembly hardware, non-fuel bearing components, and one fuel skeleton. These components are made of 304 stainless steel, zirconium IV, and Inconel. The source terms from these additional components were not independently considered in the shielding calculations, but the fuel source terms would bound this additional waste.

7.2.2 AIRBORNE RADIOACTIVE MATERIAL SOURCES

Loading of spent nuclear fuel and other wastes into the basket is carried out under water in the Spent Fuel Pool Cask Loading Pit which prevents the spread of contamination. The baskets are dried and sealed within the controlled environment of the Fuel Building. The gaseous waste from the baskets will be passed through a local HEPA filter.

Once the basket is dried and seal welded, there are no credible off normal events or accidents that will cause breaching of the basket and subsequent release of airborne radioactivity. Therefore, no airborne releases to the environment from the spent nuclear fuel assemblies or GTCC waste are expected to occur during loading and handling operations.

During normal operation of the ISFSI, the only potential source of airborne radioactivity is from surface contamination on the basket exterior, which would be deposited there from the Spent Fuel Pool water. As discussed in Chapter 5, filtered, borated water is injected into the transfer cask/basket annulus to prevent contamination of the outside surface of the basket when it is submerged in the cask loading pit and the transfer cask is washed down after being removed from the Spent Fuel Pool to remove potential contamination. The exterior of the basket will be checked for loose surface contamination while the basket is in the transfer cask to the extent possible because the basket surface is not readily accessible while the basket is in the transfer cask. As a second check, the interior of the transfer cask will be checked for loose surface contamination after the basket is removed because the interior surface of the transfer cask would be representative of the loose surface contamination on the exterior surface of the basket that was just inside the transfer cask. A limit of 10^{-4} µCi /cm^2 beta-gamma and 10^{-5} µCi /cm^2 alpha will be used and if loose surface contamination is above this limit, the need to decontaminate the basket will be evaluated. These limits were used in an analysis of a release of radioactive particulates from the basket surface while in storage on the pad. This analysis, which is described in Chapter 8, shows that the consequences of the postulated release are negligible.

Chapter 8 also describes a postulated failure of all fuel pins with a subsequent breach of the basket and a ground level release although no credible mechanism exists to create such a failure. The doses resulting from this postulated failure are within regulatory limits which further demonstrates the conservative design of the storage system when its confinement capabilities are considered.

7.3 RADIATION PROTECTION DESIGN FEATURES

7.3.1 INSTALLATION DESIGN FEATURES

Section 7.1.2 describes the design features of the installation and equipment that ensure exposures to radiation are ALARA. The ISFSI is a passive outdoor storage system. Each storage cask located at the ISFSI has sufficient natural circulation to ensure adequate air cooling of the basket. All radiation sources are confined within the sealed basket which serves as a confinement boundary and shielding. The basket is inside the storage cask which provides further shielding.

Storage cask loading and decontamination will be performed within the Trojan fuel building. Operation of Fuel Building systems for storage cask loading, control of filtered ventilation, and radioactive waste treatment are covered under the existing 10 CFR 50 license.

A detailed description of the ISFSI is included in Chapter 4. The ISFSI layout is described in Chapter 2. Storage cask operational details are included in Chapter 5. The criteria for the design of the installation features and systems are provided in Chapter 3.

Specific shielding design features are described in the next section. These features are consistent with guidance provided in Regulatory Guide 8.8.

Applicable portions of Regulatory Guide 8.8, position C.2 have been used as guidance as follows:

1. Access to the ISFSI is controlled in accordance with 10 CFR 72. Normal access to the ISFSI is through a single access point.

2. Radiation shielding is provided by the basket and storage cask and constitutes the primary method of reducing personnel exposure to radiation.

3. The ISFSI is a passive installation that has no operations to control. Therefore, no process instrumentation or controls are necessary during storage.

4. Airborne contaminants and gaseous radiation sources are confined by the seal-welded basket.

5. No crud is produced by the basket or storage cask.

6. Cask decontamination is performed prior to placement on the storage pad. Once the storage casks are in place, there are no credible mechanisms that could result in contamination of the ISFSI components.

7. Area radiation monitoring instrumentation consists of thermoluminescent devices (TLDs) posted at the perimeter of and in the Controlled Area near the storage casks.

8. No resin or sludge is produced from the basket or storage casks.

7.3.2 SHIELDING

The storage cask system is designed to maintain radiation exposure As Low As Reasonably Achievable (ALARA). The storage cask is designed to provide an average external dose rate (gamma and neutron) of less than 100 mrem/hr on the sides and 200 mrem/hr on top and at the air inlets and outlets. The design maximum dose rate at the top of the basket structural lid is 200 mrem/hr to allow limited personnel access during basket closure operations. This design satisfies the requirements of 10 CFR 72.104, 10 CFR 72.106, and OAR 345-26-390, which establish dose limits for members of the public in unrestricted areas (i.e., at or beyond the Controlled Area Boundary of 325 meters).

7.3.2.1 Radial and Axial Shielding Configurations

The radiation shielding for the stored spent nuclear fuel assemblies and GTCC waste is provided by a variety of shielding materials. Figures 7.3-1 through 7.3-4 show shielding model geometries (described below) for the basket in the storage cask for both the GTCC waste and spent nuclear fuel. Figures 7.3-5 through 7.3-8 show shielding model geometries (described below) for the basket in the transfer cask for both GTCC waste and spent nuclear fuel. Also, the densities for constituent nuclides of all shielding materials used in the calculational models are given in Tables 7.3-1 and 7.3-2.

The fuel basket contains an eight inch thick steel shield lid. This shield lid provides radiation protection for workers engaged in the basket closure and transfer operations as well as the largest majority of the shielding in the top axial direction during storage. Additional shielding in the top axial direction is provided by the 3 inch thick steel structural lid on the basket top and the steel lid on the storage cask. In addition, a steel shield ring immediately above the basket/storage cask inner liner annulus adds protection from radiation streaming up this annulus. Shielding located axially beneath the basket consists of the steel basket bottom, the steel storage cask liner bottom, and a thick section of concrete. These shielding materials are listed in Tables 7.3-1 and 7.3-2.

Radiation shielding in the radial direction during storage is provided primarily by the steel basket shell, followed consecutively by the steel storage cask inner liner and a thick concrete wall. Cooling air penetrations are from the storage cask sides, and contain at least two sharp bends to minimize radiation streaming. The four sets of air inlet and exhaust ducts in the storage cask are fabricated with 0.5" thick steel walls. Additional temporary shielding is installed to attenuate the direct gamma radiation passing through the air outlets during the transfer operations.

In the shielding models, the fuel basket interior is sub-divided into four axial subregions. In each subregion, the complicated basket internal structures are mixed into a single homogenous material that fills the subregion.

In the active fuel region, the homogenous material includes spent fuel material (modeled as pure UO_2), the Zircaloy cladding and guide tube material (modeled as pure zirconium), the carbon steel fuel support sleeve material (modeled as pure iron), and the stainless steel control component cladding material. Several other basket internal components, such as the structural support tubes, which do not necessarily surround the radiation sources, are conservatively neglected in the homogenous material density calculations. The materials from the neutron poison sheets are included in the fuel region homogenous material for neutron calculations only.

The gas plenum region homogenized material includes the Zircaloy cladding material and the iron and nickel materials in the plenum springs (which are the source of the gamma radiation). The top and bottom nozzle region homogenized materials consist of the metal materials which make up the top and bottom nozzles, and which are the source of the gamma radiation. The homogenized materials also include the Zircaloy present in the solid end caps of the fuel rods. Also, the bottom nozzle region material description includes the iron from the carbon steel support sleeves which extend into that region. The fuel support sleeves do not always extend into the top nozzle region, so the top nozzle region homogenized material does not include the support sleeve material.

The homogenized material description for the four axial subregions of the basket interior are shown in Tables 7.3-1 and 7.3-2.

The GTCC basket is similar to the fuel basket with an extra thickness of steel inside the basket shell, addition of a steel bottom plate, a steel and lead shield lid (rather than just steel), and an extra steel shielding/support plate. In the shielding models for the GTCC basket, the entire source term is conservatively considered to be in the lower half of the basket for the bottom axial and radial models and equally distributed throughout the basket for the top axial model.

The transfer cask shielding design reduces the dose from the loaded basket. Shielding at the top is provided by the basket shield and structural lids. Radially, the transfer cask is composed of a steel shell filled with two separate shielding materials. The first is a lead shield designed to attenuate gamma radiation. The second is a strong neutron absorber designed to moderate and absorb the neutrons. Steel heat transfer fins are also present in the neutron shield region. In the shielding models, these steel fins are mixed with the neutron material to create the homogenous material listed in Tables 7.3-1 and 7.3-2. The transfer cask bottom doors are thick steel which reduces the contact dose rate. However, no personnel contact with the bottom doors is required during normal operation.

During off-site transfer operations, the shipping cask lid is removed and a shielded transfer collar installed to provide additional shielding when the basket is transferred from the transfer cask to the shipping cask.

7.3.2.2 Shielding Evaluation

A combination of computer codes and manual calculations are used for radiation shielding evaluation of the standard handling, transfer, and storage operations. The results from these analyses are summarized in Figures 7.3-9 through 7.3-12 for the spent nuclear fuel and the GTCC waste.

7.3.2.2.1 Direct Dose Evaluation

QAD-CGGP (Reference 3), a three-dimensional point-kernel code, is used for gamma dose rate calculations at the transfer and storage cask ends, as well as for the transfer cask radial gamma dose rate calculations. Also, gamma dose rate calculations for the GTCC basket are performed using the QAD-CGGP code. The QAD-CGGP shielding model geometries are shown in Figures 7.3-1 through 7.3-8. The material descriptions for each shielding model material region are shown in Tables 7.3-1 and 7.3-2. QAD-CGGP calculates gamma flux levels at each detector location specified by the user.

The MCNP monte carlo shielding code is used to calculate neutron dose rates for the transfer and storage cask ends, as well as for the transfer cask side. MCNP is capable of modeling three-dimensional geometries using explicit elemental material descriptions for each material zone. MCNP uses point-wise cross section data, so no group structures are used. MCNP gives average flux levels (and spectra) over each user-defined surface area detector.

The gamma and neutron flux-to-dose conversion factors shown in Tables 7.3-3 and 7.3-4 are used to convert the QAD-CGGP and MCNP flux output into dose rates (in mrem/hr).

The radial shielding configuration for the Trojan storage cask system is identical to that of the SNC VSC-24 storage cask system, with equal thicknesses of steel and concrete. Also, the Trojan basket has an internal structure which provides shielding equal to or greater than that present in the VSC-24 system basket. The Trojan basket employs fuel support sleeves with thicker walls than those of the VSC-24 basket. The Trojan basket also contains additional internal structural materials not present in the VSC-24 basket.

For these reasons, a simple source ratio technique is used to calculate the gamma and neutron dose rates on the Trojan storage cask side using the gamma and neutron dose rates previously calculated for the VSC-24 storage cask side (Reference 7).

The neutron dose rates on the cask side are assumed to be directly proportional to the total neutron source strength. The VSC-24 system calculated neutron dose rate (before the adjustments to account for axial peaking were made) is multiplied by the ratio of the Trojan cask total neutron dose rate over the VSC-24 cask total neutron dose rate. Then, the resulting dose rate is increased by 50% to account for the 10% peaking in the fuel burnup level (as discussed in Section 7.2.1.2). Note that the neutron source spectrum is virtually identical for the Trojan and VSC-24 systems.

For the gamma dose rates, the process is somewhat more complicated because the gamma spectrum is different for the two cask systems. To fully account for spectral effects, the total cask gamma source strengths for the two systems are compared for each gamma energy line. It was found that the Trojan system gamma source strength was greatest relative to the VSC-24 system gamma source strength for the 1.25 MeV gamma energy line, the energy line which contributes the great majority of the cask external gamma dose rate. Therefore, the source strength ratio for the 1.25 MeV gamma energy line is multiplied by the VSC-24 system cask side gamma dose rate (before peaking effects are considered) to yield the Trojan storage cask side gamma dose rates. These resulting dose rates are then increased by 10% (as discussed in Section 7.2.1.1) to account for axial burnup peaking.

7.3.2.2.2 Scattered Dose Evaluation

Measurable scattered dose rates at cask surfaces are possible. This is due to the presence of the air annulus between the basket and the inner storage cask or transfer cask surface causing scattering in axial directions, and the air inlets and outlets on the storage cask causing scattering primarily in the radial direction. Neutron and gamma dose rates at the cask air inlet are directly calculated using the MCNP monte carlo code. The dose rates calculated by MCNP automatically include both the direct and scattered flux components. Dose rate contributions from the fuel region neutron and gamma sources and from the bottom nozzle region gamma source are separately calculated and summed to yield total cask air inlet dose rates. Two-dimensional MCNP analyses model the cask air inlet structure with a high degree of accuracy (with material conservatively removed in some areas). Flux levels at the air inlet entry points are directly calculated. These are converted to dose rates using the flux-to-dose conversion factors shown in Tables 7.3-3 and 7.3-4.

Gamma and neutron dose rates at the air outlets were calculated for the VSC-24 system using shielding codes along with complex manual albedo calculations. Air outlet dose rates for the Trojan storage system are scaled from those calculated for the VSC-24 system. The QAD-CGGP code is used to model both the VSC-24 and the Trojan storage systems.

The gamma flux levels are calculated by gamma energy line at the inner end of the outlet duct (i.e., on the inner surface of the cask liner). For both systems, the fuel region and non-fuel region gamma sources are considered. The flux levels at each energy level for the two systems are compared. As with the storage cask side, the Trojan storage system gamma flux was highest relative to that of the VSC-24 system for the 1.25 MeV energy line, the line which is the dominant contributor to the gamma dose rates. Therefore, the air outlet gamma dose rate calculated for the VSC-24 system is multiplied by the ratio of the 1.25 MeV flux levels (at the inside of the outlet duct) to yield an air outlet gamma dose rate for the Trojan storage system.

Since the neutron source spectrum is similar for the two cask systems, and since the system geometries are quite similar, it is assumed that the air outlet neutron dose rate is proportional to the total cask neutron source strength. Therefore, the VSC-24 air outlet neutron dose rate is multiplied by the ratio of the Trojan storage cask neutron source strength over the VSC-24 neutron source strength. The total Trojan storage cask inlet and outlet dose rates are:

	Combined Dose Rate Fuel Storage Cask (mrem/hr)	Combined Dose Rate GTCC Storage Cask (mrem/hr)
Air Inlet	12.09	27.13
Air Outlet	4.95	3.9

Dose rates over the basket/transfer cask annulus (location 3 in Figure 7.3-11) are determined as follows: A gamma source ratio technique is used for the gamma dose rate. The ratio of the Trojan storage system top nozzle region gamma source over the VSC-24 top nozzle gamma source is greater than the ratio of fuel gamma sources for any energy line. Therefore, the top nozzle gamma source strength ratio is conservatively used. Thus, the gamma dose rate calculated at this location for the VSC-24 system is multiplied by this ratio to determine the Trojan storage system dose rates.

A different method is used for the neutron dose rate over the annulus. Due to the similarity in the geometry of the systems, the ratio of the neutron dose rate over the annulus over the neutron dose rate in the center of the top lid should be about the same for both systems. Therefore, the neutron dose rate calculated for the center of the Trojan storage system top lid is multiplied by the VSC-24 system neutron dose rate over the annulus, and divided by the VSC-24 system dose rate at the lid center, to yield the Trojan storage system neutron dose rate over the annulus.

At large distances from the storage cask, air and ground scattering effects contribute significantly to the dose rates. Dose rates were calculated as a function of distance for the VSC-24 storage cask. These dose rates were calculated with the SKYSHINE II code, which included all air and ground scatter effects. The dose rate contributions from four distinct sources are separately determined: the direct gamma and neutron dose rates (from particles emitted from the cask side surface), and the indirect gamma and neutron dose rates (from particles emitted from the cask top which scatter towards the detector area).

These four dose rate contributions are determined for the Trojan system by multiplying the calculated VSC-24 system values by the appropriate ratio in the cask surface dose rate (e.g., the ratio in the side surface gamma dose rate or the top surface neutron dose rate).

The off-surface cask dose was calculated for the original VSC-24 system using MCNP. This analysis was performed using a conservative point source model that over estimates the dose due to the use of an averaged cask surface dose. The Trojan ISFSI site dose rates will be based on scaled versions of the VSC-24 off site dose rates taking the GTCC waste into consideration.

7.3.3 VENTILATION

The storage cask is designed for passive, natural convection cooling of the basket. The air flow path is formed by the channels at the bottom (air entrance), the air inlet ducts, the annulus between the basket exterior and the concrete cask interior, and the air outlet ducts. The air inlets and outlets are steel lined penetrations that take non-planar paths to minimize radiation streaming. The cask cover plate provides additional shielding to reduce skyshine radiation.

The storage cask system is designed for release of no radioactive material during normal storage conditions. No credible off normal events or accidents will result in a release of radioactive materials to the environment due to the multiple safeguards inherent in the design. Evaluations of partial and full blockage of the air inlets are presented in Chapter 8.

There are no off-gas systems required for normal operation of the ISFSI because the basket is sealed.

7.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

During fuel loading the existing Trojan plant instrumentation is utilized as described in the Trojan Nuclear Plant Defueled Safety Analysis Report in Chapter 5. During storage, area radiation and airborne radioactivity monitoring instrumentation will consist of TLDs posted at the perimeter of and in the Controlled Area near the storage casks. The TLDs will be used to monitor operation of the ISFSI for the Radioactive Effluent and Environmental Monitoring Program described in Section 7.6.

7.4 ESTIMATED ON-SITE COLLECTIVE DOSE ASSESSMENT

The transfer and storage casks are designed to limit dose rates to minimal levels for operators, inspectors, maintenance, and radiation protection personnel when the casks are being loaded, moved, and stored. Tables 7.4-1 and 7.4-2 contain the maximum design dose rates and the calculated working dose rates for loading and handling the transfer and storage casks under normal conditions. All values for dose rates include both gamma and neutron flux components.

Working dose rates and personnel requirements for the cask loading cycle, move-to-storage, and loading for off-site shipping cycle are shown in Tables 7.4-3 and 7.4-4. The operational sequence for these activities is shown in more detail in Figures 5.1-1 and 5.1-3. Based on the estimates shown in Tables 7.4-3 and 7.4-4, the collective dose for loading, moving to storage, and loading for off-site shipping for the 34 fuel casks and 2 GTCC casks is:

Dose (person-rem)			
	GTCC (2 casks)	Fuel (34 casks)	Total
Load Cask	1.56	49.07	50.63
Move to Storage	0.37	2.04	2.41
Load for Shipping	1.23	7.14	8.37
Total	3.16	58.25	61.41

From Tables 7.4-3 and 7.4-4, conservative estimates of the periodic inspection and surveillance requirements result in a collective dose of about 5 rem/yr while the casks are being stored. This dose is based on a daily visual inspection of each stored cask, twice a day temperature readings of the air outlet temperature of each cask, quarterly radiation protection surveys, and annual inspections of the storage cask concrete and concrete pad. The annual collective dose will decrease with the age of the cask because the estimate is based on a freshly loaded cask.

The annual dose estimate for surveillance is considered very conservative based on operating experience from Consumers Power (Palisades) which shows that the annual dose for daily temperature and screen checks, monthly and annual radiological surveys, security surveillances, snow removal, and pad surveillances is about 120 mrem per year for thirteen casks.

7.5 RADIATION PROTECTION PROGRAM

7.5.1 ORGANIZATION

The PGE organization that will implement the Radiation Protection Program during ISFSI construction and fuel loading is described in Section 9.1.1 and shown in Figure 9.1-1.

The PGE organization that will implement the Radiation Protection Program during long term spent nuclear fuel storage at the ISFSI is described in Section 9.1.2 and shown in Figure 9.1-2.

7.5.2 RADIATION PROTECTION EQUIPMENT, INSTRUMENTATION, AND FACILITIES

The various equipment and instrumentation for performing radiation surveys and measuring and minimizing personnel exposure, and the facilities for radiation protection activities are summarized in this section. The radiation protection equipment, instrumentation, and facilities are highly simplified because the storage cask introduces limited radiological hazards.

7.5.2.1 Radiation Protection Instrumentation

Radiation protection instrumentation, including radiation detection, airborne monitoring, and personnel monitoring instrumentation will not normally be located or maintained at the Trojan ISFSI. This instrumentation will be owned, operated, maintained, and calibrated by and at an off-site facility. The Trojan ISFSI will maintain an agreement or contract with the off-site facility to provide radiation protection services which are mainly anticipated to entail direct radiation surveys, contamination surveys, and personnel monitoring device reading and calibration. The off-site facility will provide their own instruments and personnel and will be responsible for their own training and qualification.

7.5.2.2 Area Radiation Monitoring Instrumentation

Area radiation monitoring instrumentation consists of TLDs posted at the perimeter of and in the Controlled Area near the storage casks. TLDs will be read quarterly to monitor direct radiation from the ISFSI.

7.5.2.3 Radiation Protection Facilities

Due to the minimal radiological hazards introduced by ISFSI operation, the ISFSI will not have radiation protection facilities onsite. Decontamination services, bioassay services, protective clothing, respirators, and additional instrumentation could be obtained from off-site sources, if required, although no situations are anticipated which would necessitate use of these services or equipment.

7.5.3 RADIATION PROTECTION PROCEDURES

The purpose of this section is to summarize how ISFSI procedures implement the Radiation Protection Program to maintain radiation exposure ALARA while spent nuclear fuel and GTCC waste are stored in the ISFSI.

7.5.3.1 Control of Radiation Exposure to the Public

Monitoring, analyzing, and reporting radiation levels in the environment is performed in accordance with the Radioactive Effluent and Environmental Monitoring Program to demonstrate that the dose to the public is below regulatory limits and ALARA.

Radiation monitoring will be accomplished by posting TLDs at the perimeter of and in the Controlled Area near the storage casks and reading the TLDs quarterly.

No gaseous, liquid, or solid radioactive effluents are produced by the storage system because of its sealed design. Therefore, routine monitoring for effluents is not performed.

7.5.3.2 Control of Personnel Radiation Exposure (Occupational)

Personnel radiation exposure is maintained ALARA by a combination of shielding, access control, contamination control, surveys and monitoring, work planning, training, and sound radiation protection practices implemented by procedures. The procedures for personnel radiation protection are prepared consistent with the requirements of 10 CFR 20 and are approved, maintained, and adhered to for activities involving personnel radiation exposure.

7.5.3.2.1 Shielding

The objective of radiation shielding is to reduce external doses to personnel, in conjunction with a program for controlling personnel access and occupancy in radiation areas, to levels which are both ALARA and within the regulations defined in 10 CFR 20. Radiation protection implementing procedures provide for evaluation of the use of temporary shielding for activities involving high dose rates.

7.5.3.2.2 Access Control and Area Designations

The Restricted Area, as defined in 10 CFR 20, has the same boundaries as the isolation zone that surrounds the ISFSI Protected Area. Physical access to the isolation zone is restricted by the debris fence. Access into the Protected Area is controlled as described in the ISFSI Security Plan.

A Radiologically Controlled Area (RCA) is an area where access is controlled for the purpose of protecting individuals from exposure to radiation. RCAs are determined by the radiation level, contamination level, or the presence of radioactive materials. Procedures describe the requirements for radiological postings advising workers of potential radiological hazards at the entrance and boundaries of RCAs.

7.5.3.2.3 Facility Contamination Control

Radioactive contamination of the ISFSI is not anticipated because the external surface of the basket is checked for loose surface contamination before the storage cask is moved to the pad as described in Section 7.2.2. In addition, the spent nuclear fuel is inside the seal-welded basket and there are no credible accidents that would cause a gaseous, liquid, or solid release of radioactivity. However, procedures direct the use of various practices and equipment to ensure that the potential for the spread of contamination is controlled at the source to the greatest extent possible.

7.5.3.2.4 Personnel Contamination Control

As stated above, the basket is checked for loose surface contamination prior to being placed in the storage cask and there is no credible accident that would cause radioactive contamination of the ISFSI. However, surveys for contamination at the storage cask air inlets and outlets are routinely performed to confirm that contamination is not present. If contamination was discovered by a survey, protective clothing could be obtained to prevent contamination of personnel who would need to enter the contaminated area. Similarly, respiratory protective equipment is not required because airborne radioactivity is not credible either. However, the potential for airborne radioactivity would be considered if surface contamination were discovered. Respiratory protective equipment could be obtained if by evaluation it was determined that use of respiratory equipment would result in exposures that are ALARA.

7.5.3.2.5 Area Surveys

Quarterly surveys are performed in the accessible areas of the ISFSI. These surveys consist of contamination surveys and external radiation measurements in appropriate areas. Additionally, specific surveys are performed as needed for operational and maintenance functions involving potential exposure of personnel to radiation or radioactive materials.

7.5.3.2.6 Personnel Monitoring

TLDs are worn by personnel within RCAs to measure radiation dose. Monitoring for internal deposition of radioactive materials is not required because surface and airborne radioactivity is not anticipated and there is no credible accident that would result in gaseous, liquid, or solid release of radioactivity. However, monitoring for internal contamination could be performed by contracted offsite facilities if desired.

7.5.3.2.7 Work Planning

Work in RCAs is planned prior to performance. Consideration is given to dosimetry requirements, personnel protective equipment, monitoring requirements, and special cautions pertinent to the work. The planning also considers the maximum radiation level that will be encountered.

7.5.3.2.8 Training

Individuals requiring unescorted access to the ISFSI receive training which includes radiological protection fundamentals. Individuals who require access to RCAs will receive radiation protection training commensurate with their responsibilities in accordance with 10 CFR 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations".

The need for specialized ALARA training is evaluated during work planning in accordance with radiation protection implementing procedures. Specialized ALARA training may include dry runs, pre-job briefings, and other special training classes.

7.5.3.2.9 Controls, Practices, and Special Techniques

Radiation protection implementing procedures specify that during the planning phase for activities in high dose rate areas, various engineering controls to minimize exposures should be evaluated and/or implemented. These engineering controls and practices include, but are not limited to, temporary shielding; remote surveillance equipment; multi-discipline input regarding ALARA goals; pre-job, in-progress, and post-job briefings; and adequate lighting, ventilation, work space, and work area accessibility.

7.5.3.3 Records and Reports

PGE will maintain records of the radiation protection program, surveys, and individual monitoring results, as well as records that show compliance with the dose limits for individual members of the public. PGE will submit reports of individual monitoring as required by 10 CFR 20.

7.6 ESTIMATED OFF-SITE COLLECTIVE DOSE ASSESSMENT

7.6.1 RADIOACTIVE EFFLUENT AND ENVIRONMENTAL MONITORING PROGRAM

No radioactive gas, liquid, or solid waste effluents are released from the ISFSI during operation. Therefore, a radioactive effluent monitoring system is not required and routine monitoring for effluents is not performed. The radioactive effluents released during fuel loading operations, which is a 10 CFR 50 licensed activity, are monitored and controlled by existing plant systems as explained in Chapter 6.

The ISFSI will emit direct radiation that will be monitored in the environment. The Radioactive Effluent and Environmental Monitoring Program will be implemented by posting TLDs at the perimeter of and in the Controlled Area near the storage casks. TLDs will be read quarterly to monitor radiation levels in the nearby vicinity of the ISFSI.

7.6.2 ANALYSIS OF MULTIPLE CONTRIBUTION

Once the ISFSI is completed and the Trojan Nuclear Plant is decommissioned, the only significant radiation will come from the storage installation. No other nuclear facility is projected for the vicinity of the ISFSI (i.e., within a 5-mile radius).

The incremental contribution of the ISFSI to the total dose of a member of the general public has been estimated by calculation. The dose is estimated as 24 mrem per year at a distance of 100 meters from the ISFSI. This dose satisfies the requirements of 10 CFR 72.104 for members of the general public and corresponds to the distance from the ISFSI to the Trojan Central Building. Therefore, members of the public could occupy the Trojan Central Building, or any other buildings further than 100 meters from the ISFSI, during the ISFSI storage period, if the measured doses at that building are shown to comply with 10 CFR 72.104 by measurement.

The calculated dose at 100 meters compares with a background dose of 14.2 mrem per year (based on a 2,000 hour occupancy) at the Trojan site.

7.6.3 ESTIMATED DOSE EQUIVALENTS

There are no radioactive effluents to be released from the Trojan ISFSI.

7.6.4 LIQUID RELEASE

There are no radioactive liquids to be released from the Trojan ISFSI.

7.7 REFERENCES

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3. "QAD-CGGP: A Combination Geometry Version of QAD-P5a, A point Kernel Code System for Neutron and Gamma Ray Shielding Calculations Using the GP Buildup Factor," RSIC, CCC-493 (1986)

4. Portland General Electric Co., Trojan Nuclear Plant, Specification No. TD-06, "Functional Requirements and Specifications for a Dry Independent Spent Fuel Storage Installation (ISFSI)", Revision 1.

5. WMG-9418, "Preliminary Characterization, Trojan Reactor Internals", WMG Inc., Peekskill, NY, 1/95.

6. "Characteristics of Potential Repository Wastes", DOE/RW-0184, Office of Civilian Radioactive Waste Management, U.S. Department of Energy, July 1992.

7. Sierra Nuclear Corporation, "Safety Analysis Report for the Ventilated Storage Cask System", PSN-91-001, Revision 0A, December 1993.

8. Cacciapouti, R.J., "PWR Axial Burnup Power Profile Database", Proceedings of the Fifth Annual High Level Radioactive Waste Management International Conference, Volume 2, American Nuclear Society, May 1994.

9. Multi-Purpose Canister (MPC) Subsystem Design Procurement Specification", DBG00000-01717-63000-00001, Revision 3, Office of Civilian Radioactive Waste Management, U.S. Department of Energy, May 1994.

10. Croff, A.G., "Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code", ORNL/TM-6051, Oak Ridge National Laboratory, September 1978.

Table 7.2-1

Major Isotopic Contributors to Fuel Source Term

	Burnup 40,000 MWd Enrichment 3.02% Decay Time 5 years		Burnup 45,000 MWd Enrichment 3.30% Decay Time 6 years	
Isotope	Curies/MTIHM *	% Total	Curies/MTIHM	% Total
H3	7.73E+02	0.10	7.31E+02	0.11
Fe55	1.08E+03	0.15	8.35E+02	0.13
Co60	5.91E+03	0.80	5.19E+03	0.78
Kr85	8.15E+03	1.10	7.64E+03	1.16
Sr90	7.90E+04	10.62	7.71E+04	11.65
Y90	7.90E+04	10.63	7.71E+04	11.66
Ru106	2.17E+04	2.91	1.09E+04	1.65
Rh106	2.17E+04	2.91	1.09E+04	1.65
Sb125	6.01E+03	0.81	4.68E+03	0.71
Te125M	1.47E+03	0.20	1.15E+03	0.17
Cs134	5.11E+04	6.87	3.65E+04	5.52
Cs137	1.24E+05	16.61	1.21E+05	18.25
Ba137M	1.17E+05	15.73	1.14E+05	17.26
Ce144	1.20E+04	1.62	4.94E+03	0.75
Pr144	1.20E+04	1.62	4.94E+03	0.75
Pm147	3.14E+04	4.23	2.41E+04	3.65
Eu154	1.12E+04	1.50	1.03E+04	1.56
Eu155	5.63E+03	0.76	4.89E+03	0.74
Pu238	6.05E+03	0.81	6.00E+03	0.91
Pu241	1.36E+05	18.28	1.30E+05	19.58
Am241	1.45E+03	0.20	1.66E+03	0.25
Cm244	7.74E+03	1.04	7.45E+03	1.13
Total	7.43E+05	99.49	6.61E+05	99.99

*Metric Tons Initial Heavy Metal

Table 7.2-2

Fuel Region Gamma Source Strength (gammas/sec-cask)
for Various Burnup and Cooling Time Combinations
(including control component source)

Line Energy (MeV)	Gamma Source Strength 40,000 MWd - 5 year	Gamma Source Strength 45,000 MWd - 6 year
0.01	4.022E+16	3.685E+16
0.025	9.631E+15	8.553E+15
0.0375	1.044E+16	9.847E+15
0.0575	8.000E+15	7.249E+15
0.085	5.243E+15	4.664E+15
0.125	5.431E+15	4.870E+15
0.225	4.365E+15	3.810E+15
0.375	2.665E+15	2.211E+15
0.575	7.260E+16	7.190E+16
0.85	1.807E+16	1.585E+16
1.25*	8.860E+15	8.433E+15
1.75	1.142E+14	9.709E+13
2.25	4.993E+13	2.275E+13
2.75	1.875E+12	1.024E+12
3.5	2.409E+11	1.321E+11
5.0	3.961E+08	5.134E+08
7.0	4.568E+07	5.921E+07
9.5	5.248E+06	6.801E+06
Total	1.856E+17	1.743E+17

* 1.25 MeV source strength includes Co^{60} gammas from activated control components.

Table 7.2-3

FUEL REGION NEUTRON SOURCE STRENGTHS (Neutrons/sec-cask)

Neutron Energy Range (MeV)	Neutron Source Strength 40,000 MWd - 5 yr. cool	Neutron Source-Strength 45,000 MWd - 6 yr. cool	Normalized Neutron Source Spectrum Fraction
6.43 - 20.0	1.695E+08	2.197E+08	0.0185
3.0 - 6.43	1.925E+09	2.494E+09	0.21
1.85 - 3.0	2.126E+09	2.755E+09	0.232
1.4 - 1.85	1.201E+09	1.556E+09	0.131
0.9 - 1.4	1.622E+09	2.102E+09	0.177
0.4 - 0.9	1.769E0+09	2.292E+09	0.193
0.1 - 0.4	3.464E+08	4.489E+08	0.0378
Total	9.159E+09	1.187E+10	1.0

Table 7.2-4

Relative Burnup Level and Source Strengths for PWR Assembly Axial Sub-Sections

Axial Span (inches from fuel bottom	Relative Burnup Level	Relative Neutron Source Strength
0 - 7.2	0.59	0.109
7.2 - 14.4	0.89	0.613
14.4 - 21.6	1.03	1.132
21.6 - 28.8	1.07	1.329
28.8 - 36.0	1.09	1.436
36.0 - 64.8	1.1	1.492
64.8 - 100.8	1.09	1.436
100.8 - 108.0	1.07	1.329
108.0- 115.2	1.05	1.227
115.2 - 122.4	1.02	1.087
122.4 - 129.6	0.96	0.842
129.6 - 136.8	0.82	0.435
136.8 - 144.0	0.56	0.088

Table 7.2-5

NON-FUEL REGION Co^{60}
GAMMA SOURCE STRENGTHS
(γ/sec-cask)

Non-Fuel Region	Fuel Burnup Level (MWd/MTU)	Cooling Time (years)	1.173 MeV Gamma Source Strength	1.333 MeV Gamma Source Strength
Bottom Nozzle	40,000	5	5.021E+12	5.021E+12
Gas Plenum	40,000	5	7.347E+12	7.347E+12
Top Nozzle	40,000	5	1.538E+13	1.538E+13
Bottom Nozzle	45,000	6	4.718E+12	4.718E+12
Gas Plenum	45,000	6	6.904E+12	6.904E+12
Top Nozzle	45,000	6	1.445E+13	1.445E+13

Table 7.2-6

GTCC GAMMA SOURCE
(γ/sec-cask)

NUCLIDE	ENERGY LINE (MeV)	γ/(sec-cask)
Mn54	0.8348	6.88E+13
Co60	1.333	2.92E+16
Co60	1.173	2.92E+16
Nb94	0.871	9.25E+10
Nb94	0.702	9.25E+10
Totals	N/A	5.84E+16

Table 7.3-1

ELEMENTAL DENSITIES (gm/cc)

Elements	Atomic #	Concrete	SS 304	Carbon Steel	GTCC Full Basket	GTCC Half Basket	Lead	Air	Neutron Shield	Fuel	Bottom Nozzle	Plenum	Top Nozzle
Hydrogen	1	0.013							6.500E-2				
Boron-10	5								8.000E-3	0.003			
Boron-11	5								3.800E-2	0.013			
Carbon	6								2.000E-2	0.004			
Nitrogen	7				7.401E-4	1.480E-3		9.765E-4					
Oxygen	8	1.165						2.997E-4	9.660E-1	0.198			
Sodium	11	0.040							1.000E-3				
Mg	12	0.006			2.516E-2	5.033E-2							
Aluminum	13	0.107							2.610E-1	0.044			
Silicon	14	0.737	0.059						1.000E-2		0.008		0.006
Sulfur	16	0.003							1.050E-1				
Argon	18							1.665E-5					
Potassium	19	0.045											
Calcium	20	0.194							1.310E-1				
Chromium	24		1.505		3.026E-1	6.053E-1					0.207		0.144
Mang.	25		0.159						1.000E-3		0.022		0.015
Iron	26	0.029	5.465	7.821	1.517	2.678			6.900E-1	0.511	1.189	0.504	0.525
Cobalt	27				2.319E-3	4.638E-3							
Nickel	28		0.733		1.645E-1	3.289E-1					0.101	0.025	0.070
Zirc.	40									0.319	0.227	0.319	0.031
Niobium	41				1.480E-4	2.961E-4							
Moly	42				4.276E-3	8.553E-3							
Lead	82						11.34						
U-238	92									1.476			
Total		2.339	7.921	7.821	2.017	3.678	11.34	1.293E-3	2.296	2.568	1.754	0.848	0.791

Table 7.3-2

ELEMENTAL DENSITIES (Atoms/barn-cm)

Elements	Atomic #	Concrete	SS304	Carbon Steel	GTCC Full Basket	GTCC Half Basket	Lead	Air	Neutron Shield	Fuel	Bottom Nozzle	Plenum	Top Nozzle
Hydrogen	1	7.770E-3							3.885E-2				
Boron-10	5								5.040E-4	1.720E-4			
Boron-11	5								2.070E-3	7.060E-4			
Carbon	6								1.000E-3	2.180E-4			
Nitrogen	7				3.183E-5	6.365E-5		4.199E-5					
Oxygen	8	4.390E-2						1.128E-5	3.636E-2	7.472E-3			
Sodium	11	1.050E-3							2.580E-5				
Mg	12	1.490E-4			2.759E-4	5.518E-4							
Aluminum	13	2.390E-3							5.830E-3	9.830E-4			
Silicon	14	1.580E-2	1.270E-3						2.250E-4		1.750E-4		1.220E-4
Sulfur	16	5.640E-5							1.970E-3				
Argon	18							2.510E-7					
Potassium	19	6.930E-4											
Calcium	20	2.920E-3							1.970E-3				
Chromium	24		1.743E-2		3.506E-3	7.011E-3					2.397E-3		1.673E-3
Mang.	25		1.740E-3						9.000E-6		2.400E-4		1.670E-4
Iron	26	3.130E-4	5.894E-2	8.435E-2	1.636E-2	2.888E-2			7.440E-3	5.516E-3	1.282E-2	5.436E-3	5.658E-3
Cobalt	27				2.370E-5	4.740E-5							
Nickel	28		7.520E-3		1.687E-3	3.375E-3					1.034E-3	2.515E-4	7.220E-4
Zirc.	40									2.103E-3	1.502E-3	2.103E-3	2.030E-4
Niobium	41				9.596E-7	1.919E-6							
Moly	42				2.685E-5	5.369E-5							
Lead	82						3.296E-2						
U-238	92									3.736E-3			
Total		7.504E-2	8.690E-2	8.435E-2	2.191E-2	3.999E-2	3.296E-2	5.352E-5	9.625E-2	2.091E-2	1.817E-2	7.791E-3	8.545E-3

Table 7.3-3
Neutron Energy Group Flux-to-Dose Conversion Factors
(mrem/hr per neutron/cm^2-sec)

Neutron Energy (MeV)	Conversion Factor
2.12E-07	3.78E-03
7.67E-07	3.96E-03
2.09E-06	4.14E-03
6.58E-06	4.32E-03
1.96E-05	4.50E-03
6.50E-05	4.68E-03
3.42E-04	4.68E-03
1.97E-03	4.32E-03
5.72E-02	6.48E-03
0.331	5.40E-02
.083	0.1188
1.47	0.1332
2.09	0.1296
2.41	0.1260
2.74	0.1260
3.54	0.1296
4.51	0.1332
5.66	0.1404
7.27	0.1476
9.09	0.1476
11.1	0.1656
13.56	0.2088

Table 7.3-4

Gamma Flux-to-Dose Conversion Factors
(mrem/hr per γ/cm^2-sec)

Gamma Energy (MeV)	Conversion Factor
0.01	3.96E-03
0.03	5.82E-04
0.05	2.90E-04
0.07	2.58E-04
0.1	2.83E-04
0.15	3.79E-04
0.2	5.01E-04
0.25	6.31E-04
0.3	7.59E-04
0.35	8.78E-04
0.4	9.85E-04
0.45	1.08E-03
0.5	1.17E-03
0.55	1.27E-03
0.6	1.36E-03
0.65	1.44E-03
0.7	1.52E-03
0.8	1.68E-03
1.0	1.98E-03
1.4	2.51E-03
1.8	2.99E-03
2.2	3.42E-03
2.6	3.82E-03
2.8	4.01E-03
3.25	4.41E-03
3.75	4.83E-03
4.25	5.23E-03
4.75	5.60E-03
5.0	5.80E-03
5.25	6.01E-03
5.75	6.37E-03
6.25	6.74E-03
6.75	7.11E-03
7.5	7.66E-03
9.0	8.77E-03
11.0	1.03E-02
13.0	1.18E-02
15.0	1.33E-02

Table 7.4-1

Maximum Expected Dose Rates for the Storage Cask System (Fuel)

	Dose Rate (mrem/hr)		
Location	Design	Surface	Working[a]
Transfer Cask Side	300	252.9	100.6
Basket Top (Outside Surface of Structural Lid)	200	143	120
Concrete Cask Top	200	98.7	83
Concrete Cask Side	100	18.9	10.1

[a]Working dose is the calculated dose rate one meter from the surface

Table 7.4-2

Maximum Expected Dose Rates for the Storage Cask System (GTCC)

	Dose Rate (mrem/hr)		
Location	Design	Surface	Working[a]
Transfer Cask Side	300	133.8	68.4
Basket Top (Outside Surface of Structural Lid)	200	7.1	6
Concrete Cask Top	200	3.2	2.7
Concrete Cask Side	100	52.2	31.1

[a]Working dose is the calculated dose rate one meter from the surface.

Table 7.4-3

Estimated Personnel Exposure Doses while Operating the Fuel Cask System

Activity	Personnel Work Groups	Exposure Time (hrs)	Working Dose Rate (mrem/hr)	Exposure (person-mrem)
Load Transfer Cask	2 Operators	5.5	0.2[a]	2.2
Monitor	1 R.P.	5.5	0.2[a]	1.1
Decontaminate Cask	2 R.P.	4.0	100[b]	800
Monitor	1 R.P.	4.0	10	40
Weld Shield and Structural Lid	2 Welders/ 1 Inspector	1.0	120	360
Vacuum Dry	1 Technician	8.0	10	80
Monitor	1 R.P.	1.0	120	120
Load Storage Cask	2 Operators	1.5	10	30
Monitor	1 R.P.	1.0	10	10
Totals		31.5	----	1443.3
Move to Storage	2 Operators	2.0	10	40
Monitor	1 R.P.	2.0	10	20
Totals		4.0	----	60
Load Shipping Cask	2 Operators	9.5	10	190
Monitor	1 R.P.	2.0	10	20
Totals		11.5	----	210
Annual Surveillances of Fuel Casks in Storage	1 Specialist	424.7[c]	10	4247[c]

a Radiation reading in Spent Fuel Pool area.
b Assumes worst case of dry basket. If water is left in basket as planned, dose rate will be less.
c Includes daily visual inspection of air inlets/outlets, twice daily temperature reading, and annual concrete inspection of 34 fuel storage casks. Also, includes 25 mrem/year for quarterly radiation protection surveys and annual concrete storage pad inspection for entire ISFSI.

Table 7.4-4

Estimated Personnel Exposure Doses while Operating the GTCC Cask System

Activity	Personnel Work Groups	Exposure Time (hrs)	Working Dose Rate (mrem/hr)	Exposure (person-mrem)
Load Transfer Cask	2 Operators	5.5	0.2[a]	2.2
Monitor	1 R.P.	5.5	0.2[a]	1.1
Decontaminate Cask	2 R.P.	4.0	68[b]	544
Monitor	1 R.P.	4.0	7	28
Weld Shield and Structural Lid	2 Welders/ 1 Inspector	1.0	6	18
Vacuum Dry	1 Technician	8.0	7	56
Monitor	1 R.P.	1.0	6	6
Load Storage Cask	2 Operators	1.5	31	93
Monitor	1 R.P.	1.0	31	31
Totals		31.5	----	779.3
Move to Storage	2 Operators	2.0	31	124
Monitor	1 R.P.	2.0	31	62
Totals		4.0	----	186
Load Shipping Cask	2 Operators	9.5	31	589
Monitor	1 R.P.	2.0	14	28
Totals		11.5	----	617
Annual Surveillances of GTCC Casks in Storage	1 Specialist	24.8[c]	31	768.8[c]

a Radiation reading in Spent Fuel Pool area.
b Assumes worst case of dry basket. If water is left in basket as planned, dose rate will be less.
c Includes daily visual inspection of air inlets/outlets, twice daily temperature reading, and annual concrete inspection of 2 GTCC casks.

8.0 ACCIDENT ANALYSIS

The analyses of normal and off-normal events and accident design events identified by ANSI/ANS 57.9, as applicable to the ISFSI, are presented in this Chapter. Regulatory Guide 3.48 specifies that the four event types in ANSI/ANS 57.9 be addressed. Design Events I and II consist of normal and off-normal events that are expected to occur routinely or to occur with a frequency of approximately once per year. Design Events III and IV consist of infrequent events and postulated accidents that might occur over the lifetime of the ISFSI or hypothetical events that are postulated because their consequences may result in the maximum potential impact on the immediate environment. Section 8.1 addresses the normal and off-normal events, and Table 8.0-1 lists these events and the ISFSI components evaluated for each of them. Section 8.2 addresses the infrequent events, and Table 8.0-2 lists these events and the components evaluated for each of them. In addition, this Chapter identifies the accident conditions considered in the design of the ISFSI in accordance with State of Oregon OAR 345-26-390(4)(a).

The evaluations of normal, off-normal, and postulated accident conditions assure that the ISFSI components that are classified as important to safety are capable of performing their required functions. The required functions of the PWR Basket, GTCC Basket, Basket Overpack, and Concrete Cask are identified in Section 3.3.1.

The fuel handling components that are part of the Trojan ISFSI are identified in Section 4.7. Fuel handling components classified as important to safety are the Transfer Cask and the Transfer Station. They are relied upon to safely handle the PWR Baskets and GTCC Baskets containing high-level radioactive materials and minimize the potential for their drop. The other components perform no functions that are important to safety for ISFSI operations.

When considering the differences between baskets loaded with spent fuel or GTCC waste, the analysis is based on the PWR Basket, because the PWR Basket has greater weight and has higher internal temperature and pressure than the GTCC Basket. Therefore, the PWR Basket is more limiting with respect to stresses on the basket confinement boundary. The GTCC Basket however is more limiting with respect to radiation; therefore, additional shielding is provided as discussed in Section 7.3.2.

The analyses in this chapter reflect normal, off-normal, and infrequent events that are postulated to occur while the loaded Concrete Cask is handled and stored on the ISFSI storage pad. Events that could occur during loading of the Concrete Cask in the Fuel Building and transport to the pad will be addressed in a separate submittal. Events that could occur after a PWR Basket has been loaded into a shipping cask for transport and moved out of the ISFSI will be addressed in conjunction with the shipping cask safety analysis submitted pursuant to10 CFR 71.

8.1 NORMAL AND OFF-NORMAL EVENTS

This section covers Design Events I and II: events that would be expected to occur during normal operations and those that might occur with moderate frequency on the order of once during any calendar year of operations.

Normal operation of the ISFSI equipment and appurtenances has been described in Chapter 5. These operations include:

1. Normal operational handling, lifting, loading, and transporting the PWR Basket and Transfer Cask within the Fuel Building.

2. Loading the PWR Basket into the Concrete Cask.

3. Moving and locating the loaded Concrete Cask onto the ISFSI Storage Pad.

4. Storage of the loaded Concrete Cask on the ISFSI Storage Pad.

5. Retrieval of the PWR Basket from the Concrete Cask and loading it into a shipping cask for off-site disposal.

The structural analysis of the ISFSI components for normal operations includes consideration of anticipated loads on the Concrete Cask, PWR Basket, and basket internals during storage and handling operations. The structural analysis methodology and results have been presented in Section 4.2.5. Chapter 4 also includes a structural analysis of the PWR Basket Overpack and analyses of Concrete Cask and PWR Basket thermal-hydraulic and criticality performance under normal storage conditions.

8.1.1 OFF-NORMAL STRUCTURAL ANALYSIS

8.1.1.1 PWR Basket Off-Normal Handling Load

This event consists of a lateral impact of the PWR Basket against the inside of the Concrete Cask.

8.1.1.1.1 Postulated Cause of Event

During transfer of the Concrete Cask to the ISFSI pad, an inadvertent movement may cause lateral impact of the PWR Basket against the inside of the Concrete Cask. Additionally, during placement of the PWR Basket into the Concrete Cask crane operation may cause a lateral impact against the inside of the Concrete Cask.

8.1.1.1.2 Detection of Event

This event may be detected by observation of personnel monitoring Concrete Cask movement operations or Concrete Cask loading operations.

8.1.1.1.3 Analysis of Effects and Consequences

The off-normal handling load was analyzed for the Concrete Cask assuming a 2 ft/sec cask speed. The Concrete Cask speed will be limited by administrative procedures. This is equivalent to a drop from a height of :

$$h = v^2/2g = 0.062 \text{ ft} = 0.75 \text{ in}$$

where: v = velocity (ft/sec) = 2 ft/sec

g = acceleration of gravity = 32.2 ft/sec^2

The deceleration applied to the PWR Basket during such impact can be found using the following formula (Ref. 8.1, Chapter 15).

$$a = g[1 + (\sqrt{2h/\delta_{st}})]$$

where: δ_{st} = deflection due to the dead weight load = 0.0055 in

h = drop height = 0.75 in

g = acceleration of gravity

The dead weight deflection is calculated (by the factoring of maximum deflection in the horizontal drop analysis) as 0.0055 in. Thus, the equivalent static deceleration level is:

$$a = g[1 + (\sqrt{2 \times 0.75 / 5.5E{-3}})] = 17.5g$$

The methodology applied in the analysis is a scaling of the results of the shipping cask drop analyses. The highest drop stress (horizontal or vertical) is multiplied by the ratio of handling acceleration (17.5g) to the drop acceleration. The resulting handling stresses are identified in Table 8.1-1.

The stresses due to off-normal handling impact have been combined with the stresses due to other loads and evaluated in Table 8.1-1. It can be seen that primary membrane stress intensity (P_m) and the maximum local primary membrane plus primary bending stress intensity ($P_L + P_b$) values are within ASME code allowables for Service Level C loadings. In addition, the stresses on the basket internals are within the ASME code allowables for Service Level C loadings.

8.1.1.1.4 Corrective Actions

The PWR Basket is designed to withstand acceleration loads which bound this handling load. No corrective actions are required.

8.1.2 OFF-NORMAL THERMAL ANALYSIS

8.1.2.1 Severe Environmental Condition

This event involves severe environmental conditions consisting of sustained high temperature and low temperature cases.

8.1.2.1.1 Postulated Cause of Event

Although sustained temperature extremes of the magnitude analyzed are not expected, it is assumed that the loaded Concrete Casks on the Storage Pad are subjected to sustained high and low ambient temperatures. Analyses were performed to calculate the steady state Concrete Cask, PWR Basket, and fuel cladding temperatures for sustained 100 °F ambient conditions with 24 hour average solar loads and for -40 °F ambient conditions with no solar load. These analyses assume that the Concrete Cask has reached a steady state condition relative to the assumed ambient temperature. The maximum thermal payload of 26 kW was also used for this analysis.

The maximum anticipated heat load analysis in Section 8.2.2 evaluates an ambient temperature of 125°F which envelopes the maximum historical ambient temperature of 107°F experienced in the region.

While the temperatures for the -40°F case do not represent the absolute lowest component temperatures that could occur for a -40°F condition (as a result of using the maximum thermal load), the -40°F condition with the maximum thermal load results in the highest thermal gradients in the basket structure.

8.1.2.1.2 Detection of Event

This event may be detected by the observation by personnel and confirmed by ambient temperature monitoring.

8.1.2.1.3 Analysis of Effects and Consequences

The analysis of off-normal ambient temperature uses the thermal models described in Section 4.2.6. The same models and calculations used for the normal conditions were used to model the -40°F and 100°F ambient conditions. The maximum steady state temperatures for the 100°F case and the -40°F case are provided in Table 8.1-2.

Figures 8.1-1 and 8.1-2 provide details of the temperature distributions. As these figures and Table 8.1-2 show, the component temperatures are within the acceptance criteria. The thermal gradients across the PWR Basket interior were higher for the -40°F case than for other cases. The stress analysis for this case is described in Section 4.2.5.3.1.

8.1.2.1.4 Corrective Actions

The Concrete Cask system is designed to accommodate steady state 100°F (with the design basis solar loads) or -40°F (with no solar loads). No corrective actions are required.

8.1.2.2 Blockage of One-Half of the Air Inlets

This event postulates obstructed air flow because of blockage of one-half of the air inlets.

8.1.2.2.1 Postulated Cause of Event

This event would be caused by partial air flow blockage of the air pad channel screens or air inlet area. The Concrete Cask has four wire mesh screen covered openings which permit air entry into the air pad channels. There are two horizontal air pad channels located parallel to each other on the bottom of the concrete cask. The air inlets are openings oriented perpendicular to the air pad channels and permit a continuous flow of air through the bottom of the concrete cask, up the annulus, and out the air outlet vents located on the sides of the concrete cask near the top. Figure 4.2-4 shows the configuration of the concrete cask including the orientation of the air inlet/outlet vents.

8.1.2.2.2 Detection of Event

This event may be detected by the ISFSI facility staff as they perform their required visual surveillance or by the monitoring of cask air outlet temperatures.

8.1.2.2.3 Analysis of Effects and Consequences

The analysis of this event uses the air flow model described in Section 4.2.6. Blocking two of the inlets reduces the inlet area by a factor of two which increases the loss coefficient (k/A^2) for the entrance by a factor of four. However, the Concrete Cask flow system is designed so that the inlet losses are a relatively small portion of the total pressure drop due to the air flow. Hence, the increase in the total $\Sigma k_i/A_i^2$ is about 65%. This reduces the air mass flow rate by 17%. These combined effects (one increasing the pressure loss and one decreasing the pressure loss) increase the overall pressure loss due to the air flow by 16%. The reduced air flow creates a higher ΔT between the inlets and outlets to balance the higher flow pressure losses resulting in an increase in the air outlet temperature. When these values are input to the ANSYS finite element thermal model of the cask, the resulting concrete temperatures remain below the temperature limits as shown in Table 8.1-2.

There are no radiological releases or adverse radiological consequences from this event.

8.1.2.2.4 Corrective Actions

The required action when a vent or vents are found to be blocked is to remove the foreign material blocking the air intakes. Since screens are provided for the vents, most blocking material will be on the outside and easily removed. Materials that may be located inside the screens may be removed by hand-held tools after the screen is removed.

8.1.3 OFF-NORMAL CONTAMINATION RELEASE

8.1.3.1 Small Release of Radioactive Particulates from Exterior of Baskets

This event involves the release of surface contamination on the exterior of the ISFSI baskets to the environment.

8.1.3.1.1 Postulated Cause of Event

PWR or GTCC Basket when submerged in the spent fuel pool, may become slightly contaminated. If this surface contamination is not detected and removed prior to placement of the loaded Concrete Cask on the Storage Pad, the particulate material could eventually be released to the environment.

8.1.3.1.2 Detection of Event

Release of radioactive particulate material from the basket's exterior surfaces could be identified during radiological contamination surveys. As described in Section 7.5, these surveys will be conducted quarterly.

8.1.3.1.3 Analysis of Effects and Consequences

This analysis was performed to demonstrate that the proposed contamination limits would not result in a radiological concern at a distance of 100 meters from the ISFSI. If the surface contamination were not detected, the worst case scenario would be its release after the basket is placed in the Concrete Cask and the Cask is placed on the Storage Pad. For such an atmospheric release, it is assumed that the release consists of a plume of ^{60}Co particulates. The off-site dose can then be calculated using the general methods described in Regulatory Guide 1.25. The release parameters are a wind speed of 1 m/sec, an atmospheric dispersion factor, and a two-hour period consistent with a short duration release assumed in Regulatory Guide 1.25. The methodology applied involves assuming an allowable surface contamination on the exterior of the basket that, if released, would result in a Committed Effective Dose Equivalent

(CEDE) for ^{60}Co inhalation of 2.4 mrem to a "reference man" standing 100 meters from the point of release. The equation used to determine dose for the event is:

$$\text{CEDE (mrem)} = \text{DCF(mrem/}\mu\text{Ci)} \cdot \chi/\text{Q(sec/m}^3) \cdot \text{t(sec)} \cdot \text{BR(m}^3\text{/sec)} \cdot \text{Q(}\mu\text{Ci/sec)}$$

where:

DCF	=	Dose conversion factor = 2.187 x 10^2 mrem/μCi
χ/Q	=	Atmospheric dispersion factor = 0.035 sec/m^3
t	=	Reference Man's Exposure Time = 7200 sec
BR	=	Reference Man's Breathing Rate = 3.33 x 10^{-4} m^3/sec
Q	=	Release rate (Ci/sec)

Solving this equation to determine the quantity of ^{60}Co that corresponds to the assumed CEDE yields:

$$Activity_{^{60}Co} = Qt = \frac{CEDE}{DCF(\frac{\chi}{Q})BR}$$

Based on this equation, the activity that would result in a CEDE of 2.4 mrem to a "reference man" standing at the plume centerline during the entire two-hour passage of the release at a distance of 100 meters from the release point is 941.6 μCi of ^{60}Co. If this activity were in the form of particulate contamination evenly distributed on the top and side external surfaces of the 36 baskets (181.25 inch height; 66 inch diameter) stored at the ISFSI, the allowable surface contamination will be 1.0 x 10^{-4} μCi/cm^2.

This value (1.0 x 10^{-4} μCi/cm^2) represents the allowable beta-gamma contamination on the exterior of a basket.

The corresponding limit for α contamination is 1.0×10^{-5} $\mu Ci/cm^2$ in accordance with the convention used in 10 CFR 71.87 for allowable surface contamination on transportation packages.

8.1.3.1.4 Corrective Actions

No corrective action is required. Compliance with the limit assures that the Total Effective Dose Equivalent requirements of 10 CFR 20.1301 and State of Oregon OAR 345-26-0390(4)(f) are met.

8.1.3.2 Radiological Impact from Off-Normal Operations

The specific radiological impact of the above off-normal operations is discussed within the applicable sections. A brief summary is included in Table 8.1-3.

8.2 ACCIDENTS

This section provides the results of analyses of the Design Events III and IV events from ANSI/ANS 57.9 and of several beyond design basis accidents. The results show that the Concrete Cask system provides an adequate margin of safety for the protection of the public, facility personnel, and the environment. In addition to these accidents, this section also provides the results of analyses of bounding natural phenomena.

8.2.1 FAILURE OF FUEL PINS WITH SUBSEQUENT BREACH OF PWR BASKET

This accident involves the failure of the fuel rods in 24 fuel assemblies in a PWR Basket. Thirty percent of the available fission product gas inventory is released to the environment at ground level.

8.2.1.1 Cause of Accident

This accident is a considered a beyond design basis accident, since there is no known causal factor which results in 100% fuel rod failure and the breach of basket integrity.

The off-site radiological consequences of this beyond design basis accident are used to calculate the controlled area boundary.

8.2.1.2 Accident Analysis

This analysis assumes that 100% of the fuel rods in a basket with 24 fuel assemblies fail and release 30% of the available fission product gases. The fission product gases from the fuel rods are released at ground level to the environment. This release is then used to determine the whole body and skin dose to a reference person located at the controlled area boundary for the duration of the release.

The 30% release fraction is consistent with the guidance presented in Regulatory Guide 1.25. The dose from the released activity is assumed to be dominated by ^{85}Kr. Previous analysis (Ref. 8.3) has shown that other gaseous fission products do not add significantly to the ^{85}Kr dose from five year old fuel. Assuming the PWR Basket fails and releases the available amount of ^{85}Kr, the off-site doses can be calculated by using the methods described in Regulatory Guides 1.25 and 1.109. The important parameters for these calculations are:

Wind Speed - 1 m/sec.

Dispersion - per Figure 1 of Regulatory Guide 1.25

Release Time - short duration release, 2 hours

The assumed PWR Basket contains the following:

-A loading of 0.466 MTU/assembly (this bounds 5-year cooled assemblies),

-Three 40 Gwd/MTU assemblies with 3.42% ^{235}U, and

-Twenty One 35 Gwd/MTU assemblies with 3.56% ^{235}U.

The ^{85}Kr activity released form the basket is calculated as follows:

Activity = {(7.623E3 Ci/MTU x 3 assy) + (7.094E3 Ci/MTU x 21 assy)} x 0.466 MTU/assy
= (80,078.8 Ci x 0.30 release fraction) = 24,023.6 Ci

Therefore, the release rate Q is:

Q = 24,023.6 Ci/7200 sec release duration = 3.34 Ci/sec

The applicable equation used for whole body and skin dose is:

$$D(\text{mrem}) = DCF(\text{mrem-m}^3/\text{hr-Ci}) \times \chi/Q(\text{sec/m}^3) \times t(\text{hr}) \times Q(\text{Ci/sec})$$

where:	DCF	=	Dose Conversion Factor
			$^{85}Kr_{\gamma}$ for whole body dose = 1,739 (mrem-m^3)/(Ci-hr)
			$^{85}Kr_{\beta,\gamma}$ for skin dose = 172,420 (mrem-m^3)/(Ci-hr)
	χ/Q	=	Atmospheric Dispersion
	t	=	Release Duration = 2 hrs.
	Q	=	Release Rate = 3.34 Ci/sec

8.2.1.3 Accident Dose Calculations

The results of the calculation for several distances are shown on Table 8.2-2. The requirements of 10 CFR 72.106 for design basis accidents applicable to any individual located on or beyond the nearest controlled area boundary are 5,000 mrem to the whole body or any organ. Although this accident is a beyond design basis accident, it is being used to conservatively establish the location of the controlled area boundary pursuant to 10 CFR 72.106. As shown in Table 8.2-2, the skin dose rate at 325 meters is 5,000 mrem. Thus, the controlled area boundary is established at 325 meters.

As this event is beyond the design basis for the ISFSI and has been postulated only to define the bounding consequences of the loss of confining barriers, it was not used in establishing the boundary pursuant to Oregon Administrative Rule OAR 345-26-0390(4)(c). For the purpose of operational convenience that boundary has also been conservatively established at 325 meters. As shown in the analyses and evaluations of the design basis off-normal and infrequent (accident) events, there are no design basis accidents that will result in a loss of PWR or GTCC Basket confinement barrier or the release of significant quantities of radiological material. The limiting event with regard to OAR 345-26-0390(4)(c) is discussed in Section 8.2.4.

8.2.2 MAXIMUM ANTICIPATED HEAT LOAD

This event involves severe environmental conditions consisting of high temperature.

8.2.2.1 Cause of Accident

This event results from severe environmental conditions, an assumed 125°F ambient temperature and 12 hours of insolation, occurring when a design basis thermally loaded cask is first placed in service. These parameters are beyond the anticipated range of conditions expected at the ISFSI. The temperature assumed for this event envelopes the maximum historical ambient temperature experienced in the region (107°F).

8.2.2.2 Accident Analysis

This event was analyzed to show that under extreme heat load conditions, the accident fuel cladding temperature limit of 1,058°F (570°C) and the concrete temperature limit of 350°F (177° C) are not violated.

This analysis uses the thermal models described in Section 4.2.6. The analysis assumes an ambient temperature of 125°F with 12 hours of insolation. Table 8.1-2 provides a summary of the analysis results showing that the components remain within the acceptance criteria. Figure 8.2-1 provides details of the temperature distribution.

As a result of the higher temperature increase (due to full solar loads) on the cask surface, the thermal gradient across the concrete wall (and, hence, the stress) is lower for this accident condition than for the normal (75°F ambient) case discussed in Section 4.2.

8.2.2.3 Accident Dose Calculation

There are no radiological releases or adverse radiological consequences from this event.

8.2.3 CONCRETE CASK OVERTURNING EVENT

This event involves overturning a loaded Concrete Cask on the Storage Pad.

8.2.3.1 Cause of Accident

This accident is considered a beyond design basis accident, since there is no known causal factor which results in the Concrete Cask overturning. As shown in the evaluation of tornados, earthquakes, floods, and explosions, there are no known events at the ISFSI site that would result in overturning of a Concrete Cask on the ISFSI pad.

8.2.3.2 Accident Analysis

The following parameters are important to the overturning event:

1. Concrete crush depth of the Concrete Cask;

2. Structural integrity of the Concrete Cask; and

3. Structural integrity of the PWR Basket confinement boundary as determined by deceleration loads on the basket .

The crush depth of concrete was determined by assuming the energy is absorbed by crushing of the cask concrete. The ISFSI pad is conservatively assumed to be rigid. In addition, no energy dissipation or flexural deformation of the cask is assumed. The crush depth is found by equating the mechanical work done to the kinetic energy of the falling cask. To determine the g-load of the cask, the methodology in EPRI Report NP-7551 (Ref. 8.5) was used to develop a target hardness number. The drop height was determined by applying conservation of potential energy to convert the tip-over energy to an equivalent cask drop. The planned ISFSI pad and soil properties were used in the calculation.

For the PWR Basket and internals, it was determined that the acceleration experienced during overturning is bounded by the design acceleration. An analysis has been performed to demonstrate that the PWR Basket and its internals can withstand accelerations of 124g for a vertical drop and 44g for a horizontal drop. This analysis was performed using ANSYS finite element code and evaluated PWR Basket components in accordance with ASME Code, Section III, Subsections NC, NG, and NB (Ref. 8.4). Secondary impacts would be bounded by the two considered orientations. Therefore, as long as PWR Basket accelerations during postulated overturning events at the ISFSI site are within these limiting values, basket and internals structural integrity will be maintained.

The potential energy of the cask was determined by multiplying the cask weight by the change in height of the cask center of gravity during the overturning.

$$\Delta E = W_{cask} \times (h_v - h_g) = 290{,}000 \text{ lbs} \cdot (126.9 \text{ in} - 68.0 \text{ in}) = 1.71 \times 10^7 \text{ lb-in.}$$

where: W_{cask} = weight of loaded cask

h_v = height of cask c.g. with cask in verticle orientation

h_g = height of cask c.g. with cask in horizontal orientation

Concrete Crush Depth

It was conservatively assumed that the cask strikes a rigid surface so that the impact energy is absorbed by the cask. Since the motion is rotational and the velocity is linearly distributed along the cask length, linear distribution of crush depth was also assumed. The governing equation is:

$$\Delta E = \int_0^L \int_0^{(X/L)\delta_{max}} \sigma_u S(\Delta)\, d\Delta dx$$

σ_u	= 5000 psi	-	crushing strength of cask concrete (including dynamic strength increase factors per ACI-349)
$S(\Delta)$		-	width of contact area at top as a function of depth Δ
L	= 205 in	-	length of cask contact area
R	= 68 in	-	cask radius
$\Delta(x,t)$		-	crush depth as a function of coordinate and time
δ_{max}		-	max. crushing depth
ΔE		-	potential energy

Solving for the maximum crush depth, δ_{max}, yields a value of 1.93 in. This depth was obtained using the conservative assumption that the target surface is rigid. In reality, the crushing strength of the storage pad is lower than 5000 psi, therefore, the impacted surface would be crushed instead of the cask concrete.

Loads on Components

To determine the deceleration, the crushing assumption is reversed: the cask is assumed to be rigid and the target absorbs the energy. Assuming the energy ΔE is absorbed by the target, find the equivalent height of the horizontal drop.

$$h_{eq} = \Delta E / W_{cask} = 1.71 \times 10^7 \text{ lb-in} / 290{,}000 \text{ lb} = 59 \text{ in} \approx 60 \text{ in}$$

Based on the calculated target hardness using the EPRI NP-7551 methodology (Ref. 8.5), the acceleration resulting from a drop of 60 in. is 11g. For additional conservatism, the maximum bounding value for any drop height in accordance with EPRI NP-7551, formula 10c, was determined and used. This value is 14.4g. This is less than the PWR Basket horizontal acceleration limit of 44g; therefore the PWR Basket and internals integrity would be maintained.

The Concrete Cask was evaluated using the allowable ductility ratios of ACI-349, Appendix C, dynamic response curves from Biggs's "Structural Dynamics" (Ref. 8.6), and formulae from Reference 8.1. A one foot section of the cask was modeled as an open-ended shell. Based on an acceleration of 14.4g, impact time was found using conservation of impulse assuming a triangular impulse. An allowable ductility ratio of 10 was calculated. Using this and the dynamic curves for triangular impulse, the structure was evaluated to determine if it was adequate to resist the load. Moments and shears along the circumference of the modeled section were determined and compared to the capacity per ACI-349. The maximum dynamic moment and shear were determined to be $M^- = 1{,}696$ kip-in; $M^+ = 770$ kip-in; and $V_{max} = 63.4$ kips. Applying a Rm/F factor (calculated to be 0.35), the required static values were determined to be $M^- = 593.6$ kip-in; $M^+ = 269.5$ kip-in; and $V_{max} = 22.2$ kips. These were compared to ACI-349 capacities, and safety margins (S.M.) were determined for flexure (S.M. = 1.17) and for shear (S.M. = 4.66). Therefore flexure was determined to be controlling.

Results

Based on the analysis, it has been concluded that not more than two inches of the outer concrete would be crushed due to impact with the pad. The Concrete Cask and contained PWR Basket with internals would experience an acceleration of 14.4g due to impact with the pad. It was concluded that these components are capable of withstanding the impact loads and would remain intact.

8.2.3.3 Accident Dose Calculation

This event is a beyond design basis event. There are no radiological releases or adverse radiological consequences from this event.

8.2.4 TORNADO

This event involves the potential effects of a tornado on the ISFSI.

8.2.4.1 Cause of Accident

This event would be the result of a tornado generated at or near the ISFSI.

8.2.4.2 Accident Analysis

The Trojan ISFSI is located in an area classified by Regulatory Guide 1.76 (Design Basis Tornado) as a Region III. The Trojan ISFSI Concrete Cask is designed for a Regulatory Guide 1.76 area classified as Region I which requires the cask to withstand loads associated with the most severe meteorological conditions including extreme wind and tornado. Tornado design parameters used to evaluate the suitability of the cask include tornado winds, wind generated pressure differentials, and tornado generated missiles. A comparison of design requirements is shown in Table 8.2-3.

The methods used to convert the tornado and wind loadings into forces on the cask are based on NUREG-0800, Section 3.3.1 -Wind Loadings, and Section 3.3.2 -Tornado Loadings. Loads due to tornado generated missiles are based on NUREG-0800, Section 3.5.3 -Barrier Design Procedures.

8.2.4.2.1 Wind Loads

The tornado wind velocity is transformed into an effective pressure applied to the cask using procedures delineated in ANSI A58.1, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures". The maximum velocity pressure, p, is determined from the maximum tornado wind velocity as follows:

$$p = (0.00256)\ V^2 \text{ psf} = 331.8 \text{ psf} = 2.3 \text{ psi}$$

where:

V = Maximum tornado wind speed = 360 mph

The above effective velocity pressure is assumed constant with height and, since the cask is small in relation to the radius of the tornado, is assumed to be uniform over the projected area of the cask. Gust factors are taken as unity in evaluating effects of velocity pressures on cask surfaces.

The total tornado wind loading on the projected area of the cask, W_w, is then computed as follows:

$$W_w = p(C_f)(A_p)$$

where:

p = Effective velocity pressure (psf) = 331.8 psf = 2.3 psi

C_f = Net pressure coefficient = 0.52 (Ref. ANSI A58.1, Table 12)

A_p = Projected area of cask normal to wind

= 136 x 211.5/144 = 199.75 ft^2 = 28,764 in^2

then,

W_w = (331.8) (0.52)(199.75) = 34,464 lbs

The overturning moment (M_w) acting on the cask is:

M_w = (34,464)(211.5/2) = 3.6 x 10^6 lb-in.

The calculated force and moment are insufficient to rotate or slide a cask since the resisting moment of the cask is (289,000)[(136/2) -3] = 18.8 x 10^6 lb_f-in; and the ratio of wind force to the normal force, i.e., (34,464/289,000) or 0.119, is much less than the typical value of the sliding coefficient of friction for steel on concrete (0.3).

Stresses in the concrete cask are computed using the following cross-section properties:

Cross Sectional Area	(A_x) = (π/4) (136^2 -78^2) = 9,748 in.^{2}
Moment of Inertia	(I_y) = (π/64) (136^4 -78^4) = 1.5 x 10^7 in.^{4}
Distance to Extreme Fiber	(C_x) = 68 in.

The critical section for the cask is at the bottom of the cavity. The shear stress in the Concrete Cask, conservatively ignoring the cask liner, is W_w/A_x = 3.5 psi. The cask is assumed cantilevered at the base of the liner bottom where the moment due to the wind loading is (W_w) (211.5--19.5)/2 = 3.31 x 10^6 in.-lb. Then, the bending stress is $M/(I_y/C_x)$ = 15.0 psi. Both normal and shear stresses are included in load combinations and evaluated in Table 4.2-10.

8.2.4.2.2 Tornado Missiles

The cask is designed to withstand the effects of impacts associated with postulated tornado generated missiles as identified in NUREG-0800, Section 3.5.1.4.III.4. These missiles consist of a massive high kinetic energy missile which deforms on impact, a rigid missile to test penetration resistance, and a small rigid missile of a size sufficient to just pass through any openings in protective barriers. Missiles are assumed to impact in a manner that produces the maximum damage to the cask.

The cask body and closure elements have been analyzed for penetration resistance to an armor piercing shell missile. Results confirm that sufficient thickness of concrete and steel is available to prevent perforation, spalling or scabbing of the various cask boundary elements. Overall response of the cask has been evaluated for impacts associated with the high energy deformable missile. Such analyses indicate that the cask will remain upright following the event, and that loads associated with this impact do not compromise the integrity of the cask. The analyses which have been conducted are summarized below.

The potential effects of the small rigid missile (one-inch diameter steel sphere) were not analyzed because (1) the effects on the Concrete Cask concrete and steel cover are bounded by the effects of the armor piercing artillery shell, and (2) the air passages in the Concrete Cask do not present a direct path for the sphere to enter and strike a Basket or Overpack.

8.2.4.2.3 Local Damage Prediction -Cask Body

Local damage of the cask body has been assessed using the National Defense Research Committee (NDRC) formula. This formula has been selected as the basis for predicting depth of penetration and minimum thickness of concrete to prevent spalling and scabbing. Penetration depths computed by this method have been shown to provide reasonable correlation with test results (Ref: 8.8 and 8.9, EPRI Reports NP-440 and NP-1217). The depth of penetration, X, as predicted using this approach may be expressed as follows:

For $X/2d \leq 2.0$:

$$X = [4KNWd^{-0.8} (V/1000)^{1.8}]^{0.5} = 5.69 \text{ inches}$$

where:

f_c = Design compressive strength of concrete

d = Diameter of missile (8 inches)

K = Coefficient depending on the concrete strength

= $180/(f_c)^{0.5}$

= 2.85 assuming 4000 psi concrete

N = Missile shape factor

= 1.14 for sharp nosed missiles (Ref: EPRI NP-1217)

W = Missile Weight (275 lbs)

V = Velocity (184.8 ft/sec)

The minimum depth of concrete necessary to preclude spalling and scabbing is then selected as three (3) times the depth of penetration predicted using the NDRC formula. or 17.1 inches. Since the minimum thickness of concrete in the cask body is well in excess of 17.1 inches, it is concluded that adequate protection is provided for local damage due to tornado missiles.

8.2.4.2.4 Local Damage Prediction -Cask Closure Plate

The Concrete Cask is closed with a 0.75 inch thick steel plate bolted in place. By calculating the perforation thickness of a 126 mph, 275 lb., 8 inch diameter artillery shell impacting a steel plate, the ability of the closure plate to adequately withstand tornado generated missiles is established.

The perforation thickness in a steel plate is given in Ref. 8.10, Topical Report BC-TOP-9A, Revision 2, "Design of Structures for Missile Impact", by Bechtel Power Corporation.

$T = [(0.5)(M_m)(V_s)^2]^{2/3}/672d_m = 0.52$ in

where:

T = Perforation thickness (in)

M_m = Missile mass (slugs) = W/g = 275/32.2 = 8.54 slugs

W = Missile weight = 275 lbs

g = Acceleration due to gravity = 32.2 ft/sec^2

V_s = Missile striking velocity (ft/sec) = 184.8 ft/sec

d_m = Missile diameter (in) = 8 in.

Therefore, the cask closure plate is adequate to withstand local impingement damage due to tornado generated missiles.

8.2.4.2.5 Overall Damage Prediction

Since the cask is a freestanding structure, the principal consideration in overall damage response is the likelihood of upsetting or overturning of the cask as a result of high energy missile impacts. Such assessments have been conducted using the principles of conservation of momentum during the impact event. The analyses which are summarized below indicate that the cask will remain upright.

The force developed by the missile has been calculated using methodology presented in Ref. 8.10, Topical Report, BC-TOP-9A, Revision 2, "Design of Structures for Missile Impact," Bechtel Power Corporation, 1974. The maximum force, F, is:

$$F = (0.625)(v)(W) = (0.625)(184.8)(3{,}960) = 457.4 \text{ kips}$$

From the principles of conservation of momentum, the impulse of the force from the missile impact on the cask must equal the change in angular momentum of the cask. Likewise, the impulse force due to the impact of the missile must equal the change in linear momentum of the missile. With reference to Figure 8.2-2, these relationships may be expressed as follows:

During the deformation phase, the change in momentum of the missile becomes:

$$\int_{t_1}^{t_2} (F)\,(dt) = M\,(v_2 - v_1)$$

where:

F = Impact impulse force on missile

M = Mass of missile
= 3960 lbs/g
= 123 slugs

t_1 = Time at impact

t_2 = Time at conclusion of deformation phase

v_1 = Velocity of missile at impact
= 184.8 ft/sec (126 mph)

v_2 = Velocity of missile at t_2

The change in angular momentum of the cask about a point on the bottom rim becomes :

$$\int_{t_1}^{t_2} (M_c)\, dt = \int_{t_1}^{t_2} (211.5 \cdot F)\, dt = I_c (w'_1 - w'_2)$$

where,

M_c = Moment of the impact impulse force on the cask

I_c = Cask mass moment of inertia about a point on the bottom rim

= 1.86×10^8 slug-in^2

w'_1 = Angular velocity at time t_1

w'_2 = Angular velocity at time t_2

Equating the impulse of the impact force on the missile to the impulse of the force on the cask yields (See Figure 8.2-2):

$$-123[v_2 - (184.8 \text{ ft/sec})(12 \text{ in/ft})] = (1.86 \times 10^8/211.5)(w'_2)$$

where:

$v_2 = (248.3)w'_2$

then,

$w'_2 = 0.300$ rad/sec, and

$v_2 = 74.5$ in/sec

During the restitution phase, the final velocity of the missile will depend upon the coefficient of restitution of the missile, the geometry of the missile and target, the angle of incidence, and upon the amount of energy dissipated in deforming the missile and target. It is assumed, based upon tests conducted by EPRI, (Ref. 8.9, EPRI Report NP-440, Tests 6 and 7) that the final velocity of the missile, v_f, following the impact is zero.

Equating the impulse of the force on the missile during restitution to the impulse of the force on the cask yields:

$$-[m(v_f - v_2)] = I_c/211.5(w'_f - w'_2)$$

then:

$$w'_f = 0.300 \text{ rad/sec}$$

The final kinetic energy of the cask following the impact, E_k, is then determined as:

$$E_k = (I_c)(w'_f)^2/2 = [(1.86 \times 10^8)(0.300)^2/2]\,(1/12) = 6.98 \times 10^5 \text{ in-lb}_f$$

And the energy required to overturn the cask, E_p, is:

$$E_p = (W_c)(h) = (289.000)(18) = 5.14 \times 10^6 \text{ in-lb}_f$$

where:

W_c = 289,000 lbs

h = 17.8 in. (accounting for chamfer of the cask bottom rim)

Hence, by comparison, overturning of the cask is not postulated to occur as a result of impact from tornado generated missiles. The above analysis is conservative since it assumes direct in-line impact of the missile with the cask.

The shear capacity at the location of the Concrete Cask outlets also has been calculated to evaluate resistance of the cask to tornado generated missiles. The capacity of the concrete section is calculated using shear-friction formula (ACI-349, Section 11):

$$U_s = \phi V_n = 0.85\, A_{vf} f_y \mu = 1{,}106 \text{ kips.}$$

where:

ϕ = strength reduction factor = 0.85

A_{vf} = (32) (0.44) = 14.1 in^2 -total area of reinforcement perpendicular to the shear plane

f_y = $1.1 \cdot 60{,}000 = 66{,}000$ psi -reinforcement yield strength increased by 10% for dynamic loading (ACI-349, Appendix C)

μ = 1.4 -for monolithically placed concrete.

The maximum moment due to the impact exists in the cask section adjacent to the bottom:

$$M = Fl = 457.4 \cdot (211.5 - 19.5) = 87{,}820 \text{ kips} \cdot \text{in.}$$

Section capacity of the concrete section has been conservatively calculated per Section 9.5.2.3 of ACI-349 code.

$$U_m = \phi(f_r I_g / y_t) = 94{,}170 \text{ kips} \cdot \text{in}$$

where:

f_r = $7.5\sqrt{f'_c} = 474.34$ (concrete modulus of rupture)

where:

f'_c = 4000 psi (concrete compressive strength)

I_g = $1/4\ \pi(R^4 - r^4) = 1.5 \times 10^7$ $in.^4$ (gross moment of inertia of concrete section)

where:

R = 68 in.

r = 39 in.

y_t = 68 in. (distance from centroidal axis of gross section, neglecting reinforcement, to extreme fiber in tension)

ϕ = 0.9 (strength reduction factor -Section 9.3.2, ACI-349)

These results are evaluated in combination with other loads in Table 4.2-10.

8.2.4.2.6 Combined Tornado Wind and Missile Loading

The effects of tornado winds and missiles have been considered both separately and combined in accordance with NUREG-0800, Section 3.3.2.II.3.d. For the case of tornado wind plus missile loading, the stability of the cask has been assessed and found to be acceptable. Equating the kinetic energy of the cask following missile impact to the potential energy yields a maximum postulated rotation of the cask as a result of the impact of 2.2 degrees. Applying the total tornado wind load to the cask in this configuration results in an overturning moment of 3.8 x 10^6 in-lbs with the restoring moment on the cask calculated to be 1.76 x 10^7 in-lbs. Hence, overturning of the cask under the combined effects of tornado winds plus tornado-generated missiles will not occur.

8.2.4.2.7 Baskets/Basket Overpack Under Tornado Loadings

Since the postulated tornado missile and wind loadings are not capable of overturning the cask they have no effect on the Basket or Basket Overpack. The Concrete Cask protects the basket from direct impact of missiles. The atmospheric pressure drop caused by the tornado (-3 psid) is less than the internal pressure capacity of either the Baskets or Basket Overpack, therefore tornado effects will have no adverse impact on the Baskets or Basket Overpack.

8.2.4.3 Accident Dose Calculations

As previously described, this event does not result in the release of radiological material to the environment. However, the worst tornado missile impact calculation shows that 5.69 inches of the cask side wall was dislodged. The dose rate at the surface of the cask resulting from the loss of 5.69 inches of concrete shield is 171.4 mrem/hr for a cask with PWR Basket or 480.2 mrem/hr for a cask with GTCC waste compared to original rates of 18.9 mrem/hr and 52.2 mrem/hr respectively.

The cask would be repaired by filling the damaged area with grout. It is presumed that some period of time will be required to obtain the materials needed to repair the Concrete Cask surface. Shielding materials will be maintained on site for use in mitigating the consequences of this event until such time as a repair to the Concrete Cask surface can be completed. It is estimated that shielding materials can be in place within 12 hours of the event. It is estimated that once the necessary materials are obtained two technicians would be able to complete the repair in approximately 30 minutes. The collective dose to the repair crew would be less than or equal to approximately 0.5 person-rem (240 mrem to each technician).

This Design Basis event was considered in the establishment of the appropriate controlled area boundary pursuant to 10 CFR 72.106 and Oregon Administrative Rule OAR 345-26-0390(4)(c). The direct radiation levels at 100 meters as a result of this event are minimal for the expected duration of the event. Therefore, the controlled area boundary will be conservatively established at 325 meters based on the beyond design basis event described in Section 8.2.1.

8.2.5 EARTHQUAKE EVENT

This event is a Seismic Margin Earthquake.

8.2.5.1 Cause of Accident

An earthquake that affects the ISFSI initiates this event. The Seismic Margin Earthquake is described and discussed in Section 2.6.2.4.

8.2.5.2 Accident Analysis

The loaded Concrete Cask has been analyzed for the Seismic Margin Earthquake (SME). The SME, which has a peak horizontal ground acceleration of 0.38g and a peak vertical ground acceleration of 0.25g, has been used as the design basis earthquake for the Concrete Cask. The analysis of this event is summarized below. Use of the SME in accordance with Oregon Administrative Rule OAR 345-26-390(4)(c) is also described in Section 2.6.2.4.

The Concrete Cask is a very stiff structure. Its lowest natural frequencies are well beyond the zero period acceleration (ZPA) threshold of the site spectra. No dynamic amplification of the ground motion is expected from the cask. Although free-standing, it has been analyzed as a cantilever fixed at the base (Roark and Young, "Formulas for Stress and Strain", 5th Edition, Table 36, Case 3b). For the purpose of calculating seismic loads, the cask is treated as a rigid body attached to the ground. Equivalent static analysis methods were used to calculate loads, stresses, and overturning moments.

The fundamental natural frequency of vibration for the cask was determined as shown below (Ref. 8.1):

$$f_n = [(K_n)/2\pi]\,[(E)(I)(g)/(w)(L^4)]^{0.5} = 48.8 \text{ cycles per second}$$

where:

f_n = Frequency of the n-th mode

K_n = 3.52 for first mode of vibration

E = Modulus of Elasticity = $57{,}000\,(f_c)^{0.5} = 57{,}000\,(4{,}000 \text{ psi})^{0.5}$ = 3,604,996 psi

I = 1.49×10^7 in^4

g = 386.4 in/sec/sec

L = Height of cask = 211.5 in

w = Uniform weight density of cantilever = 289,000/211.5
= 1366.4 lb/in

It can be seen from Reg. Guide 1.60 that the dynamic amplification factor for this frequency is unity and the loads can be treated as static.

The Concrete Cask has been evaluated statically for overturning by conservatively applying equivalent static loads to the cask in each of two orthogonal horizontal directions simultaneously with an upward vertical component. Combination of the three components is performed in accordance with Reference 8.12. The reference recognizes that the maximum accelerations from the three directions cannot occur at the same time. It suggests that when one of the components is at its maximum value, the other two can be taken as 40% of their corresponding peaks. Although the SME peak horizontal ground acceleration was developed from the geometric mean of the two peak horizontal ground accelerations, for conservatism in the cask design an orthogonal horizontal ground acceleration component of 40% of the peak horizontal ground acceleration (0.40 x 0.38g) will be conservatively considered to occur in combination with the peak horizontal ground acceleration. The total seismic uplift potential has been calculated by multiplying the ground acceleration in each of the two horizontal directions by the weight of the fully loaded cask and combining the resulting component forces on the basis of the square root of the sum of the squares. For the SME, the resulting factor of safety against overturning is:

F.S. = Restoring Moment/Overturning Moment

Horizontal seismic acceleration = $[(0.38g)^2 + (0.40 \cdot 0.38g)^2]^{0.5} = 0.41g$

Horizontal seismic load = $0.41\ g\ W_c$

Vertical seismic acceleration = $(0.40 \cdot 0.25g) = 0.10g$

Vertical Seismic load = $0.10\ g\ W_c$

W_c = 289,000 lbs

then:

$$F.S. = [(W_c)(1.0\text{-}0.1)(65)] / [(W_c)(0.41)(109.5)] = 1.30 > 1.10$$

Therefore, the SME criteria are satisfied and no uplift of the edge of the cask will occur.

Furthermore, as Figure 8.2-3 shows, a vertical ground displacement of approximately 5.5 feet would be required to move the center of gravity over the edge of the cask so that overturning could occur. This type of ground displacement and/or failure of the foundation is considered to be unrealistic; hence, it is concluded that in addition to not overturning due to the SME inertia loads, the cask will also not overturn due to vertical movement of the foundation. Therefore, based on this analysis, it can be concluded that the Concrete Cask will not overturn or fall during an SME.

The PWR Basket and Concrete Cask are rugged and, since overturning is precluded, their stresses due to SME are negligible. The basket stresses are bounded by the much higher drop accelerations, while the Concrete Cask seismic demands can be calculated as follows:

Shear: $V = 0.41W_c = 0.41\ (289) = 118.5$ kips

Moment $M = V\,l = (118.5)\ (211.5-19.5) = 22{,}750$ kip-in

Both of these values are lower than the section capacities calculated in Section 8.2.4.2.5. The seismic shear and moment are included in the structural evaluation Table-4.2-10 in Chapter 4.0.

8.2.5.3 Accident Dose Calculations

The seismic event will not topple or damage the Concrete Cask, thus, there are no radiological consequences. The casks are relatively undisturbed by the earthquake.

8.2.6 PRESSURIZATION

This event produces maximum internal basket pressure and establishes the values used in stress calculations applicable to the basket containment/confinement boundary. These values conservatively bound pressurization events that could occur in postulated accident scenarios.

8.2.6.1 Cause of Accident

This event is considered a beyond Design Basis event. There is no known causal factor for this event. The event would result from the breach of fuel rods in a PWR Basket with release of 30% of the fission gases combining with 100% of the fill gases (helium). This would pressurize the basket shell and structural lids.

8.2.6.2 Accident Analysis

The analysis of this accident entails calculation of the free volume in the PWR Basket, calculation of the quantity of helium fill gas and fission gas from 24 fuel assemblies, and the subsequent calculation of the pressure in the basket if these gases are added to the He backfill gas (at 1 atm) already present in the basket. The internal gas temperature is assumed to be 650°F. The free volume of the basket and fuel rod voids was calculated to be 366,052 in^3.

The fuel rods were assumed to be backfilled during manufacture with helium gas at 35 atm which resulted in a calculated 151.4 moles of available backfill gas. The quantity of fission gases was conservatively estimated assuming that 30% of the total fission gases present are released from the fuel. The total quantity of available fission gas was calculated to be 821 moles assuming 0.25 atoms of gas/fission (Ref. 8.2) and a burnup of 65,000 MWd/MTU. The quantity of helium backfill gas in the basket was calculated to be 122 moles.

Therefore, the total number of moles (N) of gas in the basket is:

$$N_{total} = N_{basket} + (N_{rod\ fill}) + (0.3)(N_{fission\ gas})$$

$$N_{total} = 122.0 \text{ moles} + 151.4 \text{ moles} + (0.3)821 \text{ moles} = 519.7 \text{ moles}$$

Using the quantity of gas in the basket and an assumed temperature of 650°F, the pressure in the basket was calculated with the ideal gas law equation PV = nRT. The calculation resulted in a PWR Basket internal pressure of 64.4 psia (49.7 psig).

This hypothetical accident pressure loading is evaluated in tandem with the normally occurring dead weight and handling loading per Section III, Division 1, Class 2 of the ASME Code. The resulting stresses (in combination with other loads) and corresponding acceptance criteria are summarized in Table 8.2-1. As shown in this table, stresses are within the Code allowables for the Service Level D loadings. The stresses in the Basket Overpack resulting from the hypothetical accident pressure were also determined to meet Code allowable limits.

8.2.6.3 Accident Dose Calculations

There are no radiological releases or adverse radiological consequences from this event.

8.2.7 FULL BLOCKAGE OF AIR INLETS

This accident postulates sudden removal of air flow through the air inlets.

8.2.7.1 Cause of Accident

This event would be caused by complete air flow blockage of the air pad channel screens or air inlet area. The Concrete Cask has four wire mesh screen covered openings which permit air entry into the air pad channels. There are two horizontal air pad channels located parallel to each other on the bottom of the concrete cask. The air inlets are openings oriented perpendicular to the air pad channels and permit a continuous flow of air through the bottom of the concrete cask, up the annulus, and out the air outlet vents located on the sides of the concrete cask near the top. Figure 4.2-4 shows the configuration of the concrete cask including the orientation of the air inlet/outlet vents.

8.2.7.2 Accident Analysis

The analysis of this event uses the air flow model described in Section 4.2.6. In this analysis, a new air flow path is established through the two lower outlets which become air inlets and the other two remain outlets. The curve shown in Figure 8.2-4 is derived by establishing flow paths in which air flows in and down along a fraction of the annulus, up through the remaining fraction, then out through the two remaining outlets. The minimum total flow resistance occurs with 50% down flow and 50% up flow through the annulus. In the absence of a driving force to cause flow imbalance, the flow will be along the path of least resistance with 50% down flow and 50% up flow. The expected maximum outlet air temperature for this condition is calculated as 263°F. Also evident from Figure 8.2-4 is that minor air flow imbalances have no significant effect on outlet temperature.

Using the calculated air temperature at the air annulus as one of the boundary conditions, the Concrete Cask model is used to calculate the temperature distributions of the concrete and the PWR Basket shell. The basket model is used to calculate the fuel cladding temperature. Steady state conditions are used in the analyses. The maximum calculated steady state temperatures for these components are summarized in Table 8.1-2 and are below the allowable temperatures. Table 8.1-2 also provides the results for a basket with overpack.

This event would be detected by the ISFSI facility staff as they perform their required visual surveillance or by the monitoring of cask air outlet temperatures. The required action when a vent or vents are found to be blocked is to remove the foreign material blocking the air intakes. Since screens are provided for the vents, blocking material will likely be on the outside and easily removed. Materials that may be located inside the screens may be removed by hand-held tools after the screen is removed.

8.2.7.3 Accident Dose Calculations

There are no radiological releases or adverse radiological consequences from this event.

8.2.8 EXPLOSIONS OF CHEMICALS, FLAMMABLE GASES. AND MUNITIONS

This analysis addresses the hazards posed by potential explosions on transportation routes and in the vicinity of the ISFSI. The effects of accidents related to operation of a natural gas turbine combined cycle plant at the Trojan site are addressed in Section 8.2.14.

8.2.8.1 Cause of Accident

As presented in Section 2.2.3.1, the only source of potential explosions near the Trojan site that could affect safety related structures is shipment of commercial explosive cargo near the plant. Trojan plant structures and the ISFSI site itself contain no explosive materials. The small quantities of gasoline or fuel oil that may be contained in the fuel tanks of vehicles (e.g., forklifts and mobile cranes) or standby power supply engines near the ISFSI present an insignificant explosion hazard. Explosions unrelated to transportation are not considered significant. Refer to Section 2.2.3.1 for additional information on potential sources of explosions in the vicinity of the site.

In addition, the Trojan DSAR analysis calculated the probability of a disabling accident based on a 2.2 psi overpressure recognizing that the actual overpressure could be 4.4 psi as a result of reflected waves. The probabilities for a disabling accident from rail and barge shipments were each less than 10^{-6} per year and would be similar for the ISFSI.

8.2.8.2 Accident Analysis

As noted in Section 2.2.3.1, the maximum anticipated transportation-related explosion overpressure at the plant site is 2.2 psi. Considering reflected shock waves from a detonation on a nearby transportation route, the resulting overpressure may increase by a factor approaching two. An overpressure of 4.4 psi is conservative for an analysis of the ISFSI Concrete Casks. As noted above, the explosion hazard from activities at the ISFSI site is insignificant.

The Concrete Casks have been shown in Section 8.2.4 to withstand a tornado wind pressure of 331.8 psf (or 2.3 psi) and missile impacts without sliding or overturning. The magnitude of

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The Concrete Casks have been shown in Section 8.2.4 to withstand a tornado wind pressure of 331.8 psf (or 2.3 psi) and missile impacts without sliding or overturning. The magnitude of explosion that would result in overturning or sliding of a Concrete Cask was determined as follows:

The force required to slide a cask is:

$$F_{slide} = W_{cask} \times 0.3 = 289{,}000 \text{ lbs} \times 0.3 = 86{,}700 \text{ lbs}$$

where:

0.3 is the friction coefficient between the Concrete Cask and the Storage Pad

W_{cask} = Loaded Concrete Cask weight

The moment required to overturn a cask is:

$$M = 289{,}000 \text{ lbs} \times 65 \text{ in} = 18.8 \times 10^6 \text{ lbs-in}$$

where:

65 in is the moment arm

The force required to develop the above moment:

$$F_{overturn} = M / (L/2) = 18.8 \times 10^6 \text{ lbs-in} / (211.5 \text{ in} / 2) = 177{,}778 \text{ lbs}$$

where:

L = Length of the Concrete Cask

The force required to slide the cask is smaller and, therefore, is controlling. The minimum pressure on the cask to result in this force is:

$$p = F_{slide} / (C_f A_p) = 86{,}700 / (0.52)(199.75) = 5.8 \text{ psi}$$

where:

C_f = Net pressure coefficient = 0.52 (Ref. ANSI A58.1, Table 12)

A_p = Projected area of cask normal to wind

$= 136 \times 211.5/144 = 199.75 \text{ ft}^2$

Therefore, the pressure required to cause cask sliding is 5.8 psi which is much greater than the assumed 4.4 psi pressure that could be caused by explosions in the vicinity of the ISFSI. An even greater pressure would be required to overturn a cask. Based on the foregoing, the integrity of the ISFSI Concrete Casks, PWR Baskets, and Basket Overpacks would not be adversely affected by postulated explosions near the site.

8.2.8.3 Accident Dose Calculations

There are no radiological consequences from this accident

8.2.9 FIRES

Section 2.2.3.3 provides information regarding the hazard to the ISFSI presented by fires. No significant fires are expected at the ISFSI. The major transient combustible used within the ISFSI would be the gasoline, propane, or diesel fuel oil used in forklifts and the mobile crane. Normally, these vehicles would not be located at the ISFSI. Forklifts would be used during the initial movement of the Concrete Casks to the pad, and a mobile crane would be used for the infrequent event of installing a Basket Overpack and for loading of Shipping Casks. When these vehicles are in use, they will be accompanied by personnel who would detect and suppress the small fires associated with fuel leaks.

The plant is protected from industrial and forest fires by natural barriers and by the distance between combustibles and the ISFSI casks. Additional protection is provided by the paved open areas surrounding the ISFSI. In addition, the massive concrete walls of the Concrete

Casks provide shielding from the effects of thermal flux generated by nearby fires. Therefore, fires pose an insignificant hazard to the ISFSI.

8.2.10 COOLING TOWER COLLAPSE

The potential hazard associated with a postulated collapse of the 492 foot high Trojan Plant cooling tower was identified in Section 2.2.3.5. The cooling tower has been determined to pose no threat to the ISFSI owing to the distance between the tower and the ISFSI. The shortest distance between the cooling tower and the ISFSI is over 800 feet. This distance is sufficiently large that the unlikely collapse of the tower would not have any adverse impact on the ISFSI.

8.2.11 VOLCANISM

8.2.11.1 Cause of Accident

Volcanic activity near Trojan Plant are addressed in Sections 2.5.6-of the TNP DSAR. Four volcanoes are located in the general area, the closest one being 34 miles from the site.

8.2.11.2 Accident Analysis

The discussion of volcanoes in DSAR Section 2.5.6 concludes that the potential eruptions pose a minimal risk to the plant. Nevertheless, the effects that are believed to be of concern are ash fall, mud flow, and flooding. The TNP DSAR discusses the maximum ash thickness that would occur from an eruption similar to the May 18, 1980 paroxysmal eruption of Mt. St. Helens if the ash clouds were directed at TNP. An ash fall depth at the ISFSI site was determined to be about 1.8 inches. This would not be sufficient ash to block the Concrete Cask air inlets. However, the full blockage of the air inlets has been analyzed in Section 8.2.7 and shown not to result in exceeding temperature limits or to adversely affect the safe storage of spent fuel.

The results of analyses performed to determine the effects of volcanically generated mud flow and flooding, including the effects of volcanically-induced dam failures, are provided in the

DSAR. These analyses show that the effects at the ISFSI site are minimal and would not pose a hazard to the ISFSI.

8.2.11.3 Accident Dose Calculations

The effects of volcanically-induced hazards pose a negligible risk to the ISFSI, and no radiological consequences are anticipated from this event.

8.2.12 LIGHTNING

8.2.12.1 Cause of Accident

This event would be caused by meteorological conditions at the site. Lightning striking one of the Concrete Casks is not a likely event, because the ISFSI pad is surrounded on two sides by an earthen berm and some lightning protection will be afforded by the lighting towers that will be located around the storage pad (Ref. 8.7). In addition, TNP is in a low isokeraunic level area; the mean annual days of thunderstorm activity at the ISFSI will be less than ten

8.2.12.2 Accident Analysis

Even if the cask were to be hit by lightning, the likely path to ground would be from the steel Concrete Cask lid to the steel base plate via the steel cask liner and the steel air inlet ducts. The PWR Basket is surrounded by these steel structures and would not provide a likely ground path. Therefore, a lightning strike would not affect basket integrity. The heat absorbed would be insignificant from the standpoint of basket cooling due to a very short duration of the event. If the lightning entered or exited the cask via the concrete shell, some local spalling of concrete may occur. A significant loss of concrete shielding would not be expected. Concrete Cask operation would not be adversely affected.

8.2.12.3 Accident Dose Calculations

Based on the evaluation above, the radiological consequences of this accident would be similar or less than those discussed in Section 8.2.4 for a localized loss of concrete shielding following a tornado missile strike.

8.2.13 OVERPACK OPERATIONS AND OFF-SITE SHIPPING EVENTS

8.2.13.1 Interference During Raising the Basket from Concrete Cask into Transfer Cask

The Basket catches the Transfer Cask edge while being extracted. While proper procedures to ensure alignment of the components should prevent this condition from occurring, it is analyzed nevertheless to bound similar occurrences.

8.2.13.1.1 Cause of Accident

The cause is operator error for failing to assure adequate clearance and/or alignment.

This event may be detected by audible noise emitted by the basket as it contacts the Transfer Cask or by upward movement of the Transfer Cask.

8.2.13.1.2 Accident Analysis

The worst case for this condition is the instant that the PWR Basket starts to lift one side of the Transfer Cask. Conservatively assuming that only two slings out of eight are carrying the load, the load in each sling and lifting bolt can be calculated as follows:

$$P = W_{BASKET}/8 + [W_{TRANS\ CASK}/2]/2 = 80{,}000/8 + [120{,}000/2]/2 = 40{,}000 \text{ lbs}$$

The hoist rings which connect the PWR Basket to the crane hook have a rated capacity of 24,000 lbs and are designed with a 5-to-1 factor of safety. Therefore, their ultimate load is:

$$P_{ult} = (24{,}000 \cdot 5) = 120{,}000 \text{ lbs in any direction.}$$

so that the safety factor of 120,000 / 40,000 = 3.0 is provided even for this infrequent condition. Similar factors of safety are provided for the other components in the load path.

To recover from this postulated event, movement of the basket and lifting equipment would immediately stop, the operator would inspect the alignment, make necessary adjustments, and complete the lift. If unable to satisfactorily correct, the operator would lower the Basket back to the bottom of the Concrete Cask, lift the Transfer Cask to an inspection area, and visually inspect for foreign objects.

8.2.13.1.3 Accident Dose Calculation

There are no radiological releases or adverse radiological consequences from this event.

8.2.13.2 Interference During Basket Lowering into a Concrete Cask

The Basket catches on the Concrete Cask edge or side while being lowered into a Concrete Cask.

While proper procedures to ensure alignment of the components should prevent this condition from occurring, it is analyzed nevertheless to bound similar occurrences.

8.2.13.2.1 Cause of Accident

The cause is operator error for failing to assure adequate clearance and/or alignment.

This event may be detected by audible noises emitted from the PWR Basket sliding on the Transfer Cask, Concrete Cask, shipping cask, or Basket Overpack by a slackening of the wire slings which connect the PWR Basket to the crane hook.

8.2.13.2.2 Accident Analysis

Since the only force acting on the Basket during lowering is gravity, the worse case condition would be a load of 1g on the PWR Basket bottom or side if it were to be completely supported from its interference. The stresses applied to the Basket in this scenario are bounded by those analyzed in Sections 8.2.13.3 and 8.2.13.4. The Transfer Cask and Concrete Cask are both analyzed to support the weight of the Basket.

To recover from this event the operator would immediately halt lowering the Basket, inspect the area for interference, and then raise the Basket back into the Transfer Cask. If interference still existed after another attempt to lower the Basket, the operator would raise the Basket back into the Transfer Cask and remove the Transfer Cask to a suitable area for an inspection for foreign objects.

8.2.13.2.3 Accident Dose Calculation

There are no radiological releases or adverse radiological consequences from this event.

8.2.13.3 Basket Drop into Concrete Cask

The Basket is dropped vertically into the Concrete Cask or Shipping Cask during overpack operations or during Basket handling for transfer off site.

8.2.13.3.1 Cause of Accident

Postulated crane failure.

8.2.13.3.2 Accident Analysis

For the postulated scenario (drop of a Basket into the Concrete Cask or Shipping Cask) only an end drop is applicable because the basket is dropped straight into a vertical cask. The end drop height is assumed to be the distance from the top of the Transfer Cask doors to the bottom of the cask cavity. The distances are 201 inches for the Concrete Cask and 209.25 inches for the Shipping Cask. A Basket end drop within a Concrete Cask or Shipping Cask during transfer operations will result in accelerations of 64g and 51g respectively as summarized below.

To analyze consequence of the vertical drops, the methodology presented in EPRI Report NP-7551, "Structural Design of Concrete Storage Pads of Spent Fuel Casks" (Ref. 8.5) was generally used. This methodology conservatively takes into account only energy dissipation by the target (pad) assuming the cask to be absolutely rigid. EPRI Report NP-7551 utilizes value tables and curves established by the publication "Introduction to Structural Dynamics", John Biggs 1964 (Ref. 8.6). The curves in Reference 8.5 are not calculated beyond 80 inches, therefore, a ratioing calculation is utilized to develop bounding analysis.

Since the Basket would strike the target (storage pad) indirectly through the Concrete or Shipping Cask, conservation of potential energy can be applied to determine the equivalent drop height of a loaded cask.

$$h = h_{drop} (W_{Basket} / W_{cask}) = 201\ (76{,}380 / 287{,}530) = 53 \text{ in}$$

where:

W_{Basket} = weight of the loaded Basket

W_{cask} = weight of the loaded cask (Concrete Cask weight is conservatively used since a higher target hardness and basket accelerations result)

The Basket accelerations from the equivalent Concrete Cask drop height bounds a basket drop into both casks. The accelerations are below design limits for the Basket.

The following parameters were calculated using Trojan site-specific soil data and an ISFSI pad concrete thickness of 18 in.

Target hardness number for the end drop is calculated as follows:

$$S = 2rAkM_u\sigma_u / [W^3(1-e^{-\beta r}\cos\beta r)]$$

where: r = cask radius (in)

A = cask footprint (in^2)

h_c = concrete pad thickness (in)

k = foundation modulus (lb/in^2) = $\pi E_s / (1-\nu_s^2)$

M_u = uitimate moment capacity of 1 ft wide strip of slab (lb-in)

σ_u = ultimate strength of concrete (psi)

E_c = concrete elastic modulus (psi)

E_s = soil elastic modulus (psi)

W = weight of cask

β = $[E_s / 4D_c]^{1/4}$ $(in)^{-1}$

D_c = concrete slab rigidity per unit of circumference ($lb\text{-}in^2$)
= $E_c h_c^3 / [12(1-\nu_c^2)]$

Entering the curves with the calculated values for target hardness and drop height, the maximum acceleration of 64 g was determined for a vertical drop in the Concrete Cask case. (The corresponding value for the shipping cask is 51g.)

The Basket and its internals have been analyzed to withstand accelerations of 124g for a vertical drop and 44g for a horizontal drop which bound this event. This analysis was performed using ANSYS finite element code and evaluated Basket components in accordance with ASME Code, Section III, Subsections NC, NG, and NB as applicable (Ref. 8.4).

The Basket and the internals would withstand this accident with no damage to the confinement boundary.

8.2.13.3.3 Accident Dose Calculation

There are no radiological releases or adverse radiological consequences from this event.

8.2.13.4 Loaded Shipping Cask Drop

A vertical or horizontal drop of a loaded shipping cask occurs during transfer to a heavy-haul trailer or rail car prior to the installation of impact limiters.

8.2.13.4.1 Cause of Accident

Postulated crane failure.

8.2.13.4.2 Accident Analysis

For this drop scenario, both end and side impacts must be considered, and a target hardness is calculated for both end and side drops.

To analyze consequences of the drop, the methodology presented in EPRI Report NP-7551, "Structural Design of Concrete Storage Pads of Spent Fuel Casks" (Ref. 8.5) was generally used. This methodology conservatively takes into account only energy dissipation by the target (pad) assuming the cask to be absolutely rigid.

The following parameters were calculated using Trojan site-specific soil data and an ISFSI pad concrete thickness of 18 inches.

Target hardness number for the end drop is calculated as follows:

$$S = 2rAkM_u\sigma_u/[W^3(1-e^{-\beta r}\cos\beta r)]$$

where:

r	= cask radius (in)
A	= cask footprint (in^2)
h_c	= concrete pad thickness (in)
k	= foundation modulus (lb/in^2) $= \pi E_s/(1-v_s^2)$
M_u	= ultimate moment capacity of 1 ft wide strip of slab (lb-in)
σ_u	= ultimate strength of concrete (psi)
E_c	= concrete elastic modulus (psi)

E_s = soil elastic modulus (psi)

W = weight of cask

β = $[E_s/4D_c]^{1/4}$ $(in)^{-1}$

D_c = concrete slab rigidity per unit of circumference (lb-in2)
= $E_c h_c^3/[12(1-v_c^2)]$

Target hardness number for the side drop is calculated as follows:

$$S = 2AE_sM_u\sigma_u/(W^3\beta)$$

where

A = footprint (in^2) = rL

β = $[E_s/4E_cI_c]^{1/4}$ $(in)^{-1}$

I_c = $(1/12)Lh_c^3(in^4)$

E_s = soil elastic modulus (psi)

L = length of cask (in)

r = half-width of footprint (in)

Entering the curves for an 80 inch drop height with the calculated values for target hardness and drop height, the maximum accelerations of 49g and 14g were determined for vertical and horizontal orientations, respectively.

The Basket and its internals have been analyzed to withstand accelerations of 124g for a vertical drop and 44g for a horizontal drop which bound this event. This analysis was performed using ANSYS finite element code and evaluated PWR Basket components in accordance with ASME Code, Section III, Subsections NC, NG, and NB as applicable (Ref 8.4). Secondary impacts are bounded by the two considered orientations.

The Basket and the internals would withstand this accident with no damage to the confinement boundary.

8.2.13.4.3 Accident Dose Calculation

There are no radiological releases or adverse radiological consequences from this event.

8.2.14 NATURAL GAS TURBINE COMBINED CYCLE POWER PLANT EVENTS

PGE has evaluated repowering the Trojan Plant with a natural gas-fired turbine combined cycle (NGTCC) power plant. This modification to the plant and site would involve locating one or two NGTCC facilities within the existing Turbine Building. A 16-inch diameter pipeline carrying natural gas would be routed onto the site generally from the south and west. The pipeline would enter a metering station from which an 8-inch diameter pipeline will be routed to each gas-fired turbine in the Turbine Building.

PGE will not implement the modifications required for repowering until the spent nuclear fuel currently in the spent Fuel Pool has been transferred to the ISFSI. This analysis has been performed to show that, when the spent fuel and GTCC waste have been stored within the ISFSI, the hazards associated with the operation of a NGTCC facility at TNP would have no adverse impact on the ISFSI.

8.2.14.1 Cause of Accident

The introduction of a NGTCC power plant at the TNP site introduces several potential accidents that have been evaluated in this section. The NGTCC design would include both natural gas and conventional steam-cycle turbine generators that would be located in the Turbine Building. Supply lines for natural gas would be routed to the plant. Tanks for alternative fuel supply (fuel oil) would be constructed. An exhaust stack would be required for each NGTCC unit.

The use of turbine generators introduces the possibility of turbine missiles generated by postulated turbine blade failures. These missiles potentially could strike the Concrete Casks.

Use of natural gas on the site requires the potential effects of fires and explosions to be considered. The fire hazard posed by the alternative fuel tanks is evaluated.

The exhaust stack location is evaluated to assure that a stack that failed during a seismic event or tornado would not adversely affect the ISFSI.

8.2.14.2 Accident Analysis

8.2.14.2.1 Turbine Missiles

The orientations of the gas and steam turbine generators would place the ISFSI outside of the low trajectory missile zone described in Regulatory Guide 1.115 which is 25 degrees to either side of a line perpendicular to the turbine shaft. The low trajectory missile zone is applicable to low-pressure stage shrunk-on wheels of the 1800-rpm turbines generally used with light-water-cooled reactors. Since the proposed NGTCC is a 3600-rpm gas turbine, the appropriate missile strike zone is expected to be similar, however the missile hazard associated with the high speed turbines would be a high-trajectory missile. An evaluation of the probability of a high trajectory gas turbine missile strike within the ISFSI area has been performed. The probability of a strike upon the ISFSI pad was determined to be approximately 2E-5. Assuming that the probability of missile generation due to turbine wheel breakup is approximately the same as that of a steam turbine (about 1E-5/yr), the probability of a missile being generated and striking within the ISFSI area would be less than approximately 2E-10/yr which is below the threshold value of 1E-7/yr for which a consequence analysis is required. In addition, the proposed location of the steam turbine is on El. 45-ft of the Turbine Building. On that location, intervening structures (floor slabs, structural steel and equipment) provide the necessary shielding to prevent high trajectory missiles from leaving the Turbine Building. Therefore, neither gas turbine or steam turbine missiles represent a significant hazard to the ISFSI.

8.2.14.2.2 Fire/Heat Flux Effects

To determine the effects of the radiant heat flux on the ISFSI casks, a conservative steady state cask surface temperature was calculated, by considering radiation and forced convection heat transfer. This temperature provides an upper limit for the temperature rise of Concrete Cask cooling air due to a postulated fire fed by a natural gas line break at a point closest to the ISFSI. The temperature rise was calculated to be less than 4°F, without taking credit for the substantial heat sink available from the cask concrete. Also, conservative absorptance and emittance values for concrete were used; a 5 mph wind was modeled even though the reference thermal flux calculation used a 24 mph wind; and the transient time to reach steady state, which would be quite long, was not credited. The 4°F air temperature rise is less than the 21°F heatup margin between the normal steady-state concrete temperature and the limiting concrete temperature for the design basis heat load.

The postulated fire would tend to increase ambient air temperature in the vicinity of the fire. However, this would have minimal effect on ambient temperature at the casks due to the large standoff distance, combined with the buoyancy of the heated air and the location of the cask inlet air ducts at the base of the cask. Even if ambient air temperature were raised near the casks, this would have little or no effect on the ΔT through the cask itself. The temperature dependence of the Reynolds number, through the kinematic viscosity of air, is minor when factored into the heat transfer balance.

The thermal flux effects from oil tank fires are bounded by those due to natural gas line breaks.

8.2.14.2.3 Partially Confined Explosion at the ISFSI

The partially confined explosion within the ISFSI cask array bounds the consequences of any unconfined explosion that may result from a natural gas deflagration.

A partially confined explosion presents a potentially greater hazard than an explosion in the free field due to flame acceleration in the vicinity of the ISFSI casks The casks are approximately 11.3 feet in diameter and 17.6 feet high. The center-to-center spacing of the casks is 15 feet. In an unconfined deflagration, the flame speed is typically low and flame acceleration does not occur or is minimal. However, partial confinement of the burning gas or the presence of obstacles in the flame path causes turbulence. Turbulence can accelerate the flame front. The effects of fuel reactivity, obstacle density, and confinement can be correlated to the flame speed. Based on the low reactivity of natural gas, and the medium density and open arrangement of casks on the pad, a flame speed of approximately 112 ft/sec was selected for flame fronts within the ISFSI array.

An explosion overpressure calculation was performed assuming that the ISFSI pad was completely enveloped by a cloud of natural gas at the upper flammability limit of 15% by volume, and ignition occurred at the center of a four cask array. The calculated overpressure generated by the explosion was 1.47 psi. This overpressure is bounded by the explosion analysis in Section 8.2.8.

The potential for the explosion to overturn a Concrete Cask or cause it to slide was calculated. The peak overpressure would induce a sliding force on the cask of 41.3 kips. Based on the weight of the cask (289,000 lb), the coefficient of static friction would have to be less than 0.14 to permit sliding. The coefficient of friction for the cask sitting on a concrete pad is much greater than this value, typically greater than 0.3. Therefore, sliding of the cask is not a concern.

The overturning moment induced by the pressure of the explosion would have to overcome the resisting moment resulting from the weight of the cask and the distance from its center of gravity to the edge of the cask base. The overturning moment on a cask generated by the blast pressure is less than the resisting moment and, thus, the partially confined explosion would not overturn a cask. (Refer to the discussion of sliding and overturning in Section 8.2.8.)

The partially confined explosion would have no adverse effects on operability of the ISFSI.

8.2.14.2.4 Confined Explosions Inside Plant Structures

Confined explosions within plant structures were investigated to determine if the effects of those explosions could generate missiles that could impact and cause damage to the ISFSI casks. The worst case confined explosions were those in the Containment and in the Auxiliary/Fuel Building. Other structures were found to be of a lighter construction than those structures and, therefore, the blast effects and ability to generate missiles was determined to be less than for those structures.

The debris that could reach the ISFSI site as a result of explosions in the Containment and Auxiliary/Fuel Building were calculated to be small (approximately 20 lb) and with impact energies insufficient to overturn or cause significant damage to the Concrete Casks. In addition, the debris would be unlikely to block more than half the Concrete Cask air inlets which is a condition analyzed in Section 8.1.2.2. Therefore, the effects of confined explosions in plant structures was found to pose an insignificant risk to ISFSI operability.

8.2.14.2.5 Seismic Event or Tornado

The proposed installation of the NGTCC plant would involve the construction of structures and systems designed to conventional Uniform Building Code requirements for seismic events and high wind exposures. Only the stack of the NGTCC unit that extends to the north of the Turbine Building could hypothetically impact the ISFSI in the event of extreme seismic ground motions or tornado winds and missiles. However, the stack is typically 210 feet in height, and it would be located approximately 280 feet from the ISFSI. Therefore, the stack would pose no hazard to the ISFSI should it fail during a seismic event or tornado.

Based on the evaluations described above, the following may be concluded:

1. A NGTCC plant turbine missile impact on an ISFSI Concrete Cask was determined to be not credible.

2. Overpressures and thermal flux from hypothetical bounding case unconfined or partially confined natural gas-air mixture deflagrations or oil tank fire would not result in effects beyond the design basis of the ISFSI.

3. Bounding case hypothetical natural gas-air mixture confined explosions inside plant structures would not generate debris missiles capable of causing unacceptable damage to the ISFSI.

4. Extreme seismic event or tornado effects on the NGTCC plant would not result in damage to the ISFSI.

Based on the above, it has been concluded that the proposed NGTCC plant at the Trojan site would not adversely affect the ISFSI.

8.2.14.3 Accident Dose Calculations

With the exception of missiles generated by confined explosions, no event associated with the use of a NGTCC facility presents a significant potential to cause increased offsite or occupational doses. The cask would be repaired following a missile impact by filling the damaged area with grout. It is presumed that some period of time will be required to obtain the materials needed to repair the Concrete Cask surface. Shielding materials will be maintained on site for use in mitigating the consequences of this event until such time as a repair to the Concrete Cask surface can be completed. It is estimated that shielding materials can be in place within 12 hours of the event. It is estimated that once the necessary materials are obtained two technicians would be able to complete the repair in approximately 30 minutes The collective dose to the repair crew would be less than or equal to approximately 0.5 person-rem (240 mrem to each technician). Direct radiation levels at the Controlled Area boundary as a result of this event are minimal for the expected duration of the event.

8.3 SITE CHARACTERISTICS AFFECTING SAFETY ANALYSIS

The ISFSI site is located as depicted in Figures 2.1-1 and 2.1-2. The installation is designed for storing 36 casks and its layout is shown in Figure 2.1-3. The Concrete Casks reside on a thick concrete slab with fifteen feet center-to-center spacing and an aisle through the middle of the array. The controlled area for the ISFSI site is shown on Figure 2.1-2. The ISFSI site is well shielded by an embankment on the north and east sides. Figure 1.1-2 shows the accessibility of the site to truck, rail, and barge transportation. Section 2.2.3 notes that the nearest natural gas line is approximately 1.5 miles from the site; operation of this gas line will not present a hazard to the ISFSI from explosion because of the distance from the site. Also, the hazards arising from the planned use of natural gas to fuel a gas-turbine generator on site have been addressed in Section 8.2.14.

Site characteristics that affect the safety analysis are summarized in Table 8.3-1.

8.4 REFERENCES

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8.2 D. R. Orlander, "Fundamental Aspects of Nuclear Reactor Fuel Elements,"TID-26711-P1, Dept. of Energy, Washington D.C. (1976)

8.3 E. Elias, C.B. Johnson, "Radiological Impact of Clad and Containment failures in At-reactor spent fuel Storage Facilities,: Electric Power Research Institute. Palo Alto, CA (1982)

8.4 Sierra Nuclear Corporation, SNC-95-71SAR, "Safety Analysis Report for the TranStor™ Shipping Cask System," Rev. 0, (Proprietary) December 1995

8.5 EPRI Report NP-7551. "Structural Design of Concrete Storage Pads of Spent Fuel Casks," August 1991

8.6 John Biggs. "Introduction to Structural Dynamics," 1964

8.7 D. Robinson, "Grounding and Lightning Protection." EPRI Power Plant Electrical Reference Series, Volume 5

8.8 EPRI Report NP-1217, "Local Response of Reinforced Concrete to Missile Impact", 1974

8.9 EPRI Report NP-440, "Full Scale Tornado Missile Impact Tests", 1977

8.10 Topical Report BC-TOP-9A, Revision 2, "Design of Structures for Missle Impact", Bechtel Power Corporation, 1972.

8.11 American Concrete Institute, "Code Requirements for Nuclear Safety Related Structures (ACI-349)".

8.12 NUREG/CR 0098, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants".

Table 8.0-1

Design Basis Normal and Off-Normal Events

Normal and Off-Normal Events (expected frequency up to one/year)	**PWR Basket Press. Boundary**	**PWR Basket Internals**	**Concrete Cask**
For analyses of Normal Events, refer to Chapter 4	-	-	-
1. Off-Normal Structural Analysis			
a. PWR Basket Off-Normal Handling Load	X	X	-
2. Off-Normal Thermal Analysis			
a. Severe Environmental Conditions	X	X	X
b. Blockage of One-Half of the Air Inlets	X	X	X
3. Off-Normal Contamination Release			
a. Small Release of Potential Basket Surface Contamination	Radiological Consequences Only		

Table 8.0-2

Design Basis and Beyond Design Basis Infrequent (Accident) Events

Design Basis and Beyond Design Basis Infrequent (Accident) Events	PWR Basket Press. Boundary	PWR Basket Internals	Concrete Cask
1. Failure of Fuel Pins with Subsequent Breach of PWR Basket	Radiological Consequences Only		
2. Maximum Anticipated Heat Load - 125°F Ambient Temperature and Full Solar Load	X	X	X
3. Concrete Cask Overturning	X	X	X
4. Tornado	X	-	X
5. Earthquake	X	-	X
6. Pressurization	X	-	-
7. Full Blockage of Air Inlets	X	X	X
8. Explosions of Chemicals, Flammable Gases, and Munitions	X	-	X
9. Fires	-	-	X
10. Cooling Tower Collapse	-	-	X
11. Volcanism	-	-	X

Table 8.0-2

Design Basis and Beyond Design Basis Infrequent (Accident) Events **Sheet 2 of 2**

<table>
<tr><th>Design Basis and Beyond Design Basis Infrequent (Accident) Events</th><th>PWR Basket Press. Boundary</th><th>PWR Basket Internals</th><th>Concrete Cask</th></tr>
<tr><td>12. Lightning</td><td>-</td><td>-</td><td>X</td></tr>
<tr><td>13. Overpack Operations and Off-Site Shipping Events</td><td colspan="3"></td></tr>
<tr><td>a. Interference During Raising the Basket from Concrete Cask into Transfer Cask</td><td colspan="3">Radiological Consequences Only</td></tr>
<tr><td>b. Interference During Basket Lowering into a Concrete Cask.</td><td colspan="3">Radiological Consequences Only</td></tr>
<tr><td>c. PWR Basket Drop Into Concrete or Shipping Cask</td><td rowspan="2">X</td><td rowspan="2">X</td><td rowspan="2"></td></tr>
<tr><td>d. Loaded Shipping Cask Drop</td></tr>
</table>

Table 8.0-2

Design Basis and Beyond Design Basis Infrequent (Accident) Events **Sheet 3 of 2**

Design Basis and Beyond Design Basis Infrequent (Accident) Events	PWR Basket Press. Boundary	PWR Basket Internals	Concrete Cask
14. Natural Gas Turbine Combined Cycle Power Plant Events	X		X

Table 8.1-1

PWR Basket Stresses (ksi) Resulting From Off-Normal Handling Event

Component		Off-Normal Handling	Dead Weight	Thermal	Pressure	Total	ASME Allowable Service Level C
Shell	P_m	5.8	0.1	N/A	0.2	6.1	18.6
	$P_L + P_b$	21.8	0.2		2.8	24.8	27.8
Bottom Plate	P_m	7.3	0.0	N/A	0.1	7.4	18.6
	$P_L + P_b$	14.7	0.2		1.9	16.8	27.8
Structural Lid	P_m	2.1	0.0	N/A	0.0	2.1	18.6
	$P_L + P_b$	9.7	0.1		0.6	10.4	27.8
Top Weld	P_m	8.2	0.1	N/A	0.4	8.7	18.6
	$P_L + P_b$	9.5	0.2		0.5	10.2	27.8
Shield Lid Weld	P_m	0.9	0.1	N/A	0.3	1.3	18.6
	$P_L + P_b$	2.1	0.1		0.4	2.6	27.8

Table 8.1-2

Summary of Cask Thermal Evaluation

Case	Temperatures (°F)						
	Ambient	Solar	Air Outlet	Outer Concrete	Inner Concrete	Basket Shell	Max Clad
Normal Operation							
Limits	-	-	-	**250**	**250**	-	**730**
(without Basket Overpack)							
Steady State Normal	75	no	180	87	204	287	609
(with Basket Overpack)							
Steady State Normal	75	no	182	87	204	371	685
Off-normal and Infrequent Events							
Limits	-	-	-	**350**	**350**	-	**1058**
(without Basket Overpack)							
Steady State Severe Cold	-40	no	43	-31	60	177	511
Steady State Severe Hot	100	yes	210	141	239	313	633
12 hour Max Thermal	125	yes	240	166	270	338	655
½ of Inlets Blocked	75	no	202	89	228	305	625
All Inlets Blocked	75	no	263	94	283	358	673
Basket in Transfer Cask w/ He	75	no	-	-	-	434	743
with vacuum	75	no	-	-	-	434	851
(with Basket Overpack)							
Steady State Severe Cold	-40	no	44	-31	59	278	601
Steady State Severe Hot	100	yes	212	141	239	393	705
12 hour Max Thermal	125	yes	242	166	269	414	725
½ of Inlets Blocked	75	no	203	88	221	381	694
All Inlets Blocked	75	no	266	94	284	432	741

Table 8.1-3

Summary of Impact from Off-Normal Operations

OFF-NORMAL EVENT	DOSE IMPACT AT CONTROLLED AREA BOUNDARY	DETECTION	CAUSES	CORRECTIVE ACTIONS	EFFECTS AND CONSEQUENCES
PWR Basket Off-Normal Handling	None	Observation	Inadvertent motion	None	None
Severe Environment	None	Observation	Weather	None	None
Half Air Inlet Blockage	None	Concrete Cask outlet temperature and Surveillance	Debris	Manual action to clear debris	Concrete Cask outlet temp increase
Particulate Release	Negligible	Radiological Survey	Release of external contamination	None	Negligible radiological release

Table 8.2-1

PWR Basket and Basket Overpack Stresses (ksi) Resulting From Accident Pressurization

Component		Accident Pressure	Dead Weight	Thermal	Normal Handling	Total	ASME Allowable
Shell	P_m $P_L + P_b$	2.4 34.6	0.1 0.2	N/A	0.6 2.0	3.1 36.8	37.1 55.7
Bottom Plate	P_m $P_L + P_b$	1.1 23.2	0.0 0.2	N/A	0.6 1.3	1.7 24.7	37.1 55.7
Structural Lid	P_m $P_L + P_b$	0.1 6.9	0.0 0.1	N/A	0.2 0.9	0.3 7.9	37.1 55.7
Top Weld	P_m $P_L + P_b$	4.4 6.1	0.1 0.2	N/A	0.8 0.9	5.3 7.2	37.1 55.7
Shield Lid Weld	P_m $P_L + P_b$	3.6 4.5	0.1 0.1	N/A	0.1 0.1	3.8 4.7	37.1 55.7
Overpack Shell	P_m $P_L + P_b$	3.2 8.8	0.1 0.2	N/A	N/A	3.3 9.0	44.8 67.2
Overpack Bottom Plate	P_m $P_L + P_b$	1.4 65.4	0.0 0.2	N/A	N/A	1.4 65.6	44.8 67.2
Overpack Structural Lid	P_m $P_L + P_b$	1.4 65.4	0.0 0.1	N/A	N/A	1.4 65.6	44.8 67.2
Overpack Top Weld	P_m $P_L + P_b$	- 1.4	0.1 0.2	N/A	N/A	0.1 1.6	44.8 67.2

Table 8.2-2

Beyond Design Basis Accident Dose Calculations

Distance from Concrete Cask, m	Skin Dose, mrem (beta and gamma)	Whole Body Dose, mrem (gamma)
100	4.0E4	407
200	1.2E4	116
300	5.8E3	58.1
325	5.0E3	50
330	4.8E3	48.8
500	2.3E3	23.2
1,000	749	7.6

Table 8.2-3

Regulatory Guide 1.76 Design Basis Comparison

Tornado Parameter	Trojan ISFSI Concrete Cask Design	RG 1.76 DBT (Region III) Requirement
Maximum Wind Speed	360 mph	240 mph
Rotational Speed	290 mph	190 mph
Translational Speed	70 mph	50 mph
Pressure Drop	3.0 psi	1.5 psi
Pressure Drop	2.0 psi/sec	0.6 psi/sec

Table 8.3-1

Summary of Site Characteristics Affecting the Safety Analysis

Site Characteristic	Effect on ISFSI Safety Analysis
Severe environmental conditions in summer and winter	Evaluation of steady state Concrete Cask, PWR Basket, and fuel temperatures for 100°F ambient temperature with 24 hour average solar loads and -40°F ambient temperature with no solar load
Tornados	Evaluation of possible Concrete Cask damage including overturning due to wind loading, failure of confinement due to pressure differential, and impact damage due to tornado generated missiles
Earthquakes	Evaluation of seismic motion including possible overturning
Explosion of Chemicals, Flammable Gases, and Munitions	Evaluation of effects on Concrete Cask including potential overturning and sliding
Fires	Evaluation of potential for fire hazard at the ISFSI site
Cooling Tower Collapse	Evaluation of impact of cooling tower collapse on ISFSI
Volcanism	Evaluation of the effects of potential ash, mud and flooding caused by a volcanic eruption
Lightning	Evaluation of the impact of a postulated lightning strike
Natural Gas Turbine Hazards	Evaluation of turbine missiles generated by the gas and steam turbines and fire/heat flux, explosions, and missiles caused by gas line ruptures.

9.0 CONDUCT OF OPERATIONS

This chapter discusses the organization and procedures established by Portland General Electric (PGE) for the construction, operation, and decommissioning of an Independent Spent Fuel Storage Installation (ISFSI). Included are descriptions of organizational structure, testing, training programs, normal operations, emergency planning, decommissioning, and security.

The Trojan ISFSI is jointly owned by Portland General Electric (PGE), 67.5 percent; the City of Eugene, 30 percent through the Eugene Water and Electric Board (EWEB); and Pacific Power and Light/PacifiCorp (PP&L), 2.5 percent. PGE is the majority owner and has responsibility for operating and maintaining the ISFSI. The Bonneville Power Administration (BPA), a power marketing agency under the United States Department of Energy (DOE), is obligated through net billing agreements to pay costs associated with EWEB's 30% share of TNP operation including decommissioning and spent fuel management costs. The financial capabilities of the joint owners, for construction, operation, and decommissioning of the ISFSI, are presented in the TNP Decommissioning Plan (PGE-1061).

9.1 ORGANIZATIONAL STRUCTURE

Section 9.1.1 describes the organization that will be in place during ISFSI design, construction, preoperational testing, fuel loading, startup testing, and initial operation. This organization is shown on Figure 9.1-1 and is referenced in the following sections as the construction and fuel loading organization.

Section 9.1.2 describes the organization that will be in place during long term operation of the ISFSI. This organization is shown on Figure 9.1-2 and is referenced in the following sections as the operation organization. PGE will transition responsibility for ISFSI operation from the construction and fuel loading organization to the operation organization following initial operation of the ISFSI. The construction and fuel loading organization may remain in place after the transition to assist with ISFSI operation or to perform other non-ISFSI related functions.

9.1.1 ISFSI CONSTRUCTION AND FUEL LOADING ORGANIZATION

9.1.1.1 Corporate Organization

The Trojan Site Executive and Plant General Manager is the corporate executive with overall responsibility for the design, construction, preoperational testing, fuel loading, startup testing, and initial operation of the ISFSI. The Trojan Site Executive and Plant General Manager reports to the Senior Vice President, Power Supply.

9.1.1.2 Site Organization

9.1.1.2.1 Trojan Site Executive and Plant General Manager

The Trojan Site Executive and Plant General Manager is also the senior site organizational position and is responsible for day-to-day management of the ISFSI and ensuring that the design, construction, preoperational testing, fuel loading, startup testing, and initial operation of the ISFSI are safely conducted.

9.1.1.2.2 Nuclear Oversight

The General Manager, Nuclear Oversight reports to the Trojan Site Executive and Plant General Manager and is responsible for evaluating the effectiveness of the Quality Assurance (QA) Program, auditing vendor activities, coordinating the Corrective Action Program, providing quality control coverage for site activities, and maintaining the PGE Nuclear Quality Assurance Program (PGE-8010). This position has the authority and independence to identify quality problems and to initiate stop work orders for any condition adverse to quality.

9.1.1.2.3 Engineering and Decommissioning

The General Manager, Engineering and Decommissioning reports to the Trojan Site Executive and Plant General Manager and is responsible for overseeing the design of structures and systems, preparation of specifications, procurement of materials and equipment, and construction of the ISFSI. The General Manager, Engineering and Decommissioning is also responsible for planning and scheduling of activities, design control, reviews of design and construction activities, and preparation of preoperational and startup test procedures.

9.1.1.2.4 Plant Support and Technical Functions

The General Manager, Plant Support and Technical Functions reports to the Trojan Site Executive and Plant General Manager and is responsible for licensing activities including reviewing, responding, and interpreting federal and state regulatory documents, physical security including the administration of the security organization, implementation of the site security program, and cost control.

9.1.1.2.5 Independent Review and Audit Committee

The Independent Review and Audit Committee (IRAC) is responsible for advising the Trojan Site Executive and Plant General Manager on matters relating to the safe storage of spent nuclear fuel. This review and audit function is independent of the organization responsible for operation or maintenance of the ISFSI.

IRAC is composed of a minimum of five regular or alternate members. The Trojan Site Executive and Plant General Manager designates in writing the Chairman, members, and alternates. The Chairman does not have any direct responsibility for operation or maintenance of the ISFSI. The IRAC collectively has experience and knowledge in spent nuclear fuel handling and storage, chemistry and radiochemistry, engineering, radiation protection, and quality assurance.

9.1.1.2.6 Operations

The Manager, Operations reports to the Trojan Site Executive and Plant General Manager and is responsible for performance of preoperational and startup testing, safe operation of the ISFSI, and maintaining personnel trained and qualified in accordance with the Certified Fuel Handling Training Program (PGE-1057) and Certified ISFSI Specialist Training Program (PGE-1072) for fuel handling operations and operation of ISFSI equipment that is important to safety.

9.1.1.2.7 Personnel/Radiation Protection

The Manager, Personnel/Radiation Protection reports to the Trojan Site Executive and Plant General Manager and is responsible for emergency preparedness, chemistry, radiation protection, the ALARA program, and industrial safety program.

9.1.1.2.8 Maintenance

The Manager, Maintenance reports to the Trojan Site Executive and Plant General Manager and is responsible for developing and implementing predictive, preventive, and corrective maintenance for the ISFSI.

9.1.1.3 Interrelationships with Contractors and Suppliers

The development of the ISFSI, including design, construction, testing, and operation are managed by PGE. The contractor for the design, safety analysis, and construction of the ISFSI is Sierra Nuclear Corporation (SNC). PGE reviews and approves SNC quality assurance procedures prior to their implementation, reviews and approves contractor and sub-contractor procedures prior to work at the Trojan ISFSI site, approves sub-tier suppliers performing quality related work prior to use, and approves quality-related lease equipment prior to use at the Trojan ISFSI.

9.1.1.4 Technical Staff

The design and construction for the ISFSI will be primarily performed by Sierra Nuclear Corporation (SNC). The design, calculations, and analyses will be reviewed and approved by PGE prior to construction. The qualifications of the PGE staff meet or exceed the requirements specified in Section 5.3.1 of the Trojan Permanently Defueled Technical Specifications.

9.1.2 ISFSI OPERATION ORGANIZATION

This section describes the ISFSI organization that will be in place during long term storage of spent nuclear fuel. The ISFSI operation organization is shown in Figure 9.1-2.

PGE will transition responsibility for ISFSI operation from the construction and fuel loading organization described in Section 9.1.1 to the operation organization following initial operation of the ISFSI. The construction and fuel loading organization may remain in place after the transition to assist with ISFSI operation or to perform other non-ISFSI related functions.

9.1.2.1 Corporate Organization

The Corporate Executive responsible for Trojan has overall responsibility for safe operation of the ISFSI.

9.1.2.2 Operating Organization

The ISFSI Manager reports to the Corporate Executive responsible for Trojan and is responsible for day-to-day management of the ISFSI. This position provides direction for the safe operation, maintenance, radiation protection, training and qualification, and security of the ISFSI and personnel.

ISFSI Specialists, who report to the ISFSI Manager, are responsible for day-to-day operation of the ISFSI.

9.1.2.3 Succession of Authority

In order to assure continuity of operation and organizational responsiveness to off-normal situations, a normal order of succession and delegation of authority will be established. The ISFSI Manager will designate in writing personnel who are qualified to act as the ISFSI Manager in his absence.

9.1.2.4 Corporate Support

Corporate support will be available by either corporate staff or contract personnel to provide support and expertise to the ISFSI Manager in the following areas: Quality Assurance, Engineering, Radiation Protection, Licensing, Maintenance, Security, and Environmental.

Quality assurance audits and inspections will be performed by personnel independent of the ISFSI line organization. The results of the audits and recommendations for improvement will be provided directly to the ISFSI Manager and the Corporate Executive responsible for Trojan.

9.1.2.5 ISFSI Safety Review Committee

The ISFSI Safety Review Committee provides independent review of matters related to the safe storage of spent nuclear fuel. The ISFSI Safety Review Committee's responsibilities are described in the Trojan ISFSI Technical Specifications (PGE-1071).

9.1.3 PERSONNEL QUALIFICATION REQUIREMENTS

During the construction and loading of the ISFSI, each member of the ISFSI staff shall meet or exceed the staff qualifications described in the Trojan Permanently Defueled Technical Specifications, Section 5.3.1. Fuel handling operations will be directly supervised by a Certified Fuel Handler as required by the Trojan Permanently Defueled Technical Specifications, Section 5.2.2.d.

The ISFSI Manager and ISFSI Specialists are qualified as described in Table 9.1-1. Operation of equipment and controls that are identified as important to safety for the ISFSI shall be limited to personnel who are trained and certified in accordance with the Certified ISFSI Specialist Training Program (PGE-1072) or personnel who are under the direct visual supervision of a person who is trained and certified in accordance with the Certified ISFSI Specialist Training Program (PGE-1072).

9.1.4 LIAISON WITH OUTSIDE ORGANIZATIONS

The ISFSI, including design, procurement, construction, preoperational testing, startup testing, and operation, is managed by PGE. These activities will be performed in accordance with the approved procedures. Sierra Nuclear Corporation provides engineering, technical support, and other services for the ISFSI project relating primarily to the design and construction of structures and components. Other qualified vendors may be selected to provide services and/or equipment.

9.2 PRE-OPERATIONAL AND STARTUP TESTING

Prior to operation of the Trojan ISFSI, inspections, pre-operational tests, and a startup test will be performed. The inspections ensure that the storage system and handling equipment satisfy the design criteria stated in Chapter 3. The pre-operational tests verify that the storage system functions as stated in the Safety Analysis Report. The startup test ensures that each storage system operates within the limits stated in the Trojan ISFSI Technical Specifications (PGE-1071).

PGE will submit a report of the pre-operational test acceptance criteria and test results at least 30 days prior to the receipt of spent nuclear fuel at the Trojan ISFSI in accordance with 10 CFR 72.82. The startup test for each storage cask will not be performed until actual loading of each storage cask has occurred. Therefore, results of the startup tests will not be available for inclusion in the report submitted in accordance with 10 CFR 72.82, but will be available at the site following the completion of each startup test.

Several additional tests of equipment involved with loading the storage system will be performed under the 10 CFR 50 license (e.g., load testing the Fuel Building crane). These additional tests are not pre-operational or startup tests of the storage system, but are discussed below due to their importance to the safe loading and subsequent operation of the storage system. The acceptance criteria and results of these additional tests will be available at the site following completion of each test.

9.2.1 ADMINISTRATIVE PROCEDURES FOR CONDUCTING TEST PROGRAM

The development, approval, and performance of pre-operational and startup test procedures will be procedurally controlled. These procedural controls will specify how needed changes to test procedures are incorporated.

The procedure that governs testing will specify how the test results will be evaluated, documented, and approved. Test results will be within the acceptance criteria specified in test procedures.

The procedure that governs testing will specify the process for identifying needed system modifications that are recognized during testing. Also, the procedure will require evaluation of whether retesting is required after a needed modification has been implemented.

Sierra Nuclear Corporation will be responsible for developing test procedures, performing tests, and ensuring that test acceptance criteria are satisfied for tests performed at their facility. PGE will review the tests performed by Sierra Nuclear Corporation and the test results for adequacy.

For tests performed at the Trojan site, PGE will be responsible for developing the test procedures, performing the tests, and ensuring that the test acceptance criteria are satisfied.

9.2.2 TEST PROGRAM DESCRIPTION

The test program is divided into two parts: pre-operational testing and startup testing. Inspections, which are quality control related activities, and other tests, which are not pre-operational or startup tests, are also briefly discussed in this section because of their importance to the proper operation and integrity of the storage system and handling equipment. The pre-operational, startup, and other tests are described in this section and a summary is provided in Table 9.2-1.

Pre-operational testing is only performed on the air outlet temperature sensing system. The storage system uses passive cooling, and therefore has no "operating" systems to test prior to the loading of spent nuclear fuel or GTCC waste. However, the inspections described below are performed to ensure the storage system will function in accordance with the design.

Startup testing is performed for each storage cask after loading with spent nuclear fuel or GTCC waste. Startup testing confirms that the storage system is operating properly once loaded with spent nuclear fuel or GTCC waste. Startup testing also ensures that the storage system loading is bounded by the safety analyses.

Inspections of the baskets, transfer cask, and storage casks will be performed prior to use of these components. The objective of these inspections is to ensure that the critical dimensions of these components are in accordance with the design drawings. The acceptance criteria for the critical dimensions are provided on the design drawings. Materials important for shielding will also be inspected for density. Steel properties will be verified by appropriate test reports. Structural adequacy of concrete will be determined by testing during construction.

9.2.3 TEST DISCUSSION

9.2.3.1 Physical Facilities

9.2.3.1.1 Basket and Associated Equipment

An actual basket and a part-length mock-up of a basket will be used for testing. The full-length basket will be loaded into the transfer cask to verify fit and suitability of the basket lift rig. The basket lifting rig, slings, rings, and crane(s) used to lift the basket will be load tested to demonstrate the ability to safely lift a fully loaded basket.

The part-length mock-up will be configured exactly like the top end of the basket with the shield lid and structural lid. The mock-up will be used to test the automated welding equipment, including actual welding of the lids and valve access port cover plates. Emphasis will be placed on acceptability of the weld, as well as compliance with approved ALARA practices.

The mock-up will also be used to test the cutting equipment, including actual removal of the lids from the basket after they have been welded in place. This test demonstrates that the basket lids can be safely removed.

The basket shield lid retainers will be tested to ensure that the shield lid will stay on the basket during and after a crane mishandling event or a transfer cask tip over.

A basket internal assembly will be loaded with a dummy fuel assembly, a failed fuel can, and a fuel debris canister to check the fit up and satisfactory operation of associated handling tools and equipment. A GTCC loading grate will be checked for fit up with the GTCC basket and ability to load a GTCC can into the GTCC basket.

A fuel debris canister will be tested for integrity and leak tightness under postulated operating conditions including internal pressurization, internal vacuum, and external pressure from submergence. The ability to manipulate the fuel debris canister with the remote handling tools will be checked.

The hydrostatic test and dewatering equipment will be tested to ensure that the hydrostatic testing and dewatering can be accomplished in the amount of time necessary to prevent boiling of the borated water in the basket as described in the Chapter 5. The vacuum drying and helium backfill equipment will be tested to ensure that the vacuum drying and helium backfill can be accomplished within the time limit described in Chapter 5.

9.2.3.1.2 Transfer Cask and Associated Equipment

Load testing of the transfer cask and trunnions will be performed at 300% of design load. The Lifting Yoke will be load test to 150% of design load. The crane(s) that lift the loaded transfer cask will also be load tested. Testing will also be performed prior to lifting a transfer cask if the load test has not been performed within the period of time specified in the test procedure, e.g., for loading a shipping cask several years after commencing ISFSI operation.

A test load equivalent to the heaviest fully loaded basket will be placed in the transfer cask to demonstrate the structural capability of the transfer cask bottom doors. The bottom doors will then be checked for proper operation after supporting the test load.

The system used to inject water into the annulus between the basket and the transfer cask will be tested to ensure that sufficient water is injected to minimize surface contamination of the basket external surface.

The load travel path at the site will be checked to ensure that the transfer cask can be safely moved from the Fuel Building bay to the Cask Wash Down pit. From the Cask Wash Down pit, the Transfer Cask will be moved to and lowered into the Cask Loading pit to verify the load travel path and clearances. The reverse path from the Cask Loading pit to the Cask Wash Down pit to the Fuel Building bay will also be checked if not identical.

9.2.3.1.3 Concrete Cask

The full length basket will be placed in the Concrete Cask and the shield ring and lid will be installed to check the fitup of these components. An air outlet temperature monitoring system will be tested and calibrated on each Concrete Cask prior to inserting a loaded basket.

9.2.3.1.4 Air Pad System

The air pad system will be tested by moving a test load equivalent to a fully loaded storage cask from the Fuel Building to the storage pad. This test will ensure that the air pad system can safely move the storage cask and will verify the travel path.

9.2.3.1.5 Overpack

The overpack will be tested by placing an overpack into a Concrete Cask, placing the full length basket in the overpack, and simulating closure of the overpack including installation of the quick connect, quick connect cover, and shield ring.

9.2.3.1.6 Transfer Station

The Transfer Station will be tested prior to use by placing the Transfer Cask in the sliding collar and moving the sliding collar and Transfer Cask through a transfer sequence in the Transfer Stand.

9.2.3.2 Operations

A startup test will be performed for each storage cask after it is loaded with spent nuclear fuel or GTCC waste. The startup test will measure external radiation dose rates to confirm that the design dose rates have been satisfied. This will confirm that the basket loading and estimates of personnel exposures are bounded by the safety analysis.

In addition, the startup test will confirm that the heat generated by each spent nuclear fuel storage cask is consistent with the spent nuclear fuel that is loaded in the basket. The heat generation will be confirmed by measuring the temperature difference between the storage cask air inlets (ambient air temperature) and air outlets and comparing the measured temperature difference against a calculated temperature difference that is based on the basket loading. This test will confirm that the basket is loaded as designed and does not exceed the heat loading specified in the safety analyses. Measured temperature differences that are higher or lower than the calculated difference by more than the uncertainty specified in the test procedure will be evaluated.

9.2.3.3 Test Response

The tests will be deemed successful if the acceptance criteria provided in the test procedures are achieved safely and without damage to any of the components or associated equipment.

9.2.3.4 Corrective Action

Modifications to equipment or components will be performed, should they become necessary, to ensure that the acceptance criteria are achieved. The modified equipment or components will be retested to confirm that the modification is sufficient. If required, pre-operational test procedure changes will be incorporated into the appropriate operating procedures.

9.3 TRAINING PROGRAMS

The main objective of the training program is to provide ISFSI staff personnel with the specialized training necessary to operate and maintain the ISFSI in a safe manner.

9.3.1 TRAINING PROGRAM DESCRIPTION

Individuals requiring unescorted access to the ISFSI will receive training in the following areas: Radiation Protection, Security, Radiological Emergency Plan, Quality Assurance, Fire Protection, Chemical Safety, and the Policy statement on worker responsibility for safe operation of the ISFSI. Individuals requiring continued unescorted access will receive refresher training on these topics annually.

Operation of equipment and controls that are identified as important to safety for the ISFSI shall be limited to personnel who are trained and certified in accordance with the Certified ISFSI Specialist Training Program (PGE-1072) or personnel who are under the direct visual supervision of a person who is trained and certified in accordance with the Certified ISFSI Specialist Training Program (PGE-1072).

Fuel Handlers certified in accordance with PGE-1057 (Certified Fuel Handler Training Program) will be retained until initial fuel loading and testing have been completed. After fuel, debris, and GTCC waste is safely transferred to the ISFSI, cask handling and transfer is effectively accomplished by Certified ISFSI Specialists who are trained and certified in accordance with PGE-1072, "Certified ISFSI Specialist Training Program".

Individuals who work in or frequent the Restricted Area will receive radiation protection training commensurate with their responsibilities in accordance with 10 CFR 19 , "Notices. Instructions and Reports to Workers: Inspection and Investigations."

Security training will be provided in accordance with the training and qualification requirements outlined in the Trojan ISFSI Security Plan (PGE-1073).

Records will be maintained on the status of trained personnel, training of new employees, and refresher training of present personnel.

9.4 NORMAL OPERATIONS

This section describes the administrative controls and conduct of operations associated with activities considered important to safety. Also described in this section is the management system for maintaining records related to the operation of the ISFSI.

9.4.1 PROCEDURES

Activities affecting quality are accomplished in accordance with approved and documented instructions, procedures, or drawings. Written procedures will be used for ISFSI operations, maintenance, and testing activities that are quality-related as defined in the PGE Nuclear Quality Assurance Program (PGE-8010). The review and approval process for ISFSI procedures will be procedurally controlled. Temporary changes to procedures and approval of procedures are addressed in Section 5.7 of the Trojan ISFSI Technical Specifications (PGE-1071).

ISFSI procedures will require that a change to the ISFSI or a change to an ISFSI procedure will be reviewed to ensure that the proposed change does not constitute an unreviewed safety question as defined by 10 CFR 72.48(a)(2). If the Trojan Nuclear Plant possesses a 10 CFR 50 license at the time of the proposed change, then the proposed change will also be reviewed in accordance with 10 CFR 50.59 to ensure that the proposed change does not represent an unreviewed safety question for the Trojan Nuclear Plant.

9.4.2 RECORDS

Administrative procedures will be established and maintained to ensure quality assurance records are identifiable and retrievable. In addition to quality assurance records, the following records will also be maintained in accordance with 10 CFR 72.174:

1. Operating records, including maintenance and modifications.
2. Records of off-normal occurrences.
3. Events associated with radioactive releases.
4. Environmental survey records.

5. Personnel Training and Qualification Records.
6. Records of ISFSI design and procedure changes made pursuant to 10 CFR 72.48.
7. Records showing the receipt, inventory (including location), disposal, acquisition, and transfer of spent fuel and related nuclear material as required by 10 CFR 72.72(a).
8. Records of material control and inventory procedures to account for material in storage as required by 10 CFR 72.72.

Storage of the above records will be in accordance with the requirements of the PGE Nuclear Quality Assurance Program (PGE-8010).

Security records, including security training and qualification records, will be maintained in accordance with the Trojan ISFSI Security Plan (PGE-1073).

9.5 EMERGENCY PLANNING

The Permanently Defueled Emergency Plan (PGE-1060) meets the requirements of 10 CFR 72 for the ISFSI.

The analyses of off-normal events and potential accidents associated with the ISFSI indicate that radiological releases beyond the Controlled Area are limited to small fractions of the U.S. Environmental Protective Agency (EPA) Protective Action Guidelines exposure levels, as detailed in EPA-400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents." For this reason, the actions in the Emergency Plan were designed to safeguard site personnel in the event of an off-normal condition or accident that potentially involves the release of radioactive materials. The actions in the Emergency Plan are based on the reduced likelihood of a radiological emergency and the reduced consequences associated with operation of the ISFSI.

9.6 ISFSI DECOMMISSIONING PLAN

This section describes the plans for decommissioning the ISFSI. Included are discussions of the method of decommissioning, anticipated costs, design and operational features that facilitate decommissioning, and record keeping utilized during the life of the ISFSI. Details pertaining to the financial assurance for ISFSI decommissioning (10 CFR 72.30) are provided in the Trojan Nuclear Plant Decommissioning Plan (PGE-1061).

9.6.1 DECOMMISSIONING PROGRAM

Decommissioning of the ISFSI primarily consists of transferring the spent nuclear fuel and GTCC waste contained in the sealed storage baskets to a facility for final disposal or storage. The spent nuclear fuel and Greater Than Class C waste that will be stored at the ISFSI are not eligible for near surface disposal in accordance with 10 CFR 61. The DOE is responsible for the acceptance of spent nuclear fuel and related nuclear material in accordance with the terms of the 1982 Nuclear Waste Policy Act. PGE is assuming that the DOE will accept the GTCC waste at the time of decommissioning. If the DOE will not accept the GTCC waste, then PGE will pursue other disposal alternatives.

After the spent nuclear fuel and GTCC waste are transferred to the DOE for disposal or storage. contamination and radiation surveys will be performed to determine if the ISFSI is contaminated or if ISFSI components are activated. If contamination is detected, then decontamination can be accomplished by routine radiation protection practices. The resultant radioactive waste would be packaged and shipped off site as radioactive waste. If the ISFSI components are activated. then the components would be packaged and shipped off site as radioactive waste

9.6.2 COST OF DECOMMISSIONING

The cost for decommissioning the ISFSI is estimated at approximately $18.5 million (1993 dollars). A breakdown of cost estimates based on activities is provided in Table 9.6-1. Further details of the ISFSI decommissioning costs are contained in the Trojan Nuclear Plant Decommissioning Plan (PGE-1061).

9.6.3 DECOMMISSIONING FACILITATION

The ISFSI was designed to minimize the decontamination efforts required for decommissioning. The spent nuclear fuel is loaded into steel baskets in the Trojan Spent Fuel Pool. After loading is completed, water is drained from the basket and the first of two closure lids (shield lid) is welded in place. The shield lid contains penetrations which allow draining and drying of the interior of the basket by placing the interior volume under a vacuum. The interior volume is then purged with helium to remove remaining moisture and air. After drying is completed, a slight over pressure of helium is used to fill the interior of the basket to allow testing of the basket confinement boundary for leakage and facilitate heat transfer from the spent fuel to basket shell. When leak tightness requirements are satisfied, the penetrations are welded-closed and a second lid (structural lid) is welded to provide closure of the vent penetrations and provide additional isolation of the contained radioactive material.

The process described above will take place prior to placing the basket into the concrete storage cask and moving it to the ISFSI. If necessary, the exterior surface of the basket will be decontaminated prior to moving the concrete cask to the ISFSI pad. The design of the basket and the operational process for handling the basket ensure that the radioactive materials are contained within the sealed basket which minimizes the potential for contamination of the ISFSI components and structures.

9.6.4 RECORD KEEPING FOR DECOMMISSIONING

Records of information important to the safe and effective decommissioning of the ISFSI will be maintained for the life of the ISFSI. The types of information that will be maintained as records for decommissioning are listed in 10 CFR 72.30(d).

9.7 PHYSICAL SECURITY PLAN

The purpose of the Trojan ISFSI Security Plan (PGE-1073) is to establish and maintain a program for protection of spent fuel stored within the ISFSI in accordance with 10 CFR 72, Subpart H, "Physical Protection". The Trojan ISFSI Security Plan (PGE-1073), which includes the security training and qualification requirements, has been submitted separately because it contains Safeguards Information.

TABLE 9.1-1

ISFSI STAFFING QUALIFICATIONS

Operation Organization

1. <u>ISFSI Manager</u>:

The ISFSI Manager, at the time of appointment to the position, should have a minimum of eight years of power plant experience, of which a minimum of three years shall be nuclear power plant experience. A maximum of two years of the remaining five years of power plant experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one basis. The ISFSI Manager will be trained and certified in accordance with the Trojan Certified ISFSI Specialist Training Program (PGE-1072).

2. <u>ISFSI Specialists</u>:

The ISFSI specialists, at the time of appointment to the position, should have a High School diploma or successfully completed the General Education Development (GED) test. Consistent with the assigned duties, ISFSI Specialists will be trained and certified in accordance with the Trojan Certified ISFSI Specialist Training Program (PGE-1072) and the Trojan ISFSI Security Plan (PGE-1073) training and qualification requirements.

Table 9.2-1

Pre-Operational, Startup, and Other Tests

Component	Type	Test Purpose/Objective(s)
Basket lifting equipment (attaches to shield lid)	Other	1. Check fit up with shield lid and lifting cranes. 2. Load test demonstrates ability to safely lift a fully loaded basket.
Basket automated welding system and cutting equipment	Other	1. Check fit up of shield lid, structural lid, quick connect valves, and valve access port cover plates. 2. Demonstrate ability to install and remove the lids and valve access port cover plates.
Basket shield lid retainers	Other	1. Check fit up of retainers with shield lid. 2. Demonstrate ability to keep the shield lid on the basket during/after mishandling event or transfer cask tip over.
Fuel basket internal assembly	Other	1. Check fit up with fuel basket. 2. Load dummy fuel assembly into basket internal assembly.
GTCC basket 28-slot grating	Other	1. Check fit up with GTCC basket. 2. Load GTCC can into basket using the grating.
Basket hydrostatic test, dewatering, vacuum drying, and helium backfill systems	Other	1. Check fit up with basket quick connects. 2. Demonstrate ability to pressurize/evacuate basket to required test pressure/vacuum. 3. Demonstrate ability to dewater and evacuate basket in the time required to prevent boiling. 4. Demonstrate ability to vacuum dry and backfill the basket with helium within the required time.
Fuel debris canister	Other	Test fuel debris canister under operating conditions (internal pressure, internal vacuum, and external pressure - submergence) to demonstrate structural integrity and leak tightness.

Table 9.2-1

Pre-Operational, Startup, and Other Tests

Fuel debris canister underwater handling tools	Other	Demonstrate ability to manipulate the fuel debris canister underwater.
Transfer cask lifting crane(s)	Other	Load test demonstrates ability to safely lift a fully loaded transfer cask.
Transfer cask and trunnions	Other	300% load test demonstrates ability to safely lift a loaded transfer cask.
Lifting Yoke	Other	1. Check fit up with transfer cask and crane. 2. 150% load test demonstrates ability to safely lift a loaded transfer cask.
Transfer cask bottom doors	Other	Demonstrate proper operation of bottom doors after supporting the weight equivalent to a fully loaded basket.
Transfer cask annulus water injection system	Other	Demonstrate the ability to inject sufficient water into the basket/ transfer cask to minimize contamination of basket external surfaces.
Concrete cask air pads	Other	Demonstrate ability to lift the weight equivalent of a fully loaded storage cask.
Concrete cask air outlet temperature monitoring system	Pre-op	Demonstrate proper operation of the temperature monitoring system prior to placing a loaded basket into the concrete cask.
Concrete cask shield ring and cask lid	Other	Check fit up.
Overpack automated welding system and cutting equipment	Other	Check fit up of overpack, structural lid, quick connect, and quick connect cover and demonstrate the ability to install and remove the lid and quick connect cover.
Overpack shield ring	Other	Check fit up.

Table 9.2-1

Pre-Operational, Startup, and Other Tests

Transfer station, sliding collar, and, side members	Other	1. Check fit up of transfer station components. 2. Demonstrate ability to move sliding collar with transfer cask through a transfer sequence.
Storage system performance	Startup	1. Measure external radiation dose rates to confirm estimated personnel exposures. 2. Measure fuel decay heat to confirm proper loading of basket and proper heat removal by storage system.
Component compatibility/load travel path	Other	1. Check fit up of components with each other. 2. Check load path from Spent Fuel Pool to pad. - Basket into transfer cask. - Transfer cask into Cask Loading Pit. - Transfer cask moved from Cask Loading Pit to Cask Wash Pit. - Transfer cask moved from Cask Wash Pit to Fuel Loading Bay and placed on top of concrete cask. - Basket lowered from transfer cask into concrete cask. - Concrete cask shield ring and lid installed. - Concrete cask moved from Fuel Loading Bay to storage pad. - Overpack placed in concrete cask. - Basket transferred into a concrete cask containing an overpack.

TABLE 9.6-1

ISFSI DECOMMISSIONING COSTS

ACTIVITY	ESTIMATED COST (thousands of 1993 dollars)
Demolition of ISFSI	374
Transfer Spent Nuclear Fuel and Miscellaneous Costs	2,699
Professional Services	672
Burial Cost. Low Level Waste[1]	3.291
Burial Cost. Greater Than Class C Waste[2]	11.406
Total Decommissioning Cost	18,442

[1] Separate burial of the casks as Low Level Radioactive Waste.

[2] Portions of the reactor vessel internals will be stored in the ISFSI and will require disposal as part of ISFSI decommissioning.

10.0 OPERATING CONTROLS AND LIMITS

The Concrete Cask system is passive during the storage mode and requires few operating controls. Design criteria and functional descriptions of safety features are contained in Chapter 3 (Principal Design Criteria) and Chapter 4 (Installation Design). Required functional and operating limits, limiting conditions for operations, technical bases, surveillance requirements and administrative controls are contained within the ISFSI Technical Specifications.

11.0 QUALITY ASSURANCE

Portland General Electric Company implements a Nuclear Quality Assurance (QA) Program which directs quality-related activities at the Trojan Nuclear Plant. This QA Program is described in PGE-8010, "Trojan Nuclear Plant Nuclear Quality Assurance Program."

PGE-8010 complies with Title 10, Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

In addition to 10CFR50 activities, PGE-8010 applies to activities covered by 10CFR71, Subpart H, "Quality Assurance for Packaging and Transportation of Radioactive Material," and 10CFR72, Subpart G, "Quality Assurance for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."

The Sierra Nuclear Corporation Quality Assurance Program is discussed in Section 13.2 of the Safety Analysis Report (SAR) for the Ventilated Storage Cask System by Pacific Sierra Nuclear Associates and Sierra Nuclear Corporation, dated October 1991. The Sierra Nuclear Corporation SAR is addressed in the Certificate of Compliance issued by the NRC effective May 7, 1993, on Docket Number 72-1007. The program was approved by the NRC as noted in Section 10.0 and 14.1.3 of the NRC "Safety Evaluation Report for the Pacific Sierra Nuclear Associates Safety Analysis Report for the Ventilated Storage Cask System," dated April 1993.

The Sierra Nuclear Corporation Quality Assurance Program is incorporated by reference as noted in Chapter 1, Section 1.5.

Appendix A

Drawings

FIGURE WITHHELD UNDER 10 CFR 2.390

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