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SECTION 1

INTRODUCTION AND GENERAL DESCRIPTION OF STORAGE SYSTEM

1.1 INTRODUCTION

Discharged spent fuel assemblies from Prairie Island Nuclear Generating Plant, Units 1 and 2, are currently stored on-site in a spent fuel pool. The spent fuel pool provides for long term storage of 1,386 assemblies in high-density storage racks. Typically, 48 spent fuel assemblies per unit are discharged to the spent fuel pool each cycle, which occurs approximately every 16 months. Section 3.1.1 provides a detailed description of the spent fuel. Additional information is contained in the Prairie Island Updated Safety Analysis Report (USAR) (Reference 1).

Under the current fuel cycle management strategy, the spent fuel pool will lose capacity for discharge of a full core in 1993. Storage capacity will be exhausted completely in 1994. Accordingly, additional spent fuel storage capacity is needed.

To support this need and provide storage until the Department of Energy (DOE) accepts title to spent fuel under the requirements of the Nuclear Waste Policy Act of 1982, as amended in 1987, Northern States Power Company (NSP) requested permission to build and operate an on-site Independent Spent Fuel Storage Installation (ISFSI) in compliance with restrictions and requirements of 10CFR72.

NSP chose the TN-40 and TN-40HT dry cask storage system designed by Transnuclear, Inc. The TN-40 system is more fully described in Sections 3.3 and 4.2.3 and Appendix 4A. 01244873

The TN-40HT system is described in Addendum A. The Sections in Addendum A also contain the analyses and evaluations associated with the design and operation of the TN-40HT system. 01244873

The Prairie Island ISFSI includes an Equipment Storage Building which will be used to store the cask lifting yoke and transport vehicle.

Construction of the ISFSI is scheduled to commence in March 1992. The ISFSI is scheduled to begin operation in February 1993.

The capacity of the ISFSI will enable Prairie Island to store an additional 1,920 spent fuel assemblies in 48 casks. The ISFSI will provide adequate capacity to enable Units 1 and 2 to continue operation until expiration of their respective Operating Licenses in 2013 and 2014.

The ISFSI will be licensed in accordance with the requirements set forth in 10CFR72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste. This Safety Analysis Report has been prepared pursuant to guidelines contained in Regulatory Guide 3.62 (Reference 2). An amendment to the Prairie Island Operating License and Technical Specifications will also be required to provide for cask handling and fuel loading in the Auxiliary Building and spent fuel pool.

Per the requirements of 10CFR72, a Safety Evaluation addressing the impact of the operation of the Prairie Island ISFSI on the Prairie Island Nuclear Generating Plant was completed (Reference 5). This Safety Evaluation, prepared in accordance with the requirements of 10CFR50.59, concludes that operation of the ISFSI will not pose an undue risk to the safe operation of the Prairie Island Nuclear Generating Plant.

1.2 GENERAL DESCRIPTION OF LOCATION

The Prairie Island Nuclear Generating Plant site encompasses about 578 acres and is located within the city limits of Red Wing, Minnesota, in Goodhue County. Northern States Power Company, Minnesota (NSPM[*]) owns most of the land in the site in fee. The U.S. Army Corps of Engineers controls the land that is not owned by NSPM. The Corps has entered into an agreement with NSPM to prevent residential construction on this land for the life of the power plant.

The Prairie Island site is located on a low island terrace associated with the Mississippi flood plain. It is surrounded by the Vermillion River on the west and by the Mississippi River on the east. The site has been evaluated under the criteria of 10CFR100 prior to issuance of an Operating License for each unit (References 3 and 4). The term of each license is 40 years from the date of issuance.

NSP began commercial operation of Prairie Island Nuclear Generating Plant Units 1 and 2, on December 16, 1973, and December 21, 1974, respectively. Westinghouse Electric Corporation designed and supplied the nuclear steam supply system for each unit. Each reactor was originally rated at 1,650 MWt, which is equivalent to approximately 575 MWe (gross), and was uprated in 2010 through a measurement uncertainty recapture power uprate to 1,677 MWt, which is equivalent to approximately 584 MWe (gross) (Reference 8). A complete description of the power plants is contained in the Prairie Island USAR. 01249569

Figure 1.2-1 shows the location of the ISFSI and cask transporter access road in relation to other facilities on the Prairie Island site. The protected area fence surrounding the ISFSI is within the Prairie Island site boundary and exclusion area. The controlled area, which is required by 10CFR72.106 to be established around the ISFSI, corresponds to the site exclusion area boundary. Earthen berms surrounding the ISFSI provide radiological shielding.

[*] Northern States Power Company was incorporated in Minnesota as a wholly owned subsidiary of Xcel Energy, Inc. effective August 18, 2008.

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1.3 GENERAL STORAGE SYSTEM DESCRIPTION

Figure 1.3-1 shows the general arrangement of the ISFSI. Two concrete pads provide for two parallel rows of 12 casks per row on each pad. Both concrete pads will be installed prior to initial operation of the ISFSI in order to minimize future dose and security considerations resulting from staged construction. Casks will be placed on the concrete storage pads in a sequence which will provide for future access to any individual cask.

Each storage cask consists of the following components:

- basket assembly for support of the fuel assemblies
- containment vessel enclosing basket assembly and fuel
- gamma shield
- neutron shield
- outer shell
- weather cover
- pressure monitoring system
- trunnions

A set of reference drawings is presented in Figures 1.3-2 through 1.3-7. The casks are self-supporting cylindrical vessels. Dimensions and design characteristics are shown in Table 1.3-1. Table 1.3-2 is a list of components of the TN-40 cask.

The containment vessel for the TN-40 cask consists of: an inner shell which is a welded, carbon steel cylinder with an integrally-welded, carbon steel bottom closure; a welded flange forging; a flanged and bolted carbon steel lid with bolts; and penetration assemblies with bolts. The overall containment vessel length is 175.0 in. with a wall thickness of 1.5 in. The cylindrical cask cavity has a diameter of 72.0 in. and a length of 163.0 in.

There are two penetrations through the containment vessel, both in the lid: one is for a drain opening and the other is for venting. A double-seal mechanical closure is provided for each penetration. The containment lid is 4.50 in. thick and is fastened to the body by 48 bolts. Double metallic O-ring seals with interspace leakage monitoring are provided for lid closure. To preclude air in-leakage, the cask cavity is pressurized above atmospheric pressure with helium. The interspace between the metallic seals is monitored and pressurized with helium to a higher level than the cavity so that any seal leakage would be into rather than out of the cavity. For additional protection a toruspherical weather cover with a viton O-ring is provided above the lid.

A gamma shield is provided around the walls of the containment vessel by an independent shell of carbon steel which is welded to a bottom shield plate and to the closure flange. The gamma shield completely encloses the containment vessel inner shell and bottom closure.

Neutron shielding is provided by a resin compound surrounding the body. The resin compound is enclosed in long, slender aluminum containers. The array of resin-filled containers is enclosed within a smooth outer steel shell constructed of two half cylinders. In addition to serving as neutron shield containers, the aluminum also provides a conduction path for heat transfer from the cask body to the outer shell. A disk of polypropylene is attached to the cask lid to provide neutron shielding during storage.

The basket structure consists of an assembly of stainless steel cells joined by a proprietary fusion welding process and separated by aluminum and poison plates which form a sandwich panel. The panel consists of two 0.25 in. thick aluminum plates which sandwich a poison plate 0.075 in. thick. The aluminum provides the heat conduction paths from the fuel assemblies to the cask cavity wall. The poison material provides the necessary criticality control. This method of construction forms a very strong honeycomb-like structure of cell liners which provide compartments for 40 fuel assemblies. The open dimension of each cell is 8.05 in. x 8.05 in. which provides a minimum of 1/8 in. clearance around the fuel assemblies. The overall basket length (160 in.) is less than the cask cavity length to allow for thermal expansion and fuel assembly handling.

The cask cavity surfaces and the outer shell have a sprayed metallic coating of Zn/Al for corrosion protection. The external surfaces of the cask are also painted for ease of decontamination. A stainless steel overlay is applied to the O-ring seating surfaces on the body.

Four trunnions are attached to the cask body for lifting and rotation of the cask. Two of the trunnions are located near the top of the body and two near the lower end of the body. The lower trunnions may be used for rotating the unloaded cask between vertical and horizontal positions.

An Equipment Storage Building is located on the ISFSI site. Figure 1.3-8 shows the plan and elevation drawings for this building. The building is a steel frame structure with painted steel walls and roof panels. Two smaller block buildings house the security system equipment and the alarm monitoring equipment.

The ISFSI is surrounded by 8 ft. high security and nuisance chain link fences. In addition, a 17 ft. high earthen berm surrounds the ISFSI, except for the access road opening. The berm is designed for radiation shielding but also visually screens the facility from plant exclusion area boundaries.

The ISFSI is designed as a passive installation with no requirement for active controls. The facility is not continuously staffed. Periodic surveillance of the facility will be performed as described in Section 10.3 and the ISFSI Technical Specifications.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

Transnuclear, Inc. was contracted to provide the casks for use in the ISFSI. Transnuclear is responsible for the design of the casks. Transnuclear subcontracted the fabrication, testing, and delivery of the casks.

Stone & Webster Engineering Corporation was contracted to provide engineering design of the ISFSI, excluding the casks, and to assist in the preparation of the license application, excluding the Security Plan. Stone & Webster also developed a specification used by NSP for cask transport vehicle procurement.

Ederer, Inc. was contracted to provide a cask transport vehicle for use in transporting the casks from the Auxiliary Building to the ISFSI. Ederer is responsible for the fabrication, testing, and delivery of the transport vehicle. The design of the transport vehicle was subcontracted to Nova-Tech Engineering.

Northern States Power Company, Minnesota (NSPM) is responsible for all equipment 01153239
procurement, development of security plans, construction management and construction services using specialty subcontractors, as required.

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1.5 REFERENCES

1. Northern States Power Company, Prairie Island Nuclear Generating Plant Updated Safety Analysis Report, Revision 10, Docket 50-282 (Unit 1) and 50-306 (Unit 2).

2. U.S. Nuclear Regulatory Commission, Regulatory Guide 3.62, Standard Format and Content for the Safety Analysis Report for On-site Storage of Spent Fuel Casks, February 1989.

3. U.S. Atomic Energy Commission, Prairie Island Nuclear Generating Plant, Unit 1, Facility Operating License, License No. DPR-42, August 9, 1973

4. U.S. Atomic Energy Commission, Prairie Island Nuclear Generating Plant, Unit 2, Facility Operating License, License No. DPR-60, October 29, 1974

5. Northern States Power Company, Prairie Island Nuclear Generating Plant Safety Evaluation No. 344, Independent Spent Fuel Storage Installation, May 4, 1993.

6. Safety Evaluation 72-448, TN-40 Cask Weight/Storage Slab Design, May 4, 1996.

7. Safety Evaluation 72-423, TN-40 Cask Lid Alignment Pin Material, October 31, 1995.

8. Letter, T. Wengert (NRC) to M. Schimmel (NSPM), Prairie Island Nuclear Generating Plant, Units 1 and 2 – Amendment Re: Measurement Uncertainty Recapture Power Uprate," August 18, 2010. 01249569

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TABLE 1.3-1
DIMENSIONS AND WEIGHT OF TN-40 CASK

Overall length (with protective cover, in.)	202.00
Outside diameter (in.)	103.50
Cavity Diameter (in.)	72.00
Cavity length (in.)	163.00
Body wall thickness (in.)	9.50
Lid thickness (in.)	10.50
Bottom thickness (in.)	10.25
Resin compound thickness (in.)	4.26
Outer shell thickness (in.)	0.50
Top neutron shield (in.)	4.50
Cask weight:	
Loaded on storage pad (lbs.)(1)	240,690
Loaded on Auxiliary Building crane hook (lbs.)	237,533

(1) Actual weight(s) will vary. Fully loaded cask weight is 244,000 lbs. max. (Reference 6).

TABLE 1.3-2
LIST OF TN-40 CASK COMPONENTS

Item No.	No. Req'd	Shown On Figure No.	Description	Material
1	1	1.3-2	Shell	SA-105 or SA-516, Gr 70 (Note 1)
2	1	1.3-2	Lid	SA-350, Gr LF3
3	1	1.3-2	Inner Containment	SA-203, Gr D or Gr E
4	1	1.3-2	Bottom	SA-105 or SA-516, Gr 70 (Note1)
5	1	1.3-2	Bottom Inner Containment	SA-203, Gr D or Gr E
6	2	1.3-2	Upper Trunnion	SA-105 or SA-266 Class 4
7	2	1.3-2	Lower Trunnion	SA-105 or SA-266 Class 4
8	1	1.3-2	Radial Neutron Shield	Borated Polyester
9	1	1.3-2	Outer Shell	SA-516, Gr 55 (Note 2)
10	1	1.3-2	Protective Cover	SA-516, Gr 55 (Note 3)
11	1	1.3-2	Top Neutron Shield	Polypropylene SA-516, Gr 55 Shell (Note 2)
12	60	1.3-3	Radial n-Shield Box	Aluminum Alloy
13	48	1.3-4	Lid Bolt	SA-320, Gr L43
14	48	1.3-4	Protective Cover Bolt	SA-193, Gr B-8
15	1	1.3-4	Lid Seal	Aluminum Jacketed Double Metallic O-ring
16	1	1.3-4	Protective Cover Seal	Viton O-ring
17	1	1.3-2	Over Pressure Port Cover	SA-240, TP 304
18	1	1.3-2	Over Pressure Port Cover Seal	Aluminum Jacketed Single Metallic O-ring
19	8	1.3-2	Top Neutron Shield Bolt	SA-193, Gr B-8
20	1	1.3-2	Over Pressure Tank	SA-106, Gr B Pipe; SA-234 Caps
21	1	1.3-2	Drain Port Cover	SA-240, TP 304
22	1	1.3-2	Vent Port Cover	SA-240, TP 304
23	2	1.3-2	Drain and Vent Port Cover Seal	Aluminum Jacketed Double Metallic O-ring
24	11	1.3-2	Drain and Vent Port Cover Bolt	SA-193, Gr B-8
25	4	1.3-2	Over Pressure Port Cover Bolts	SA-193, Gr B-8
26	1	1.3-2	Lid Alignment Pin	A-193 Gr B16 (Note 4)
27	40	1.3-7	Fuel Compartment	SA-240, TP 304
28	78	1.3-7	Aluminum Plate	6061, T651
29	78	1.3-7	Poison Plate	boral
30	1	1.3-4	Shield Plate	SA-105 or SA-516, Gr 70

Notes:
Note 1: SA-266 Class 4 is an equivalent substitution for SA-105.
Note 2: or equivalent.
Note 3: or equivalent.
Note 4: A 479 Type 316 or similar low carbon steel is preferred (Reference 7).

Figure Withheld Under 10 CFR 2.390

Figure 1.2-1

ISFSI SITE LOCATION

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

ISFSI SITE LOCATION			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 2	FIG1.2-1
	PAGE. NO.	DATE: 03-28-06	

NORTHERN STATES POWER COMPANY
Xcel Energy
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA

ISFSI SITE PLAN

DRAWN BY:	VLS	REVISION:	3
PAGE. NO.		DATE:	03-28-06

FIG1.3-1

Figure Withheld Under 10 CFR 2.390

Figure 1.3-1

ISFSI SITE PLAN

PRAIRIE ISLAND ISFSI Rev 3 4/94
SAFETY ANALYSIS REPORT

TRANSNUCLEAR, INC.

TN-40 DRY STORAGE CASK LONGITUDINAL SECTION

Figure Withheld Under 10 CFR 2.390

TN-40 DRY STORAGE CASK LONGITUDINAL SECTION			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 8	FIG1.3-2_REV_8
	PAGE. NO.	DATE: 03-28-06	

TRANSNUCLEAR, INC.

TN-40 DRY STORAGE CASK CROSS SECTION

1092-70-2

Figure Withheld Under 10 CFR 2.390

TN-40 DRY STORAGE CASK CROSS SECTION			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 8	FIG1.3-3_REV_8
	PAGE. NO.	DATE: 03-28-06	

FIGURE 1.3-4 REVISION 8 10/99

TRANSNUCLEAR, INC.

TN-40 DRY STORAGE CASK LID ASSEMBLY & DETAILS

D 1092-30-3 5

Figure Withheld Under 10 CFR 2.390

TN-40 DRY STORAGE LID ASSEMBLY & DETAILS			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 8	FIG1.3-4_REV_8
	PAGE. NO.	DATE: 03-28-06	

TRANSNUCLEAR, INC.

Figure Withheld Under 10 CFR 2.390

TN-40 DRY STORAGE CASK PRETECTIVE COVER

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 8	FIG1.3-5_REV_8
	PAGE. NO.	DATE: 03-28-06	

TRANSNUCLEAR, INC.

TN-40 DRY STORAGE CASK BASKET GENERAL ARANGEMENT

SAR

NONE | B | 1042-70-5 | 5

Figure Withheld Under 10 CFR 2.390

TN-40 DRY STORAGE CASK BASKET GENERAL ARANGMENT			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 8	FIG1.3-6_REV_8
	PAGE. NO.	DATE: 03-28-06	

TRANSNUCLEAR, INC.

TN-40 DRY STORAGE CASK BASKET TYPICAL CROSS SECTION

APPROVALS	DATE
M.M.	11 JULY 90
L.C.	11 JULY 90
A.R.	11 JULY 90
M.M.	11 JULY 90
J.T.G.	9 MAR 94

SCALE	SIZE	DWG. NO.	REV.
NONE	B	1042-70-6	9

SAR

Figure Withheld Under 10 CFR 2.390

TN-40 DRY STORAGE CASK BASKET TYPICAL CROSS SECTION

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 8	FIG1.3-7_REV_8
	PAGE. NO.	DATE: 03-29-06	

Figure Withheld Under 10 CFR 2.390

Figure 1.3-8

EQUIPMENT STORAGE BUILDING

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

EQUIPMENT STORAGE BUILDING			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG1.3-8_REV_3
	PAGE. NO.	DATE: 03-28-06	

SECTION 2

SITE CHARACTERISTICS

2.1 GEOGRAPHY AND DEMOGRAPHY OF SITE SELECTED

2.1.1 SITE LOCATION

Information concerning site geography and demography is contained in the Prairie Island USAR (Reference 1). This information is summarized and supplemented below.

The ISFSI is located within the site boundary of the Prairie Island Nuclear Generating Plant, which encompasses about 578 acres. The Prairie Island site is located in the city limits of the City of Red Wing, Minnesota (population 15,134 according to 1990 census figures) on the west bank of the Mississippi River. The ISFSI site and all appurtenant facilities are located in Section 5, T113N, R15W in Goodhue County, Minnesota, at approximately 92° 37.9' west longitude and 44° 37.3' north latitude.

The ground surface near the Prairie Island site is fairly level to slightly rolling, ranging in elevation from 675 ft. to 706 ft. above mean seal level (msl) (1929 adjustment). The surface slopes gradually toward the Mississippi River to the northeast and Vermillion River on the southwest. Normal water level is 674.5 ft.

Steep bluffs run parallel to this stretch of the Mississippi River and rise to an elevation of over 1,000 ft. above mean sea level approximately 1.5 miles northeast and southwest of the site. Northeast and southwest of these bluffs, the ground elevation above mean sea level ranges from 1,000 to 1,200 ft. and is marked by many eroded coulees.

Figure 2.1-1 is a regional map showing the site location. Figure 2.1-2 is an area map showing topography in the site vicinity. The protected area fence is shown to define the ISFSI site area. The controlled area for the ISFSI, as defined in 10CFR72.3, corresponds to the exclusion area of the nuclear station.

2.1.2 SITE DESCRIPTION

Figure 2.1-3, sheets 1-3, are maps showing local topography in the vicinity of the ISFSI. The figures also show the topography in the vicinity of the access road which will be used for transportation of the casks from the Auxiliary Building to the ISFSI.

The overburden materials in the site are sandy alluvial soils. Vegetation in the site area consists of prairie grass and brush, with some isolated stands of trees. Accordingly, there is potential for grass fires and soil erosion in the vicinity of the ISFSI, although the ISFSI itself is covered with a gravel surface. Additional information concerning soil and vegetative cover is contained in the Environmental Report filed in conjunction with the ISFSI license application.

The Prairie Island Nuclear Generating Plant site and exclusion area is owned in fee by Northern States Power Company, Minnesota with the exception of areas controlled by the U.S. Army Corps of Engineers. The protected area of the ISFSI and the access road connecting the ISFSI and Auxiliary Building is on land owned by NSPM. 01153239

2.1.3 POPULATION DISTRIBUTION AND TRENDS

The nearest population centers are Eagan (1990 population of 47,409) 26 miles northwest of the site; Minneapolis – St. Paul metropolitan area (1990 population of 2,407,090), 30 miles northwest of the site; and Rochester (1990 population of 70,745), 41 miles south of the site. No other population centers with more than 25,000 people lie within 50 miles of the site. Table 2.1-1 shows the estimated 1998 population distribution within a 50 mile radius of the plant. The Environmental Report submitted in conjunction with this license application provides additional information concerning population projections.

2.1.4 USES OF NEARBY LAND AND WATERS

Goodhue County, in which the site is located, and the adjacent counties of Dakota and Pierce (in Wisconsin) are predominantly rural. Dairy products and live stock account for most of the three-county farm products with field crops and vegetables accounting for most of the remainder. Principal crops are corn, oats, hay, soybeans and barley.

The region within a radius of five miles of the site is devoted almost exclusively to agricultural pursuits. Principal crops include soybeans, corn, oats, hay and some cannery crops at about four miles from the plant site. The nearest dairy farm is located more than two miles southwest of the plant site. Some beef cattle are raised approximately two miles southwest of the site. Cattle are on pasture from early June to late September or early October. During the winter, cows are fed on locally produced hay and silage. Beyond the site boundary and within a one-mile radius of the plant, there are approximately 30 permanent residences or summer cottages. The closest occupied offsite residence is approximately 0.45 mi. northwest of the ISFSI site.

Approximately one mile northwest of the ISFSI site, the Mdewakanton Sioux Indian community owns and operates a combination resort hotel/bingo/casino gambling facility. Another business facility consisting of a gasoline station, convenience store and lounge is located about two miles west-northwest of the ISFSI site.

2.2 NEARBY INDUSTRIAL, TRANSPORTATION AND MILITARY FACILITIES

The following industrial facilities (workforce population greater than 30) are located within 5 miles of the ISFSI site (Reference 2):

- Advertising Unlimited, 3-4 miles South
- DB Industries, 3-4 miles SSE
- IRC Industries, 3-4 miles SSE
- Josten's, 3-4 miles SSE
- Dayco PTI, 3-4 miles SSE
- Protein Technologies, 4-5 miles SSE
- RAM Center, Inc., 3-4 miles South
- Red Wing Shoe Company, 3-4 miles SSE
- Republican Eagle (Manufacturing), 4-5 miles SSE
- Riedell Shoe Company, 3-4 miles SSE
- Riviera Cabinets, 3-4 miles SSE
- Central Research Laboratory, 3-4 miles South

No activities at the above facilities present a hazard to the safe operation of the ISFSI.

No military installations are within 5 miles of the ISFSI site. No large natural gas pipelines pass close to the ISFSI site. Transportation activities which could potentially have an effect on the ISFSI are described below.

The Red Wing airport is located about 7 miles ESE of the ISFSI site in Bay City, Wisconsin, at 44° 35' 15" north, 92° 29' 30" west. It is at elevation 781 ft. msl and has one asphalt paved runway which is 4,000 ft. long by 75 ft. wide and an adjacent taxiway. The runway designation is 09/27 with a 50/1 glide path. There are 30 aircraft based at the airport.

Railroad traffic occurs on the following railroads, which pass within five miles of the site:

- CP Line Railroad (Soo Line) runs across the southwest portion of the Prairie Island Nuclear Generating Plant site.
- Burlington Northern Railroad runs within 2 miles east of the ISFSI in Wisconsin.

Truck traffic occurs on Minnesota State Highway 61, which runs within 2 ½ miles of the ISFSI site to the south. In addition, a number of country roads and truck highways are located within 5 miles of the site.

Barge traffic occurs on the Mississippi River, which flows in its main channel no closer than ½ mile from the ISFSI site. A transportation accident involving a barge explosion has been postulated as having the worst case impact on the safe operation of the ISFSI. This analysis is considered to conservatively bound the impacts which might be postulated for railroad, highway or barge accidents on the routes described above.

A munitions explosion on a barge at a mid-channel location opposite the plant along the Minnesota-Wisconsin line was analyzed in the Prairie Island USAR. The explosion was postulated to occur at a distance of 2,600 ft. from the existing Prairie Island control room, at a point nearly due east from it. An overestimate of the blast effect is given by assuming impacts calculated at the control room would occur at the ISFSI site, which is farther away from mid-channel.

The size and nature of a hypothetical cargo explosion was postulated in the Prairie Island USAR. It was conservatively assumed that an explosion of a jumbo barge took place. A jumbo barge (195 ft. long, 35 ft. wide, 8 ½ ft. draft), is the largest hauler of dry cargo. It was assumed that the barge was completely laden with 1,400 tons of TNT. It was assumed that this cargo detonated in mid-channel directly opposite the plant. The resulting 1.4 kiloton explosion would be comparable to the Texas City disaster which resulted from detonation of 2 to 4 kilotons of equivalent explosive. Surface detonation of 1.4 kilotons was calculated to result in a peak overpressure of 2 ¼ psi at 2,600 ft. distance, plus minor dynamic pressure due to 78 mph transient wind. Cask design for external pressure is discussed in Section 3.2.5.3.4.

2.3 METEOROLOGY

2.3.1 REGIONAL CLIMATOLOGY

Regional climatology is described in the Prairie Island USAR, Section 2.3. Information from the USAR concerning climatology is summarized and supplemented below.

The climate of the site region is basically continental and influenced by the general storms which move eastward along the northern tier of the United States. The geographical location results in frequent changes in weather systems as polar and tropical air masses alternate. Climatic characteristics are illustrated in Figure 2.3-1 which shows average and extreme temperatures, precipitation, and extreme winds at Minneapolis, Minnesota. Rainfall averages about 25 in. per year, with 65% falling in the months of May through September. Figure 2.3-1, based on records through 1969, shows maximum rainfall during 24 hours was 7.80 inches in July 1892. This record was exceeded in July 1987, when 10.0 in. of rain fell within a 6-hour period (Reference 3). Snowfall averages about 44 in. per year with a maximum of 19.9 in. in 24 hours in January 1982.

Minnesota lies to the north of the principal tornado belt in the United States. During the seven-year period 1963-1969, 15 tornadoes and 21 funnel clouds were reported in the 1° square centered about the site and encompassing nearby Wisconsin and Minnesota. Over 90% of these occurrences were reported in the months of May, June and July. The probability of a tornado striking a point in the 1° square has been calculated to be between 5.8 x 10^{-3}/year and 3.8 x 10^{-3}/year with 95% confidence limits. At these confidence limits, the recurrence intervals are between 170 and 260 years. Section 2.3.3.2 of the Prairie Island USAR specifies that the design basis tornado results in a wind speed of 300 mph with a forward progression of 60 mph.

The site area is subject to 30-40 thunderstorm days per year. The number of lightning flashes to ground per year per square mile is estimated to be between 0.05 and 0.8 times the number of thunderstorm days per year (Reference 4).

2.3.2 LOCAL METEOROLOGY

2.3.2.1 DATA SOURCE

Meteorology in the Prairie Island Nuclear Generating Plant site area was evaluated as part of its Operating License application review. This evaluation was based on 25 months of site data from May 1968 to May 1970. The pre-operational meteorological data program is discussed in Section 2.8 of the Prairie Island FSAR (Reference 5). One year of supplemental data for the period June 1, 1971 through May 31, 1972, using NRC recommended delta-T stability classification, is presented in Appendix H of the Prairie Island USAR.

The Prairie Island Nuclear Generating Plant was provided with a new 60 meter meteorological tower in October 1982. Wind speed, direction, and temperature difference instrumentation is located at approximately 10 meters and at the 60 meter level. In addition, temperature and rainfall instruments are installed. The tower is located about 1,500 ft. north of the ISFSI site.

Recent joint frequency distributions for the 10 and 60 meter tower elevation for the period January 1, 1991 through December 31, 1991 are presented in Tables 2.3-1 through 2.3-16 of the Prairie Island USAR. The distributions are for Stability Classes A through G as defined in Regulatory Guide 1.23.

Meteorological data is used to compute dispersion *(χ/Q)* and deposition *(D/Q)* factors and is used in the dose assessment of airborne releases in accordance with the requirements of the Offsite Dose Calculation Manual. Wind speed, direction and atmosphere stability class are averaged over the release period and serve as inputs to a dispersion model. Stability class is determined using temperature difference measurements between the 10 meter elevation and the 60 meter level. Tables 2.3-18 and 2.3-19 of the Prairie Island USAR contain a tabulation of annual average dispersion and deposition factors.

2.3.2.2 TOPOGRAPHY

Figure 2.3-2 is a map showing detailed topographic features within an 8 kilometer (5 mile) radius of the ISFSI site.

2.3.3 ONSITE METEOROLOGICAL MEASUREMENT PROGRAM

The meteorological tower is provided with instrumentation at the 10 meter and 60 meter elevations. The location of the tower is shown on Figure 1.2-1. The purpose of this tower is to collect meteorological data for use in the Prairie Island Nuclear Generating Plant Emergency Response Program. Data from the meteorological tower will not be used to estimate offsite concentrations of airborne effluents from the ISFSI, as no credible mechanism for the release of airborne effluents has been postulated.

The potential impact of spent fuel storage at the site of the ISFSI on meteorological measurements at the meteorological tower have been investigated. The investigation considered possible changes in air temperature, wind speed and direction, and turbulent wind fluctuations that could be induced at the tower location by heat dissipation from the storage casks.

It has been concluded that changes in meteorological parameters, as measured at the tower, will generally be insignificant. The thermal energy contributed to surface layer air by the spent fuel will be of the same order of magnitude as natural energy inputs from solar radiation. Turbulent energy production resulting from the facility heat input will normally be much less than natural energy production from boundary layer wind shear at the earth’s surface. Accordingly, effects on air motion will be small. Temperature changes in air reaching the meteorological tower are estimated to be a maximum of 1.0 deg K and usually less than 0.25 deg K.

There are approximately 60 hours per year during which winds are directed from the location of the fuel storage facility to the meteorological tower, and during which ambient wind speeds are low with strong stability (Classes E, F and G). During these hours, enhanced turbulence (larger values of sigma theta) and increased temperature (up to 1.0 deg K increase) may be observed at the 10m level of the tower. These effects could result in an inferred stability classification that is more unstable than general ambient conditions. Because of its lower frequency of occurrence, this potential effect is considered to represent only a minor influence on measured site climatology. However, appropriate plant personnel have been made aware of the possibility of misleading indications of site stability during these specific meteorological conditions.

All evaluations assumed maximum heat dissipation from the spent fuel storage facility, and all heat transfer by conduction to the surface air. Both of these assumptions represent worst-case situations, and will seldom if ever be experienced. Thus, it is believed that the analyzed impacts are likely to be overestimates. Under normal operation conditions with typical meteorology, impacts on meteorological measurements will be less than natural variations due to land surface irregularities at the plant site and typical errors in meteorological measurement.

2.3.4 DIFFUSION ESTIMATES

2.3.4.1 BASIS

No routine or accidental releases are planned or postulated as a result of ISFSI operation. Nevertheless, 1-to-2 hour χ/Q values have been calculated which can be used to estimate radiological doses from the accidental release analyzed in Section 8.2.9.

2.3.4.2 CALCULATIONS

Table 2.3-1 provides a tabulation of short term χ/Q at the site boundary in each of 16 sectors. Table 2.3-2 provides χ/Q values at varying downwind distances. These values were calculated based on the methodology outlined in Regulatory Guide 1.145 (Reference 6). The values are based on an assumed ground level release with 1.0 meter/second wind speed under conditions of moderate stability.

Plume meander under stable atmospheric conditions (Stability Classes E, F and G) is considered in this methodology. No credit for building wake effect was included, due to the distance of the casks from any large structure.

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2.4 HYDROLOGY

Hydrology of the Prairie Island Nuclear Generating Plant site area was evaluated as part of its Operating License application review. Information concerning site area hydrology is contained in the Prairie Island USAR, Section 2.4, and Appendix E. Pertinent information is summarized and supplemented below.

2.4.1 SURFACE WATER

The principal surface waters in the vicinity of the site are the Mississippi River, Sturgeon Lake, the Vermillion River and the Cannon River. The levels of the Mississippi River and Sturgeon Lake are controlled by Lock and Dam Number 3 which is located approximately 1 ½ miles downstream from the plant. The normal pool upstream from the dam is at elevation 674.5 ft. The locations of the major streams and gaging stations are shown on Figure 2.4-1. The Vermillion River enters the main stream of the Mississippi below the dam.

There are no withdrawals of river water for supply of city water for at least 300 miles downstream from the site. Minor withdrawals of river water for irrigation purposes do occur, the nearest being 53 miles downstream.

Stream flow records are available for the Mississippi River at Prescott and at Winona, and for the Cannon River at Welch. Figure 2.4-2 shows flow duration curves for the two stations on the Mississippi River.

Table 2.4-1 of the Prairie Island USAR summarizes the consecutive-day low-flow characteristics of the Mississippi River at Prescott. The discharge characteristics of the Mississippi and Cannon Rivers are summarized in Table 2.4-2 of the Prairie Island USAR.

Siting evaluation factors set forth in 10CFR72.90(f) specify that an ISFSI must be sited to avoid to the extent possible the long term and short term adverse impacts associated with the occupancy and modification of floodplains. Floodplains are defined in 10CFR72.3 to mean the lowland and relatively flat areas adjoining inland waters, subject to a 1% or greater chance of flooding in any given year. The 100-year flood level at Prairie Island has been determined to be 687.7 ft. above mean sea level (Reference 7).

The 1965 flood, which is the highest on record, has a recurrence interval of 150 years. The peak stage at Lock and Dam Number 3 during this flood was about 687.8 ft. above mean sea level. A flood having a 1,000-year recurrence interval would have a peak stage of about 693.5 ft. at Lock and Dam Number 3, and a discharge of about 335,000 cfs.

A stream profile of the section of the river near the site is presented in Figure 2.4-3 showing profiles of the highest and lowest flow of record. An approximate longitudinal and land profile of the Prairie Island site is also shown in Figure 2.4-3.

A study to determine the magnitude of the probable maximum flood has been done. The probable maximum flood is the hypothetical flood that would result if all the factors that contribute to generation of the flood were to concurrently reach their most critical values that could occur. The probable maximum flood is derived from hydrometeorological and hydrological studies and is independent of flood frequency. It is the estimate of the boundary between possible floods and impossible floods. Therefore, it would have a return period approaching infinity and a probability of occurrence in any particular year approaching zero.

The maximum discharge calculated for the probable maximum flood was determined to be 910,300 cfs, with a corresponding peak stage of elevation of 703.6 ft. The flood would result from meteorological conditions which could occur in the spring and could reach maximum river level in about 12 days. It was estimated that the flood stage would remain above elevation 695 ft. for approximately 13 days. The detailed results of the study are presented in Appendix F of the Prairie Island USAR. Wind generated waves would be of maximum height when the wind is from east to west in the direction of the circulating water intake canal. With a persistent wind speed of 45 mph the height of the significant wave would be less than 1.8 ft. (crest to trough). The maximum 1% wave height, consistent with the highest significant wave, is estimated to be less than 3.10 ft. Significant and maximum 1% wave height and run-up are defined in "Shore Protection and Planning and Design," Technical Report No. 4, 3rd edition, 1966, U.S. Army Corps of Engineers Coastal Engineering Research Center. If the conservative assumption is made that run-up equals the approaching wave height, then the maximum water level would be 706.7 ft. above mean sea level.

Lock and Dam Number 2 is located 17 miles upstream of the plant site. The dam consists of a 3,250-foot long dike, two single-lift locks with chambers 110 ft. x 600 ft. and 110 ft. x 500 ft., and a spillway section of 20-30 foot-wide tainter gates. The difference in normal pool elevations is 12.2 ft. An accident to the dam could result in a sudden release of water, temporarily producing the effect of a flood in the river channel downstream of the dam.

Flood-like flows could be produced only by a major failure to the spillway section with initial upper and lower pools at their normal elevations. Even then, the storage effect of the lower channel basin and the resulting loss of head in the upper reservoir will greatly attenuate flooding effects at the ISFSI site.

There is no flood hazard resulting from a dam break at Lock and Dam Number 2. This conclusion was substantiated by determining stable water level elevations at Lock and Dam Number 3 resulting from sustained flow with the loss of 10 tainter gates at Lock and Dam Number 2. Sustained flow will maintain the upper pool elevation at Lock and Dam Number 3, and will provide the volume of water needed in the lower pool to produce the maximum pool level consistent with steady flow supplied through 10 spillway bays. The flow resulting from these postulated extreme conditions would produce a river level at an elevation of 684.5 ft. in the lower pool at Lock and Dam Number 2 with a corresponding level in the upper pool at Lock and Dam Number 3 of 676.5 ft.

2.4.2 GROUND WATER

Regionally, the movement of ground water is toward the Mississippi River and its main tributaries. The ground water table slopes from the higher glaciated bedrock areas toward these surface streams, generally at low gradients. Ground water enters the river valley from along the base of the bordering bedrock bluffs in the form of springs or as subsurface flow.

Beneath the flood plains and low terraces which border the Mississippi River, ground water levels closely coincide with the elevations of the river surface, and vary in accordance with river fluctuations. The average ground water gradient in these bottomlands is downstream, and essentially parallel to the stream gradient.

Pool elevation on the Mississippi River adjacent to the site is controlled by Lock and Dam Number 3. The Vermillion River bypasses the dam and therefore is not directly controlled by it. Elevations on the Vermillion River and connected lakes are therefore lower than the Mississippi River and the ground water table slopes southwestward between the two rivers. Due to the permeable nature of the sandy alluvial soils forming Prairie Island, the ground water table responds quickly to changes in river stage.

There is only minor usage of ground water for domestic, agricultural and irrigation purposes near the site or immediately downstream. A deep well believed to penetrate bedrock aquifers exists at Lock and Dam Number 3. The nearest ground water consumption of important magnitude is in the town of Red Wing, six miles downstream. This community derives its water supply from four deep wells which penetrate sandstone aquifers of the Dresbach and Hinckley formations. The wells pump from depths of 400 to 730 ft., and each well is capable of providing the municipal requirements of about 1,400 gpm. A high degree of hardness is characteristic of the water from these wells.

Several industries in the Red Wing area also utilize ground water in quantities exceeding the municipal consumption, and derive their supplies principally from the bedrock aquifers. Total well production from the bedrock at Red Wing probably exceeds 3,000 gpm, and fairly large quantities may also be extracted from the alluvium for certain industrial uses.

Communities further downstream from the plant site which supply their water needs from wells in bedrock are Lake City, 25 river miles downstream, and Wabasha, 37 miles downstream.

A survey of 58 wells in the vicinity of the plant site is summarized in Table 2.4-3 of the Prairie Island USAR. The location of the wells and an outline of the survey area are shown in Figure 2.4-4. Dispersion of effluents entering the ground water system from the plant would take place principally in the upper portion of the saturated zone of the river alluvium. Due to the numerous surface waterways in the vicinity of the site, the majority of effluents would permanently leave the ground water environment and would mix with surface waters at the borders of Prairie Island.

The coefficient of horizontal permeability for the shallow aquifer beneath the site is 1,000 to 4,000 gallons/day/sq. ft. Vertical permeability is about 30% of horizontal permeability. Due to the relatively permeable nature of the soils at the site a path exists to the deeper bedrock aquifers directly below the site. Silt deposits lining the bottom of the river channels, however, have virtually eliminated exposure of the bedrock aquifers to the main stream of the river. There is no reason to believe effluents will penetrate to the depth of bedrock aquifers. It is very unlikely that significant amounts of effluents could succeed in reaching shallow wells at Red Wing by way of continuous ground water paths.

Any possible recharging of the shallow aquifer at Red Wing by river water would be expected to entail dilution of the river water with stored ground water and other sources, time decay of activity, and removal of activities other than tritium by filtration through the soil.

The annual average dilution factor for the Mississippi River at Red Wing considering substantial use of cooling water is about 18. A minimum factor is 3.3 based on the 7-day average low flow having a once-in-ten-year probability of recurrence. Credit is not taken for the relatively minor flows contributed between the site and Red Wing by the Vermillion and Cannon Rivers.

The overall dilution factor before release to the Mississippi River is expected to result in less than 0.1 MPS discharge to the river. Therefore, an overall minimum dilution factor of at least 33 would occur at Red Wing.

2.5 GEOLOGY AND SEISMOLOGY

2.5.1 BASIC GEOLOGIC AND SEISMIC INFORMATION

The geology and seismology of the Prairie Island Nuclear Generating Plant site area was evaluated as part of its Operating License Application review. Information concerning geology and seismology is contained in the Prairie Island USAR, Sections 2.5 and 2.6, and Appendix E, and is summarized and supplemented below.

2.5.1.1 STORAGE SITE GEOMORPHOLOGY

The Prairie Island site is located on a low island terrace associated with the Mississippi River flood plain. It is separated from other parts of the lowland by the Vermillion River on the west, and by the Mississippi on the east. Ground surface elevations range from approximately 675 to 706 ft. Lowland areas near the site are forested or covered by swamp vegetation.

The Mississippi River flood plain in this area is confined within a valley about 3 miles wide. Rocky bluffs and heavily forested slopes rise abruptly from both sides of the valley to a height of about 300 ft. The uplands immediately surrounding the valley reach elevations ranging from approximately 1,000 to 1,200 ft. They are deeply trenched by numerous streams emptying into the Mississippi River.

The overburden materials at the site are permeable sandy alluvial soils which were deposited as glacial outwash and as recent river sedimentation. Preliminary borings made prior to licensing of the Prairie Island Nuclear Generating Plant, such as that shown in Figure 2.5-1 and other borings included in Appendix E of the Prairie Island USAR, have indicated that the overburden soils at the site vary from 158 to 185 ft. thick.

The uppermost bedrock unit at the site is sandstone and is believed to be part of the Franconia formation (See Table 2.5-1 of Prairie Island USAR). Its thickness at this location is unknown, but would be much less than 180 ft., the total measured thickness of the Franconia formation in complete sections. Underneath the Franconia formation are several hundred feet of lower Cambrian and Precambrian sandstone with minor shale horizons.

2.5.1.2 GEOLOGIC HISTORY OF STORAGE SITE AND SURROUNDING REGION

Precambrian granite, gneiss, schist, and volcanics comprise the oldest bedrock in the Minnesota-Wisconsin region. The basement rock is overlain by as much as 800 ft. of Paleozoic sandstone, shale and dolomite. Younger formations originally present in the region have been removed by erosion, and an irregular topography has been developed on the exposed bedrock surface. Except for local areas in southeastern Minnesota and parts of Wisconsin, bedrock is concealed under 100 to 300 ft. of Pleistocene glacial drift. In contrast, the extreme southeastern tip of Minnesota, including the site vicinity, is covered by only a thin veneer of drift. It is therefore considered a part of the “driftless” area commonly referred to by glacial geologists. In this driftless area of Minnesota and central and southwestern Wisconsin, the unconsolidated materials consist primarily of loess, recent alluvium and residual soil. Drainage in the region is controlled by the Mississippi River. The Mississippi River originated as an outlet for early glacial meltwaters. Its major present day tributaries were developed by the draining of glacial lakes at the close of the Pleistocene.

A geologic column showing the thickness and age relationships of the various bedrock units and surficial deposits of the region is presented on the generalized stratigraphic column presented in Table 2.5-1 of the Prairie Island USAR (Reference 1).

The regional extent of the consolidated strata is shown in Figure 2.5-2.

2.5.1.3 SPECIFIC STRUCTURAL FEATURES OF SIGNIFICANCE

The dominant structural feature in Southeastern Minnesota and adjacent areas of Wisconsin and Iowa is the Keweenawan Basin. The basin was formed in early Precambrian time and extended from Lake Superior into Iowa. It provided a site for the deposition of thick sequences of later Precambrian and Paleozoic strata consisting of volcanics and sediments. These beds were gently warped by subsequent compressive forces into several subordinate structures. A large basin in the Paleozoic rocks extended northward from Iowa into the southeastern corner of Minnesota. This basin is separated from a smaller basin in the Twin Cities area by the Afton-Hudson anticline. The anticline begins at Farmington and trends northeastward through Hastings, Minnesota and Hudson Wisconsin. A syncline lies to the east of the structure in the vicinity of River Falls, Wisconsin. Near the extreme southeastern corner of Minnesota, a second anticline extends from Rochester through Red Wing and is postulated to extend a short distance into Wisconsin. The site is located on the west limb of the Red Wing anticline, as evidenced by a gentle westward dip of the bedrock.

Several major faults in the Minnesota-Wisconsin region have been inferred from geophysical surveys. The principal movements along these faults, which amounted to thousands of ft., appear to have been restricted to Precambrian time. The Douglas fault and the Lake Owen fault penetrate Precambrian rocks along the north and south sides of the Keweenawan Basin, respectively.

A southern extension of the Lake Owen fault, which is known as the Hastings fault, trends southwest near the city of Hastings, about 13 miles northwest of the site. Minor activity occurred along the Hastings fault during both Precambrian and Paleozoic times. Other minor movements occurred in the Paleozoic strata 6 miles southeast of the site near the city of Red Wing, and approximately 20 miles northeast of the site in the River Falls syncline near Waverly, Wisconsin.

There is no evidence of recent activity along any of the known fault zones in the Minnesota-Wisconsin region.

The locations of these structural features are shown in Figure 2.5-2. Regional geology is further shown in Figures 2.5-3 and 2.5-4.

2.5.1.4 LARGE SCALE GEOLOGIC MAP

The locations of the structural features discussed in Section 2.5.1.3 are shown in Figure 2.5-2.

2.5.1.5 PLOT PLAN AND SITE INVESTIGATIONS

A number of borings were made as part of the development of the Prairie Island Nuclear Generating Plant Operating License application. The boring logs and site maps showing the boring locations are contained in the Prairie Island USAR, Appendix E.

Additional field investigations at the ISFSI site were made in June 1991. The field program consisted of drilling 9 borings (B-8 through B-16) located roughly on a grid pattern as shown on Figure 2.5-5. The borings were located utilizing four corner control points. From the control points the borings were located within the blocked out area of interest using a transit and tape. The elevations at the tops of the borings were surveyed in from a point of known elevation.

The drilling and surveying of boring locations and elevations were performed by Twin City Testing, Inc. The borings were drilled utilizing a Central Mining Equipment, Model 55, truck-mounted drilling rig. Borings were advanced using 6 in. diameter, hollow stem augers from the surface down to the intersection with the ground water table. At this point a switch to mud-rotary drilling was undertaken. Borings were advanced below the water table utilizing a 2 15/16 in. diameter tricone bit and gel/water mud. Boring B-8, B-10, B-12, B-14 and B-16 were drilled to 61 ft., and borings B-9, B-11, B-13 and B-15 were drilled to 51 ft. Samples were collected at 5 or 10 foot intervals with a standard split spoon sampler driven with a 140 lb. hammer following the ASTM D1586 standard penetration test (SPT) procedure (Reference 8). Blow counts, sample descriptions and water level elevations were recorded, and samples were collected for testing. Boring logs prepared for the nine borings are shown in Appendix 2A.

Fourteen soil samples, representing the various soil types encountered, were tested by Twin City Testing, Inc., for grain size distribution using ASTM D422 (Reference 9) procedures. The results of these tests are set forth in Appendix 2B.

2.5.1.6 GEOLOGIC PROFILES

Based on the information gathered from the 9 borings and associated samples obtained in June 1991, the following stratigraphic sequence is inferred to underlay the site. The site is covered with a layer of reddish brown silty sand from 0 to 12 ft in depth. Underlying the top layer is a uniform light brown fine sand with intermittent thin gravel seams. The light brown sand is found between depths of 12 ft and 44 to 46 ft. Below the light brown fine sand gravelly sand exists. The gravelly sand contains gravel pieces to 1 3/4 in. in size and is found between depths of 44 to 46 ft and 61 ft, the depth of the deepest boring. This sequence of soil layers was found in all 9 borings.

The density of the soils as measured by Standard Penetration Test (SPT) blow counts generally increases with depth from the surface. Penetration resistance down to the intersection with the marked increase in gravel content (approximately elevation 44 to 46 ft.) ranges from 3 to 28 blows/ft. Below this elevation the penetration resistance generally increases and ranges from 8 to 45 blows/ft.

The depth to ground water table was measured in each of the borings through hollow steam augers prior to switching to mud-rotary drilling. The ground water was measured at depths from 16.0 to 20.7 ft. below ground surface.

2.5.1.7 PLAN AND PROFILE DRAWINGS

Figure 2.5-6 is a site grading plan. Figure 2.5-7 shows site sections and details.

2.5.1.8 LOCAL GEOLOGIC FEATURES AFFECTING SITE LOCATION

There are no zones of alteration, deformation zones, previous earthquakes, zones of structural weakness or unstable rock formations which will affect storage cask placement.

2.5.1.9 SITE GROUNDWATER CONDITIONS

The depth to the unconfined water table was measured in each of the borings taken in June 1991 through hollow stem augers prior to switching to mud-rotary drilling. The ground water table was measured at depths from 16.0 ft. to 20.7 ft.

Site groundwater conditions area discussed in Section 2.4.2. Additional information is provided in the Prairie Island USAR.

2.5.1.10 GEOPHYSICAL SURVEYS AND STUDIES

The Prairie Island USAR, Section 2.5 and Appendix E, summarize geophysical surveys and studies which have been performed to evaluate the stratigraphic structure and bedrock in the Prairie Island site region.

2.5.1.11 SOIL AND ROCK PROPERTIES

Soil and rock properties are as shown on the boring logs in Appendix 2A.

2.5.1.12 ANALYSIS TECHNIQUES

Based on inspection of the gradation curves from the site, some of the soils are expected to be frost susceptible. This is not a significant consideration due to the depth to the water table. To avoid any potential problems footings and slabs will be founded below the anticipated frost depth or on non-frost susceptible fill extending below frost depth.

Analyses were performed to evaluate allowable bearing stresses for the ISFSI slabs and to estimate settlement associated with anticipated loading of the slabs. The allowable net bearing stress for the slabs was based on a depth below grade of 3 ft. It was also assumed that the water table might be at or above ground surface during flooding, since this condition reduces bearing capacity, and that topsoil with significant organic or clay content will be removed. Given the SPT blow counts shown in the boring logs in Appendix 2A, a net allowable bearing stress of 4 ksf is expected to be acceptable if settlement is not a major consideration.

A settlement calculation was performed based on the same conditions, resulting in an estimated settlement of roughly 1 to 1.5 in. for each 1 ksf of net bearing stress up to 4 ksf allowable. It is expected that this rate of settlement will be acceptable since the anticipated bearing stress is near 1 ksf.

Settlement is expected to occur very rapidly upon loading of the slab because sandy soils such as occur at the site are typically relatively free-draining (Reference 10). Flexibility of the slabs was considered, since loaded portions will tend to settle more than adjacent unloaded portions.

2.5.2 VIBRATING GROUND MOTION

For sites that have been evaluated under the criteria of 10CFR100, Appendix A, 10CFR72.102 provides that the design earthquakes (DE) must be equivalent to the safe shutdown earthquake (SSE) of the nuclear power plant. The appropriate response spectrum must also be specified.

The Prairie Island USAR, Section 2.6 and Appendix E, provide information and analyses concerning seismology, including discussions of earthquake history and probability. The SSE has been calculated to be 0.12g maximum ground acceleration. Figure 2.5-8 shows the ground level response spectra associated with the SSE.

Appendix E of the USAR provides detailed discussions of procedures used to determine the SSE. The USAR also summarizes seismic wave propagation and soil structures interaction analyses performed for the Prairie Island site.

2.5.3 SURFACE FAULTING

Section 2.5.1.3 provides a discussion of faults in the site area. Additional analysis is presented in the Prairie Island USAR, Appendix E. The probability of surface faulting in the site area is extremely low.

2.5.4 STABILITY OF SUBSURFACE MATERIALS

Requirements set forth in 10CFR72.102 provide that sites other than bedrock sites must be evaluated for their liquefaction potential or other soil instability due to vibratory ground motion. Site specific investigations and laboratory analyses must show that soil conditions are adequate for the proposed foundation loading.

A liquefaction analysis was performed on the basis of standard penetration test N values (blow counts) obtained during the June 1991 site investigation (Reference 12). The analysis is based on a normal water table elevation of approximately 18 ft. below the ground surface, since it is not intended to evaluate flood and earthquake conditions together. The liquefaction analysis was performed using a procedure developed by Seed, Seed and Idriss (Reference 11). Following this procedure, the stress ratio, which is the ratio of horizontal shear stress to vertical effective stress, is calculated at various depths for the anticipated horizontal acceleration N values which would allow liquefaction, given the expected stress ratio, is then determined. A curve of these critical N values versus depth (Figure 2.5-9) was developed for a maximum horizontal ground acceleration of 0.12g for a Richter magnitude 5 earthquake. This curve is compared with the N value profile for each of the borings as shown in Figure 2.5-9. The comparison shows that two marginally liquefiable zones were found in the borings at depths of 20 ft. and 35 ft. throughout the site.

The potentially liquefiable zone at 20 ft. is only 2 ft. below the groundwater table and any pore water pressure buildup due to strong ground shaking would dissipate quickly due to the short drainage path to the top of the ground water. Therefore, there is little chance of this layer liquefying. The potentially liquefiable zone at 35 ft. was analyzed based on the ratio of the depth of the unliquefiable surface layer to the thickness of the liquefiable sand layer. In comparison to numerous similar sites noted in Reference 11, if the marginally liquefiable (at the 35 ft. depth) is taken to be 5ft. thick, based on the low blow count (N) values shown in Figure 2.5-9, then the ratio between layer depth and layer thickness is 7. According to observations mentioned in Reference 11 surface effects are not expected at ratios above 1.

The studies from Reference 11 also state that no evidence of liquefaction was found when nonliquefiable surface layer has a thickness of greater than 9 ft. For these reasons, even if liquefaction of the sand layer at 35 ft. were to occur, it is anticipated that the effects at the surface will be minor to non-existent and will have no effect on the stability of the ISFSI slabs.

2.5.5 SLOPE STABILITY

The ISFSI site within the perimeter fence is located on both cut and fill material. The berms surrounding the site are constructed of earth fill material. To limit the site area and amount of required fill, a slope of one horizontal to one vertical is used for the berms.

The berms are reinforced with geofabric. Erosion control material and natural vegetation give the berms a natural appearance.

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2.6 REFERENCES

1. Northern States Power Company, Prairie Island Nuclear Generating Plant Updated Safety Analysis Report, Revision 18, Docket 50-282 (Unit 1) and 50-306 (Unit 2).

2. Earth Tech, Evacuation Time Estimates for the Prairie Island Nuclear Generating Plant Plume Exposure Pathway Emergency Planning Zone, December 1997.

3. National Oceanic and Atmospheric Administration, Local Climatological Data Minneapolis – St. Paul, MN, July 1987.

4. Uman, Martin A., Understanding Lightning, Bek Technical Publications, Inc., Carnegie, PA 15106.

5. Northern States Power Company, Prairie Island Nuclear Generating Plant Final Safety Analysis Report, May 3, 1974.

6. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants, Revision 1, November 1982.

7. Northern States Power Company, Operating Procedure AB-4, Flood, Revision 13.

8. American Society for Testing and Materials, ASTM D1586, Penetration Test and Split-Barrel Sampling of Soils, 1984.

9. American Society for Testing and Materials, ASTM D422, Method for Particle-Size Analysis of Soils, 1963.

10. Peck, Hanson, and Thornburn, Foundation Engineering, 2nd Edition, John Wiley & Sons, 1974.

11. Committee on Earthquake Engineering, Commission on Engineering and Technical Systems, National Research Council, Liquefaction of Soils During Earthquakes, National Academy Press, Washington, D.C., 1985.

12. Estimation of Liquefaction Potential, SWEC Calculation No. 1291154-G-2, Rev. 1, July 20, 1991.

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TABLE 2.1-1
ESTIMATED 1998 POPULATION DISTRIBUTION WITHIN THE PRAIRIE ISLAND EMERGENCY PLANNING ZONE

Sector	Direction	0-1	1-2	2-3	3-4	4-5	5-6	6-7	7-8	8-9	9-10	10+	Total
Plant	Plant	0	0	0	0	0	0	0	0	0	0	0	0
A	N	0	22	60	25	26	28	47	74	117	83	84	566
B	NNE	0	53	49	42	47	63	77	125	150	138	357	1,101
C	NE	0	20	31	16	29	86	154	153	187	200	47	923
D	ENE	0	11	62	29	37	29	73	71	86	80	30	508
E	E	0	2	42	49	42	61	40	58	64	51	41	450
F	ESE	0	0	1	56	196	203	725	296	600	438	14	2,529
G	SE	0	3	0	3	330	1,379	6,948	3,433	233	93	101	12,523
H	SSE	8	12	5	962	756	165	91	193	61	69	24	2,346
J	S	0	5	14	254	454	46	36	53	57	57	12	988
K	SSW	0	7	33	73	14	16	55	70	103	146	62	579
L	SW	0	7	32	18	15	36	105	62	89	70	98	532
M	WSW	0	0	37	36	31	35	49	27	53	62	26	356
N	W	9	11	4	28	59	140	75	111	142	144	0	723
P	WNW	135	57	11	13	34	280	363	514	505	110	133	2,155
Q	NW	23	7	7	3	10	23	2	138	175	188	192	768
R	NNW	22	0	0	4	13	21	92	119	115	144	180	710
	Total	197	217	388	1,611	2,093	2,611	8,932	5,497	2,737	2,073	1,401	27,757

TABLE 2.3-1

SITE BOUNDARY DISPERSION FACTOR (χ/Q)
(FROM CENTER OF ISFSI SITE)

Downwind Sector	Downwind Distance (m)	χ/Q (sec/m^3)
N	1,015	1.70×10^{-4}
NNE	905	1.99×10^{-4}
NE	650	3.18×10^{-4}
ENE	550	3.89×10^{-4}
E	550	3.89×10^{-4}
ESE	775	2.21×10^{-4}
SE	795	2.21×10^{-4}
SSE	445	5.53×10^{-4}
S	345	9.02×10^{-4}
SSW	255	1.45×10^{-3}
SW	255	1.45×10^{-3}
WSW	195	2.49×10^{-3}
W	180	2.95×10^{-3}
WNW	195	2.49×10^{-3}
NW	245	1.45×10^{-3}
NNW	1,055	1.55×10^{-4}

Table 2.3-2

DOWNWIND DISPERSION FACTOR (χ/Q)

Downwind Distance (miles)	χ/Q (sec/m³)
0.45	2.66×10^{-4}
0.50	2.23×10^{-4}
1.50	7.45×10^{-5}
2.50	4.54×10^{-5}
3.50	3.24×10^{-5}
4.50	2.50×10^{-5}
7.50	1.46×10^{-5}
15.00	6.84×10^{-6}
35.00	4.35×10^{-6}
35.00	3.94×10^{-6}
45.00	3.67×10^{-6}

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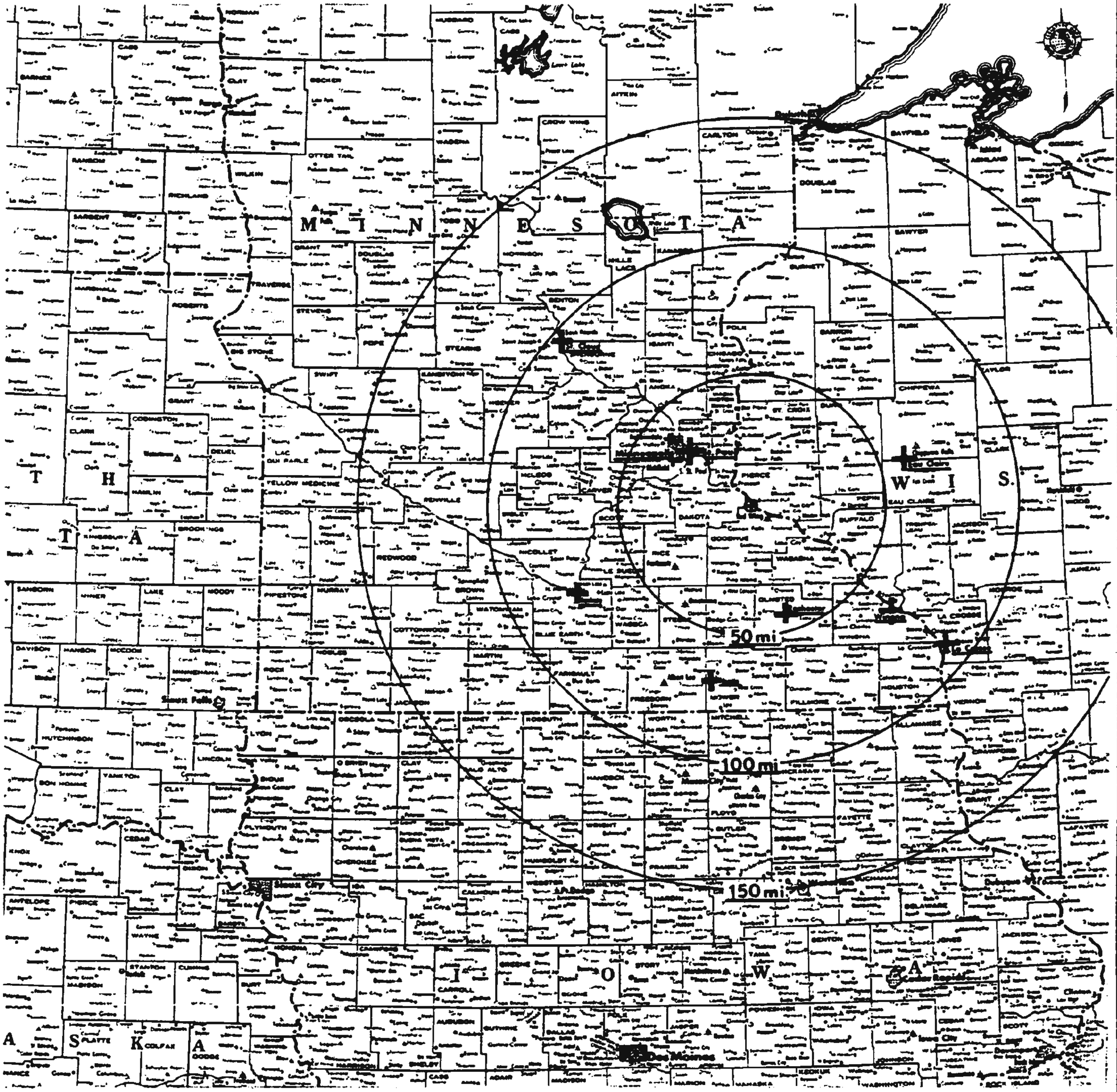

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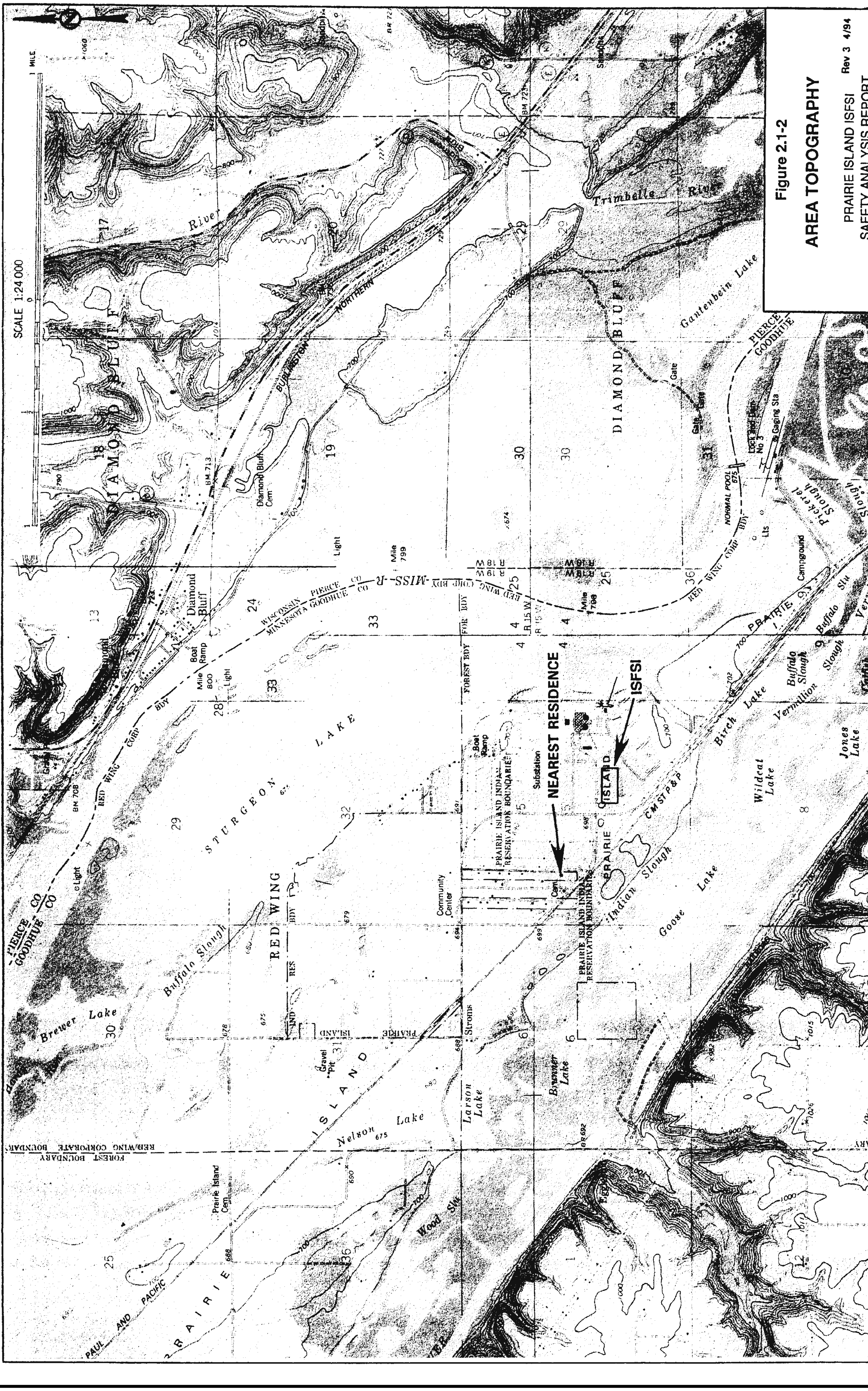
Figure 2.1-1

REGIONAL MAP

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

ISFSI SITE LOCATION

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG2.1-1_REV_0
	PAGE. NO.	DATE: 03-29-06	



AREA TOPOGRAPHY

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG2.1-2_REV_3
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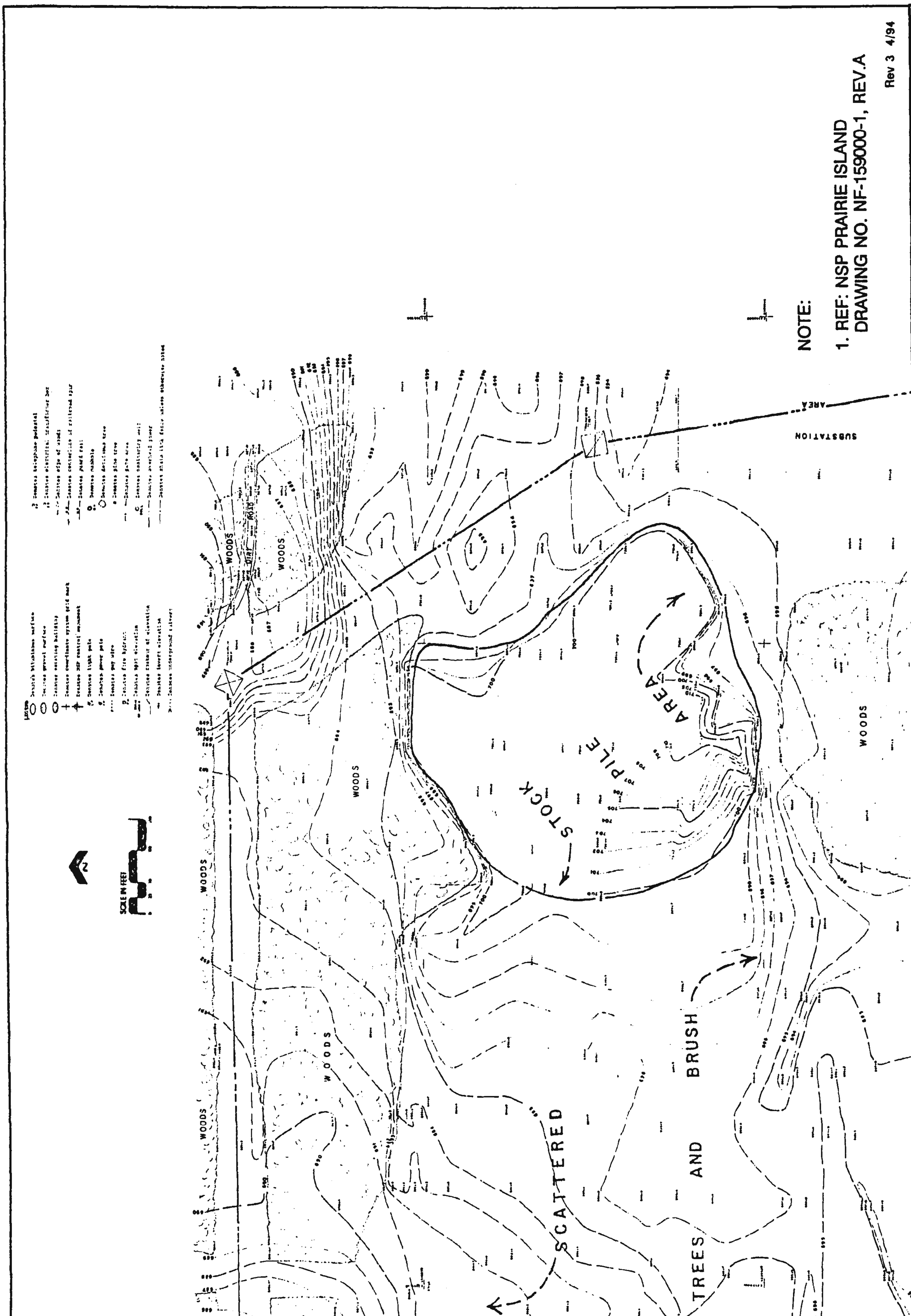

NOTE:

1. REF: NSP PRAIRIE ISLAND DRAWING NO. NF-159000-1, REV.A

Rev 3 4/94

Figure 2.1-3
SHEET 1
SITE TOPOGRAPHY
PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

SHEET 1 SITE TOPOGRAPHY

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG2.1-3_S1_REV_3
	PAGE. NO.	DATE: 03-29-06	

Figure Withheld Under 10 CFR 2.390

Figure 2.1-3

SHEET 2
SITE TOPOGRAPHY

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

SHEET 2 SITE TOPOGRAPHY			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG2.1-3_S2_REV_3
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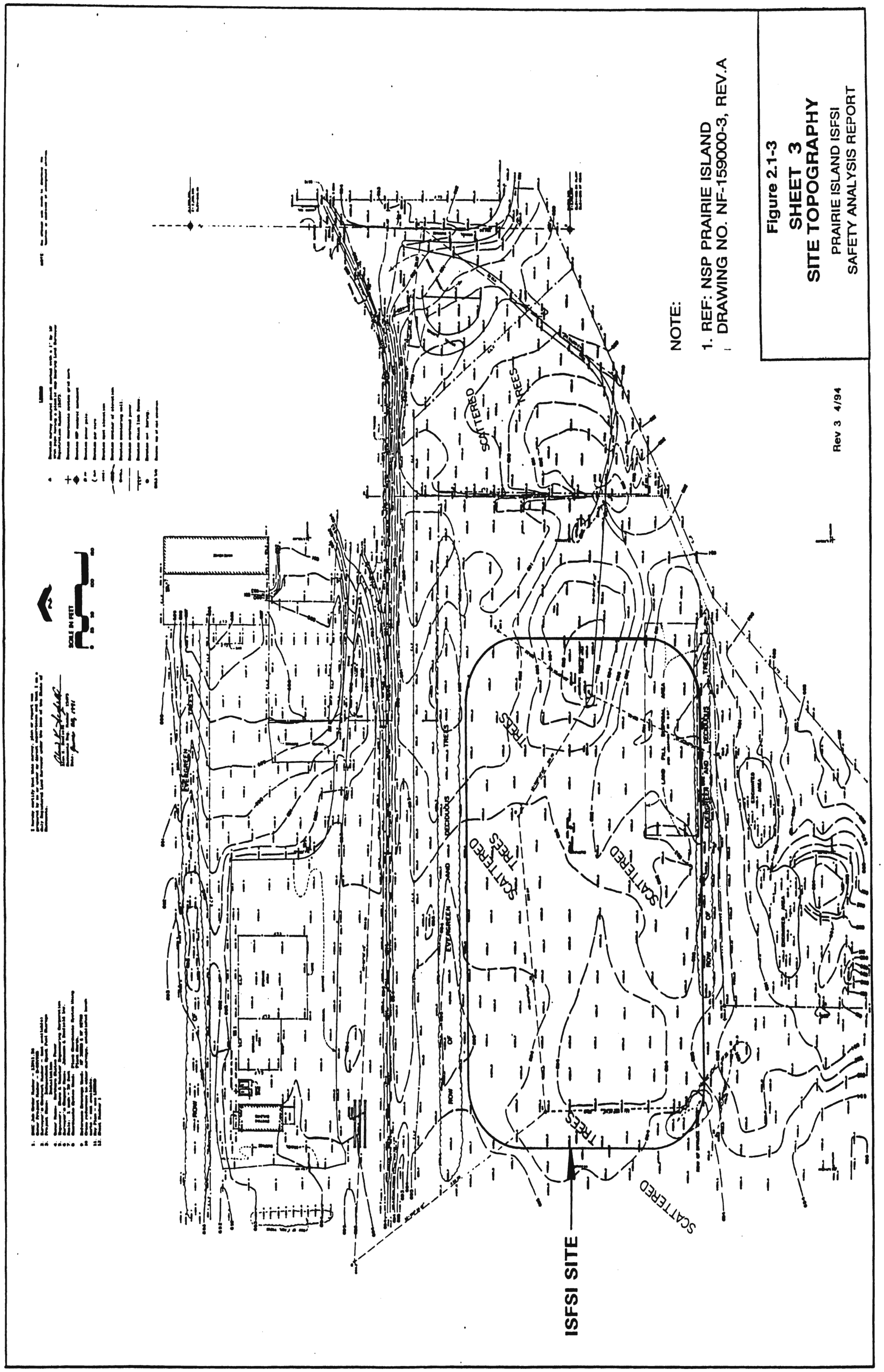

SHEET 3 SITE TOPOGRAPHY

NORTHERN STATES POWER COMPANY
Xcel Energy
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA

DRAWN BY: VLS	REVISION: 3	FIG2.1-3_S3_REV_3
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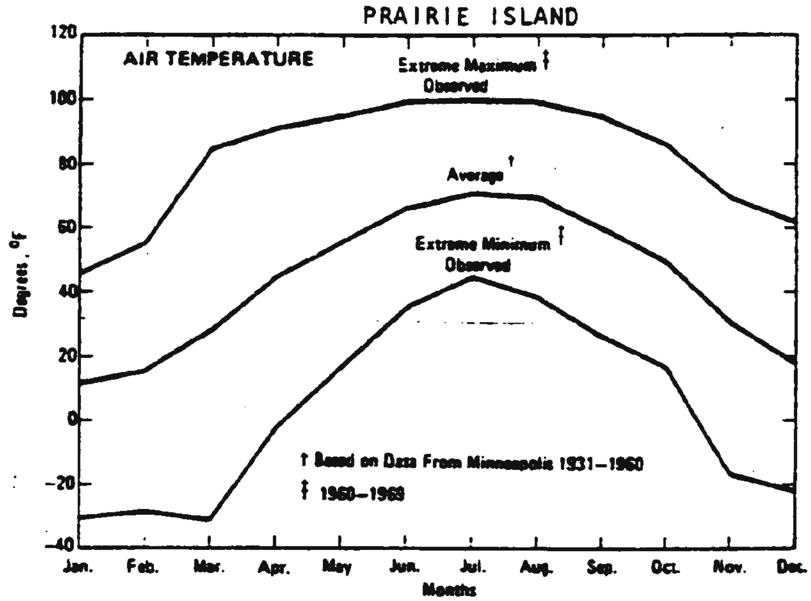

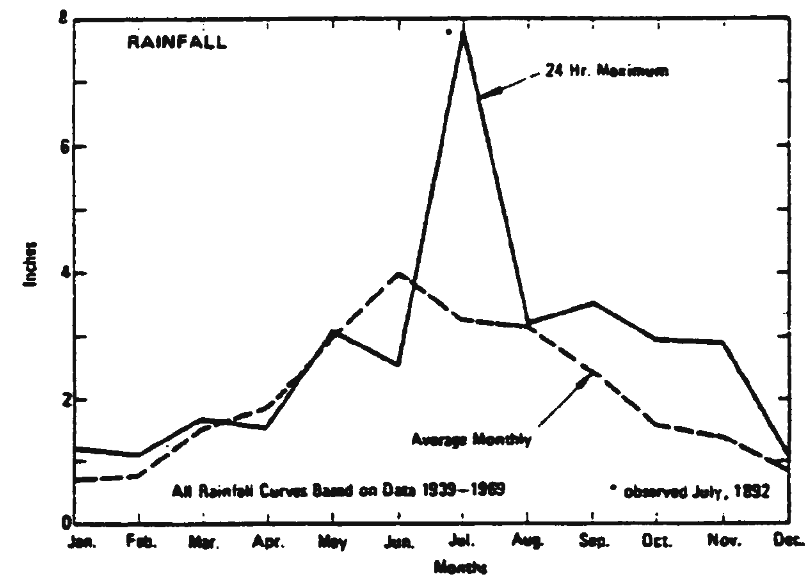

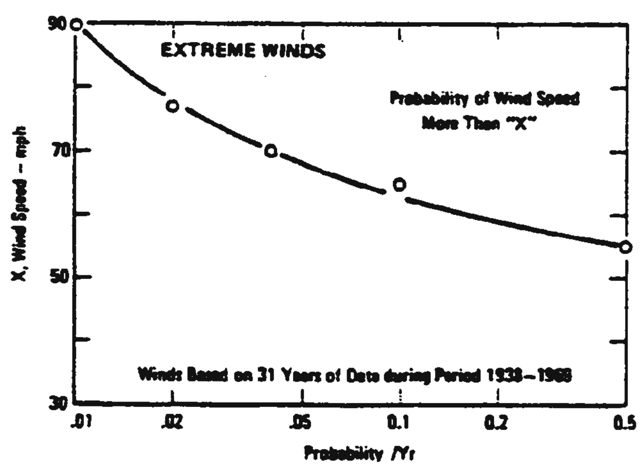

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Figure 2.3-1

CLIMATIC DATA

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

CLIMATIC DATA			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG2.3-1_REV_0
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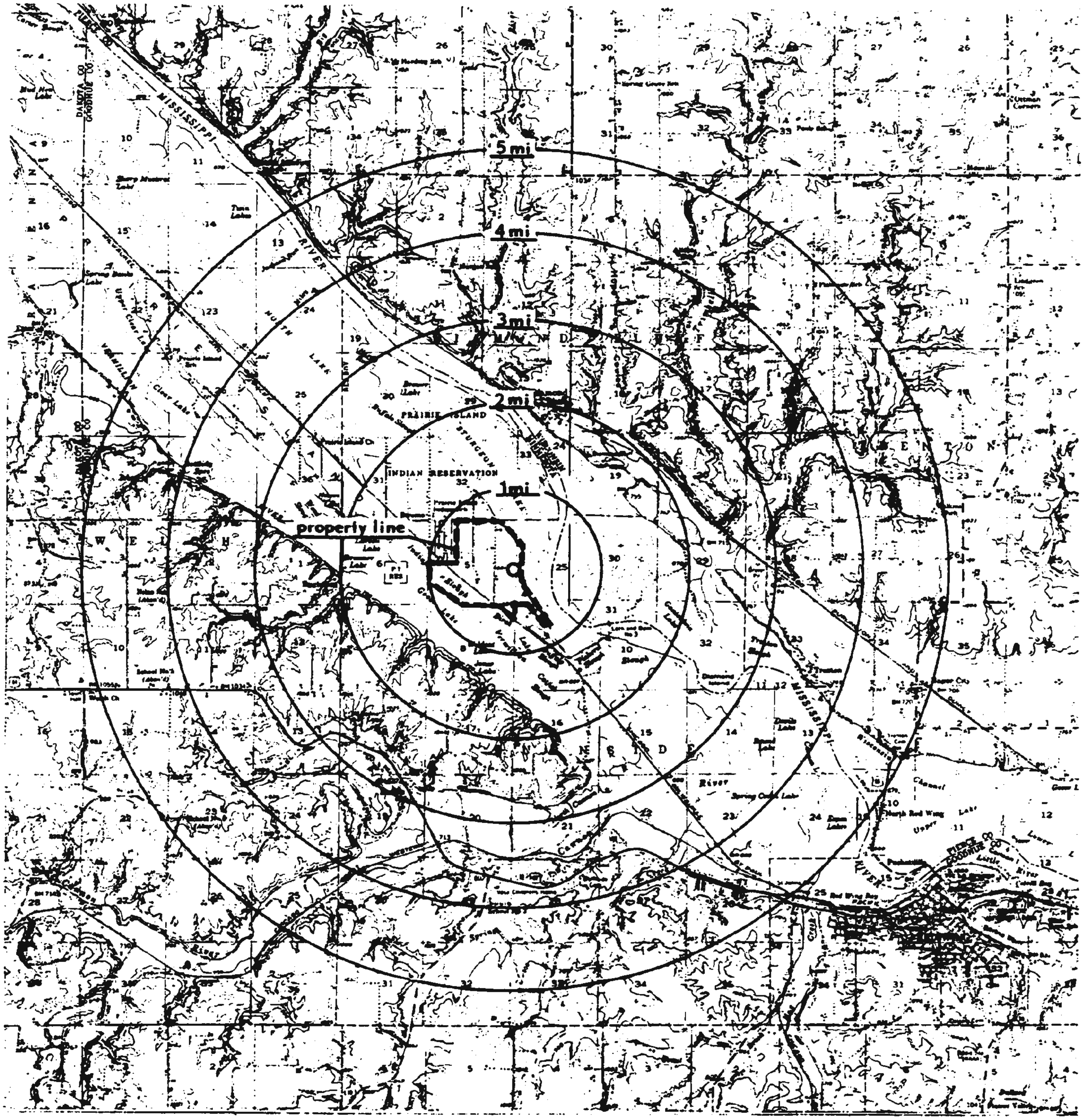

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Figure 2.3-2

REGIONAL TOPOGRAPHY

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

REGIONAL TOPOGRAPHY

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY:	VLS	REVISION:	0	FIG2.3-2_REV_0
	PAGE. NO.		DATE:	03-29-06	

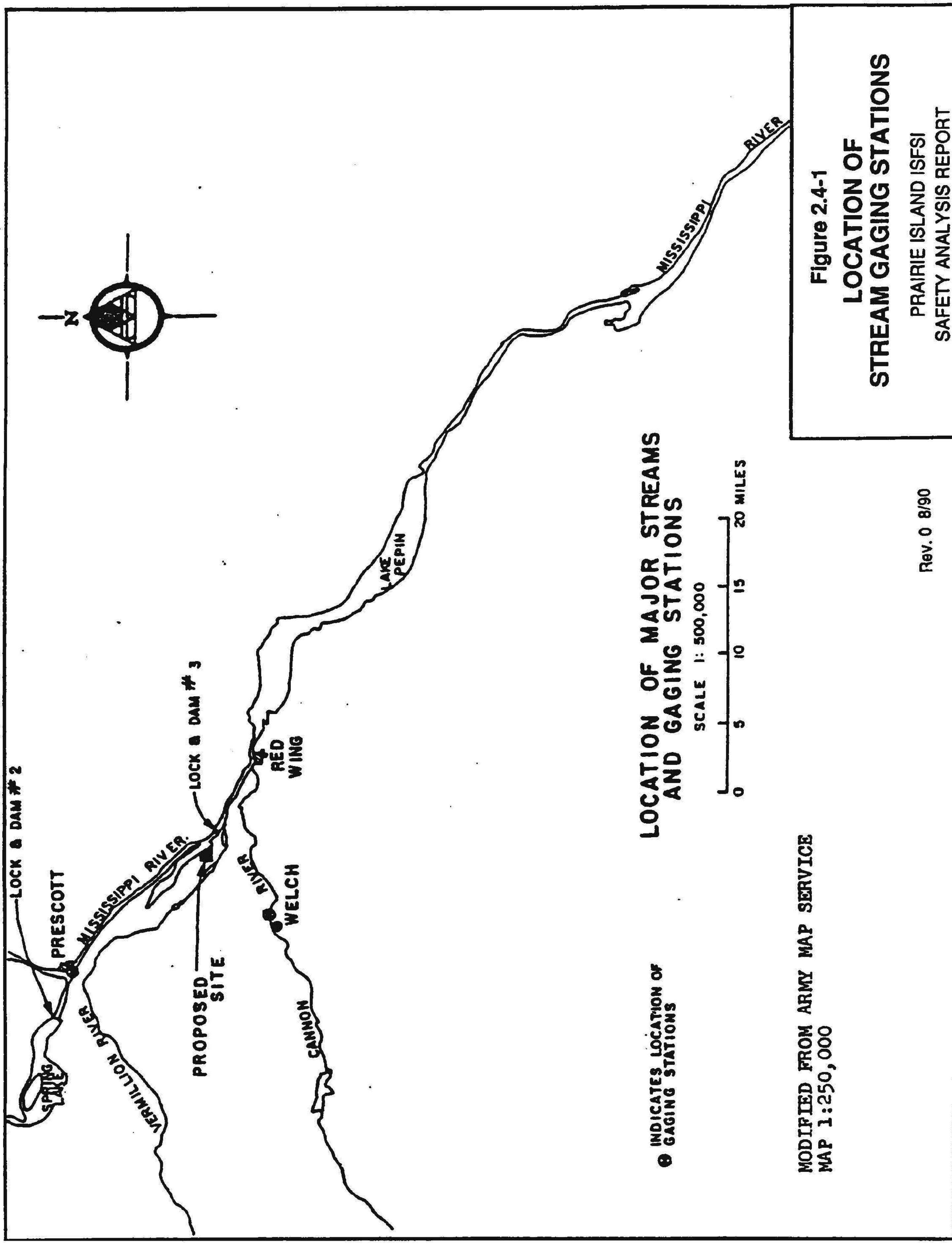

Figure 2.4-1
LOCATION OF STREAM GAGING STATIONS
PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

LOCATION OF STREAM GAGING STATIONS

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG2.4-1_REV_0
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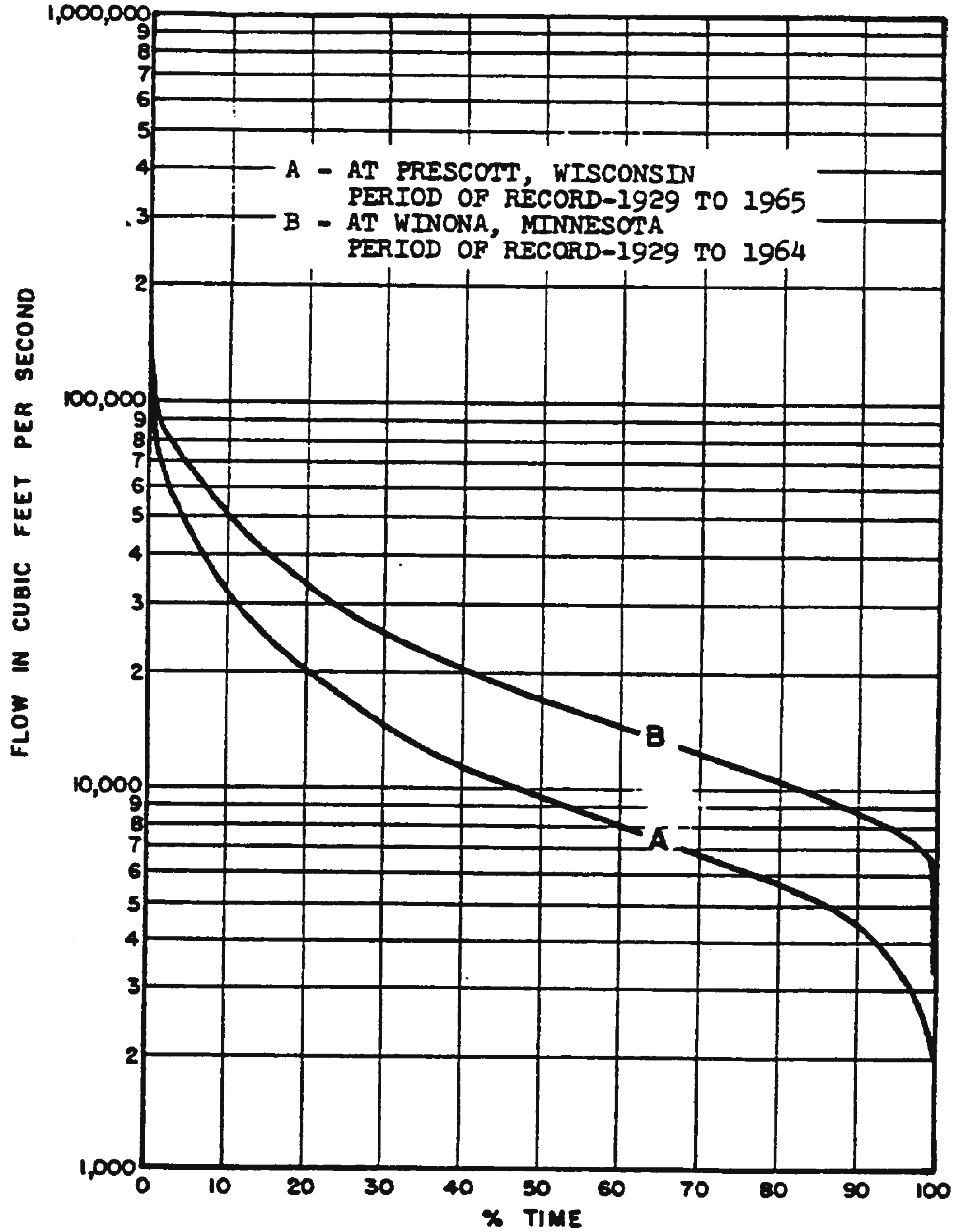

Rev. 0 8/90

GRAPH COMPILED FROM DATA OBTAINED
FROM U.S.GEOLOGICAL SURVEY
STATISTICAL SUMMARIES

Figure 2.4-2

FLOW DURATION CURVES FOR MISSISSIPPI RIVER

PRAIRIE ISLAND ISFSI SAFETY ANALYSIS REPORT

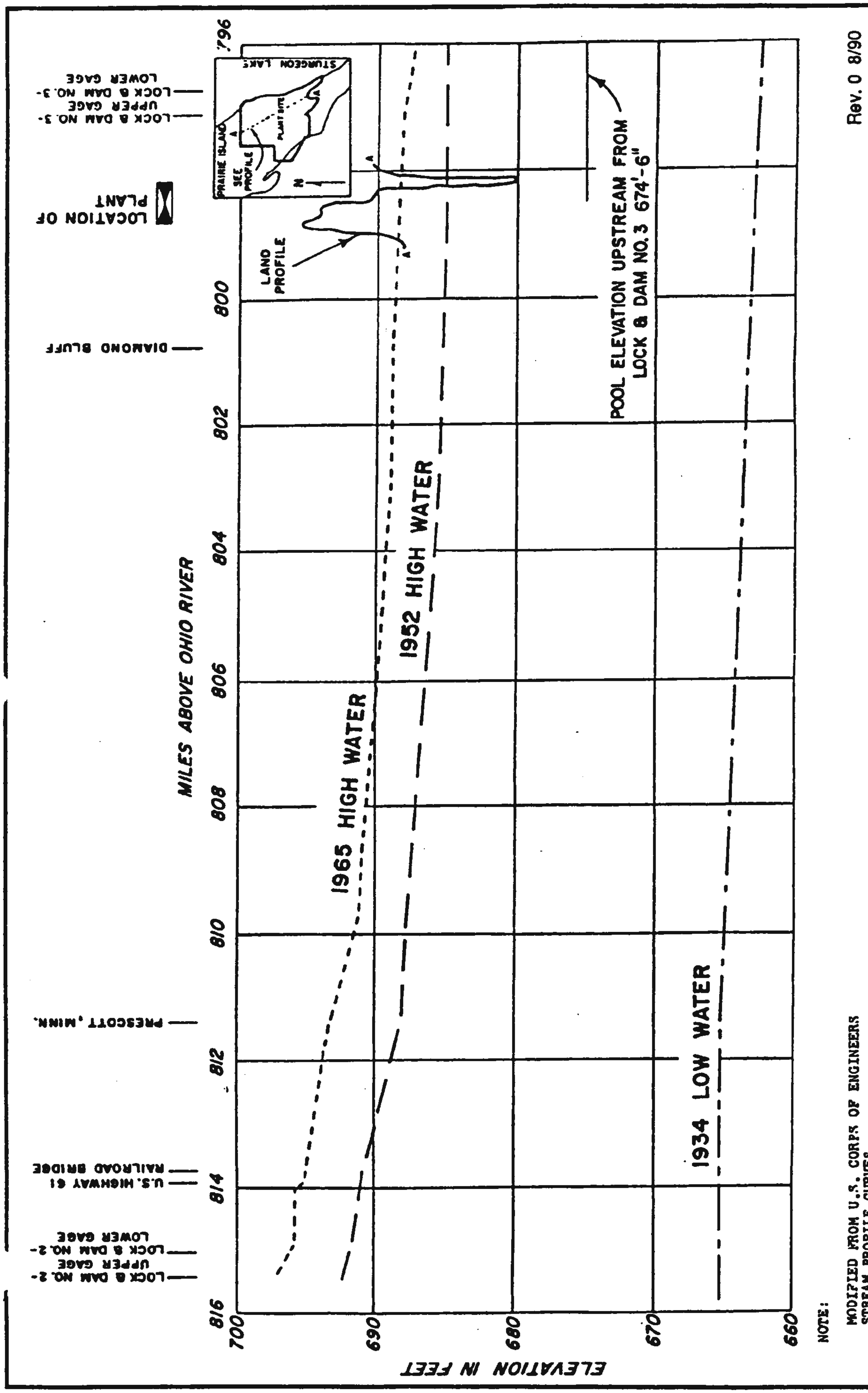

Figure 2.4-3

STREAM PROFILE

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

STREAM PROFILE			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG2.4-3_REV_0
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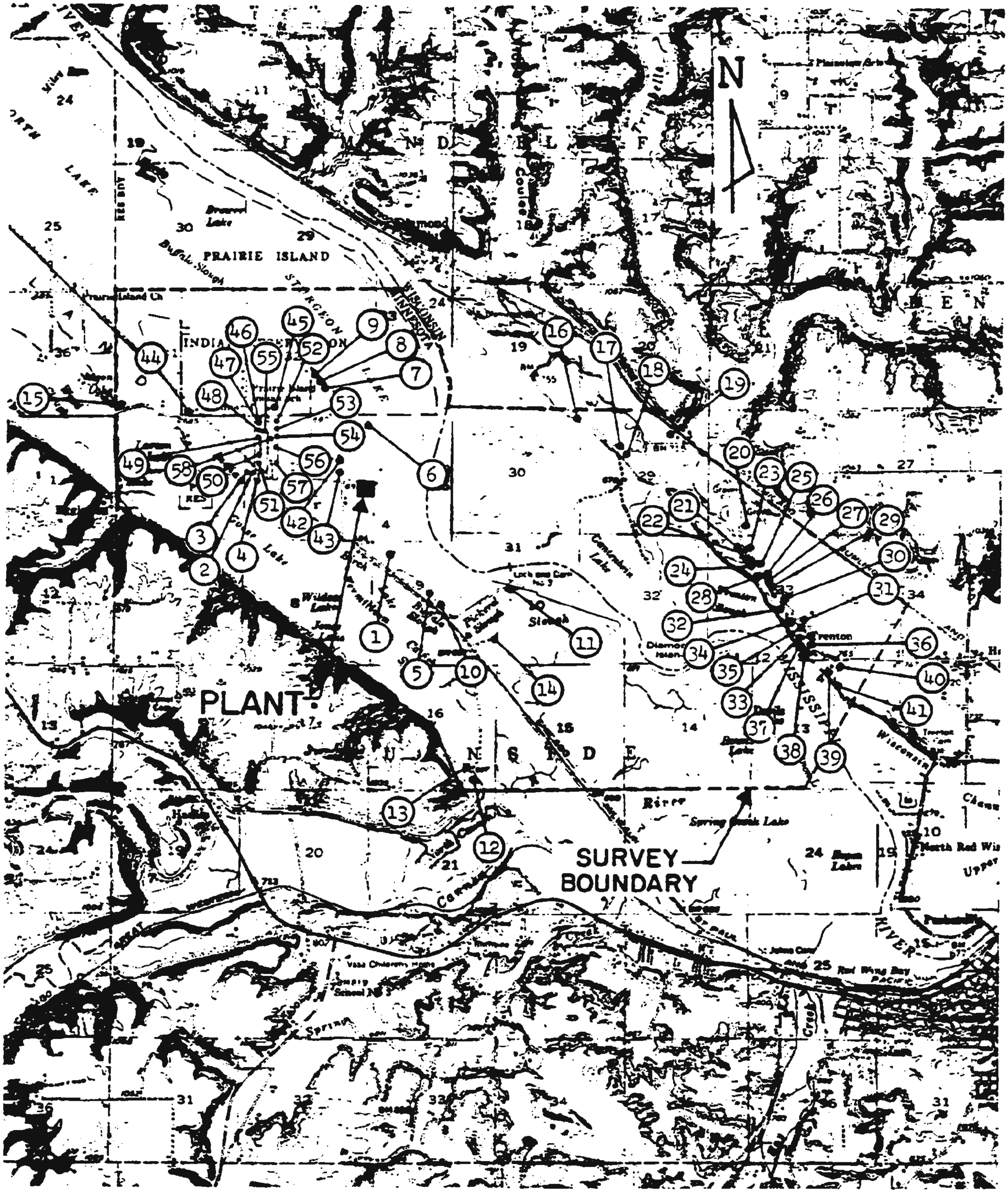


Figure 2.4-4

LOCATION OF WELLS

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

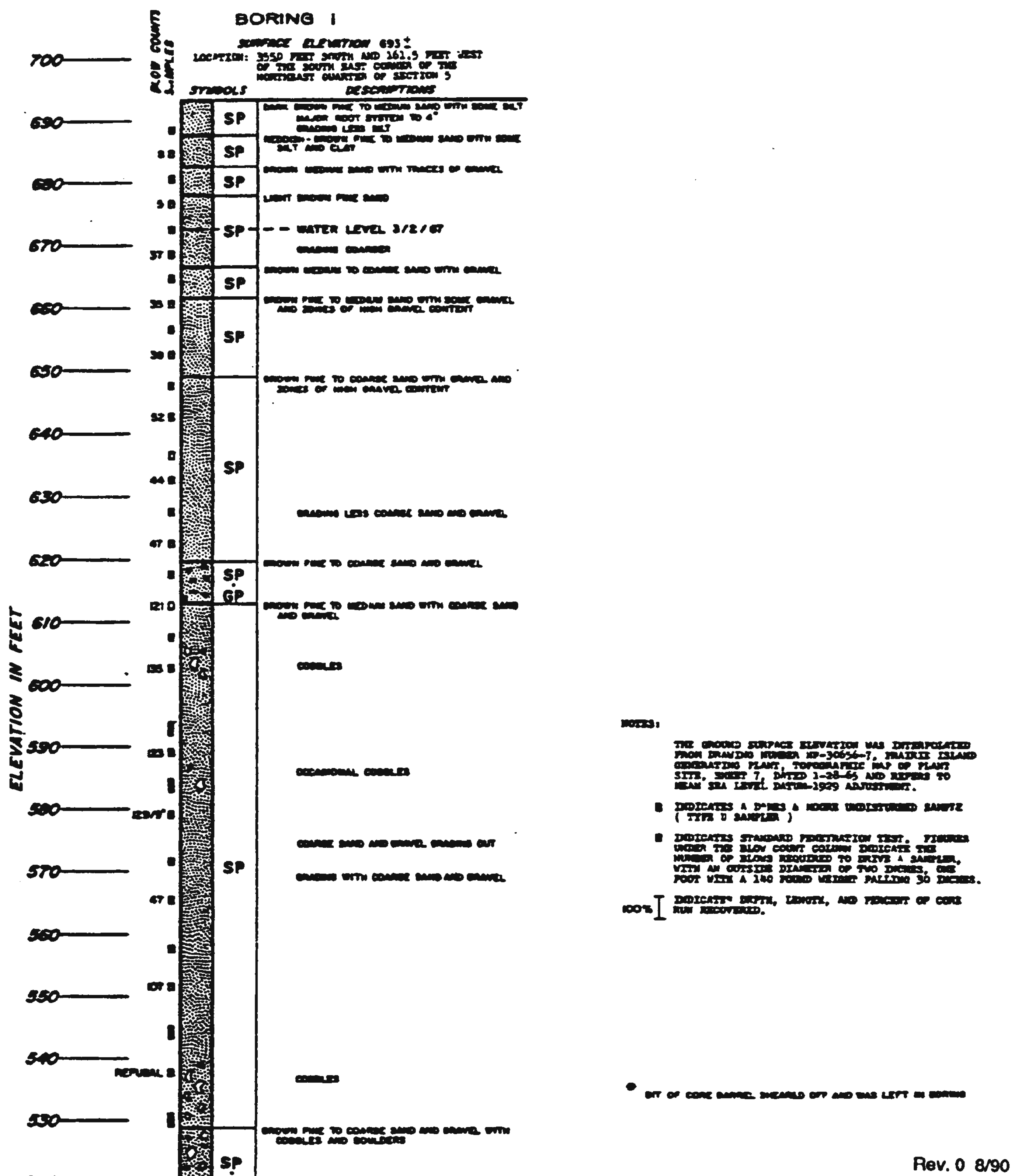
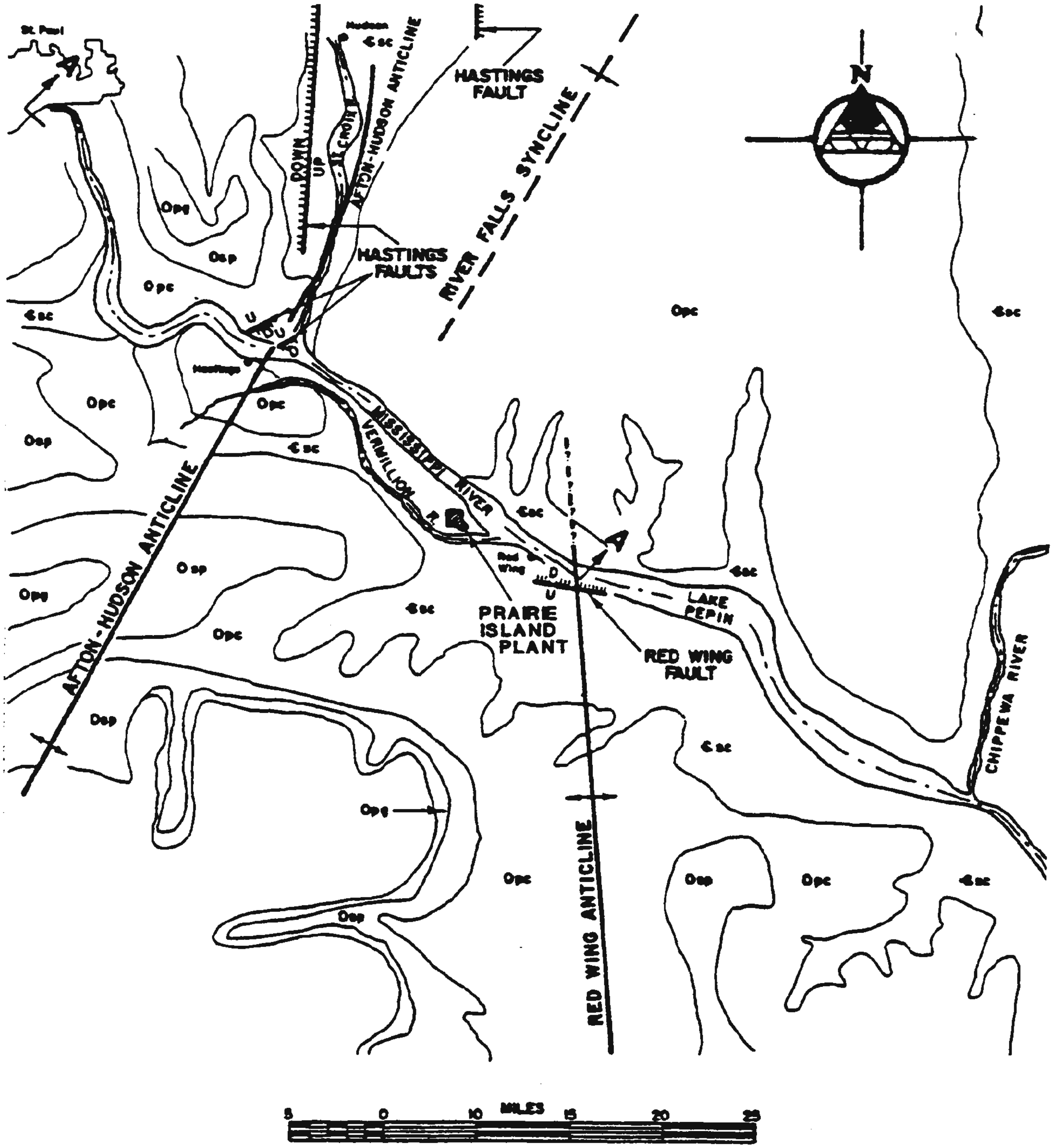


Figure 2.5-1

BORING LOG

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT



LEGEND

- Opg LIMESTONE AND DOLOMITE - PLATTVILLE Fm. SHALE - GLENWOOD Fm.
- Osp SANDSTONE - ST. PETER Fm.
- Opc DOLOMITE - SHAKOPEE Fm. SANDSTONE - NEW RICHMOND Fm. DOLOMITE - ONEOTA Fm.
- €sc SANDSTONE - JORDAN Fm. DOLOMITE AND SILTSTONE - ST. LAWRENCE Fm. SANDSTONE - FRANCONIA Fm., DRESBACH Fm.
- FAULT - KNOWN AND INFERRED
- ANTICLINE AXIS
- CROSS SECTION A-A'
- SYNCLINE AXIS

REFERENCE: GEOLOGIC MAP OF MINNESOTA ST. PAUL SHEET 1966

Rev. 0 8/90

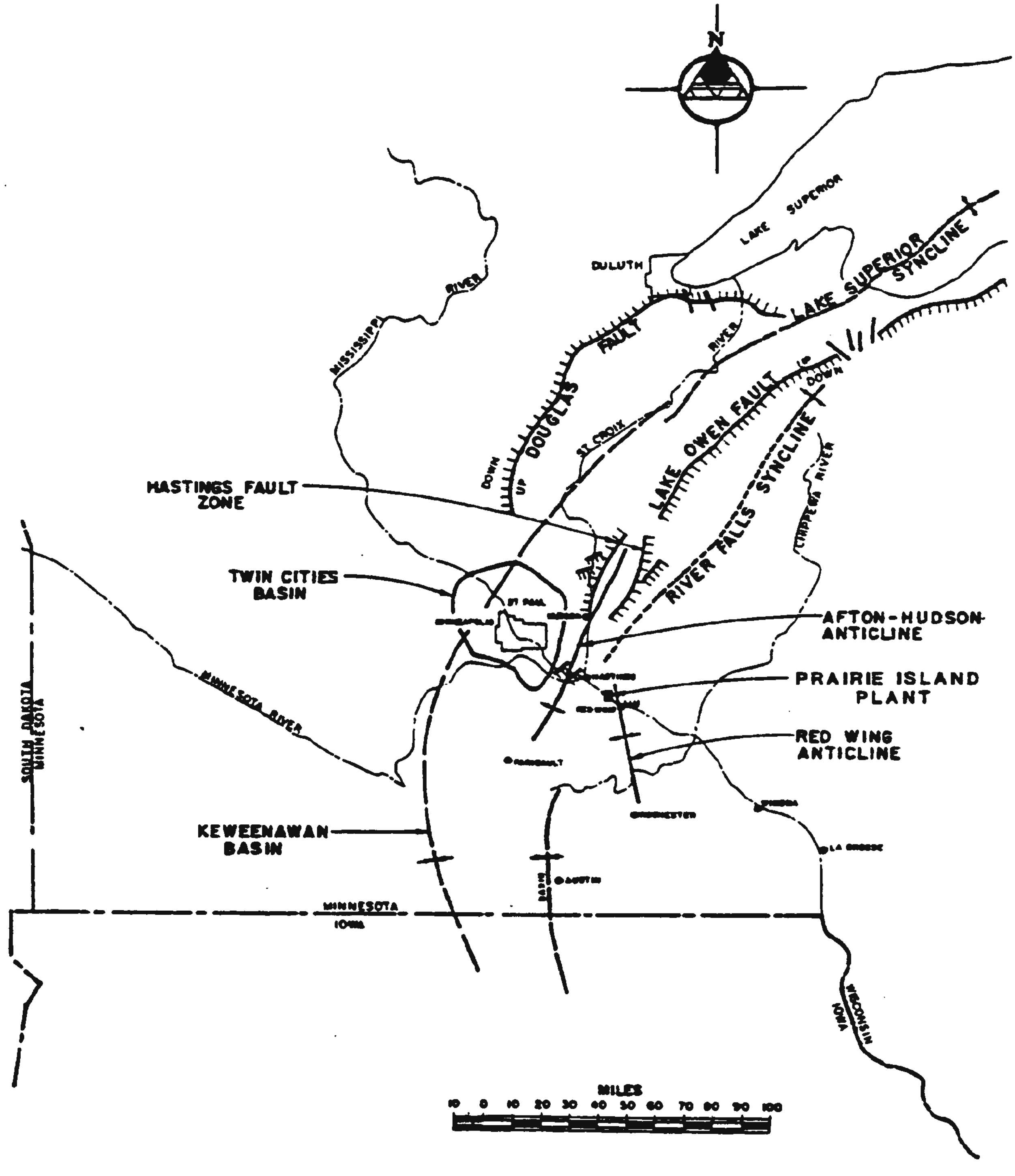
Figure 2.5-2

REGIONAL GEOLOGIC MAP

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

REGIONAL GEOLOGIC MAP

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG2.5-2_REV_0
	PAGE. NO.	DATE: 03-29-06	



REFERENCES:

1-TECTONIC MAP OF THE UNITED STATES BY U.S.G.S. AND THE AMERICAN ASSOCIATION OF PETROLEUM GEOLOGISTS 1961

2-UNPUBLISHED STRUCTURE MAP BY MINNESOTA GEOLOGICAL SURVEY

Rev. 0 8/90

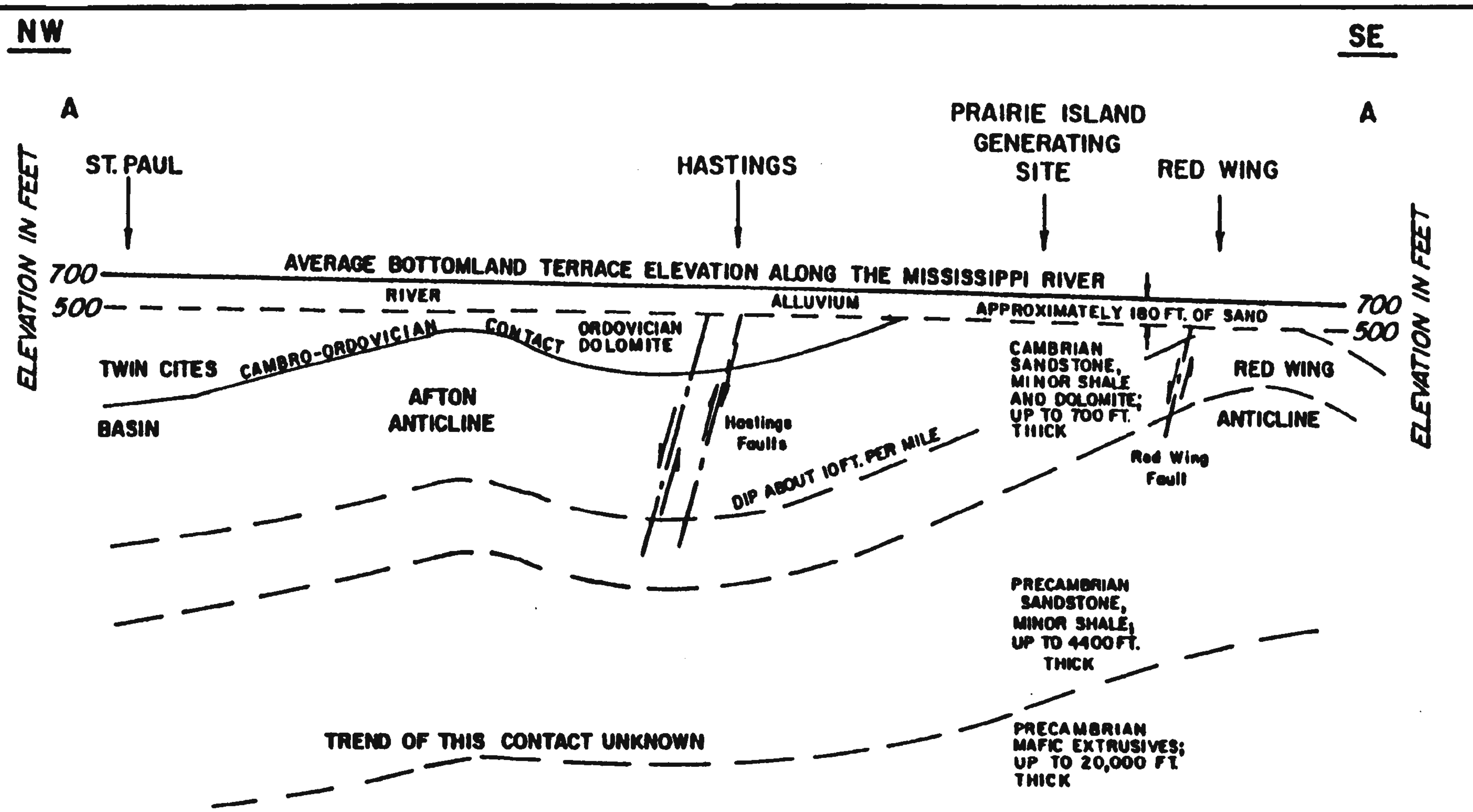
Figure 2.5-3

REGIONAL GEOLOGIC STRUCTURE

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

REGIONAL GEOLOGIC STRUCTURE

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG2.5-3_REV_0
	PAGE. NO.	DATE: 03-30-06	



REGIONAL GEOLOGIC CROSS SECTION A-A
PRAIRIE ISLAND GENERATING SITE
RED WING, MINNESOTA

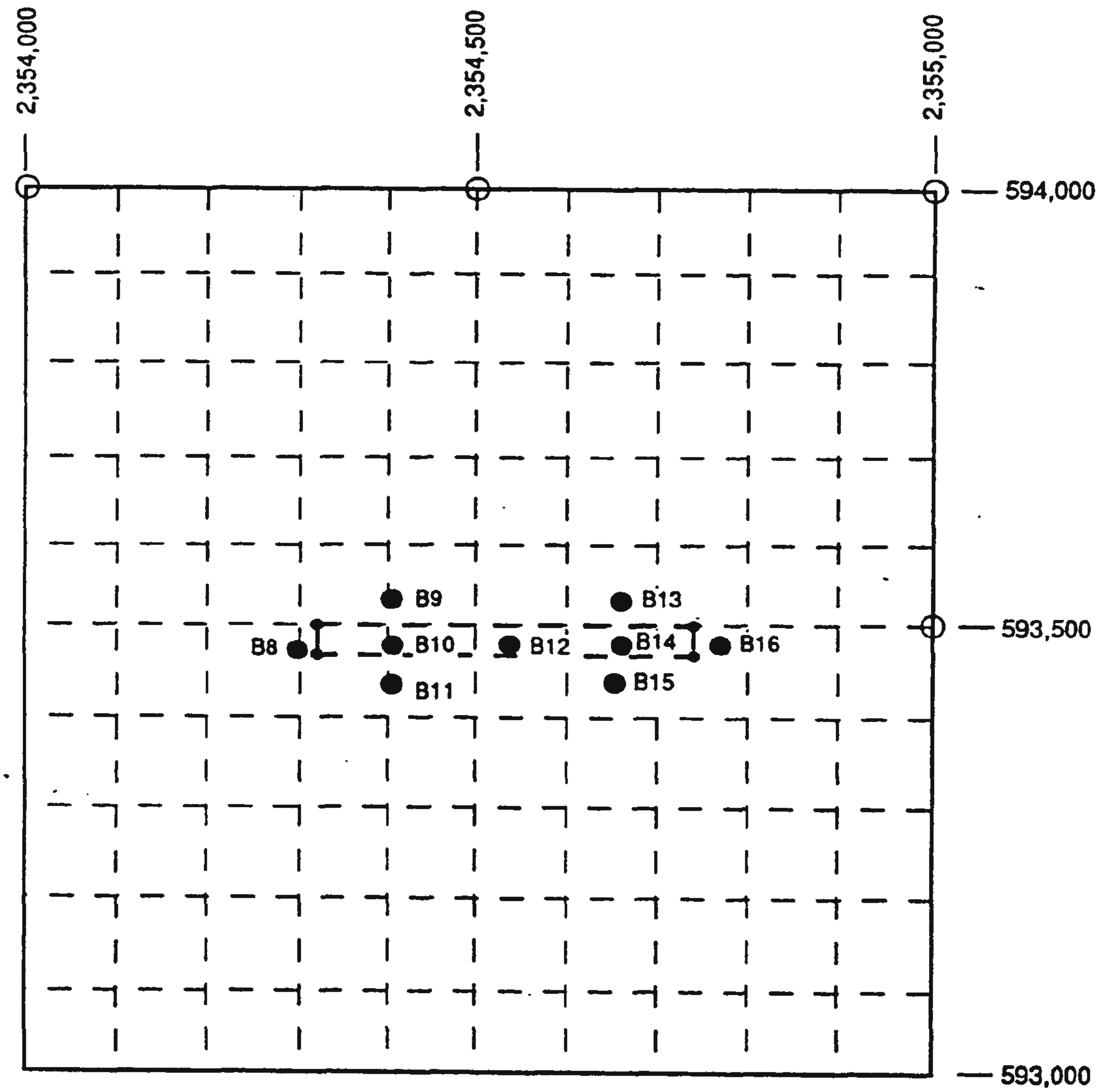
HORIZONTAL SCALE: 1 IN. = APPROX. 6 MILES
VERTICAL SCALE: VARIABLE AND HIGHLY EXAGGERATED

Rev. 0 8/90

Figure 2.5-4
REGIONAL GEOLOGIC CROSS SECTION
PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

REGIONAL GEOLOGIC CROSS SECTION

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG2.5-4_REV_0
	PAGE. NO.	DATE: 03-30-06	



SCALE: 1" = 200'

N

Rev. 2 9/91

○ NSP SURVEY CONTROL POINTS

● BORING LOCATION

• NSP SURVEY POINTS CORNERS OF PORPOSED CONCRETE SLAB

Figure 2.5-5

SITE BORING LOCATIONS

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

SITE BORING LOCATIONS			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 2	FIG2.5-5_REV_2
	PAGE. NO.	DATE: 03-30-06	

Figure 2.5-6
SITE GRADING PLAN
PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

Figure Withheld Under 10 CFR 2.390

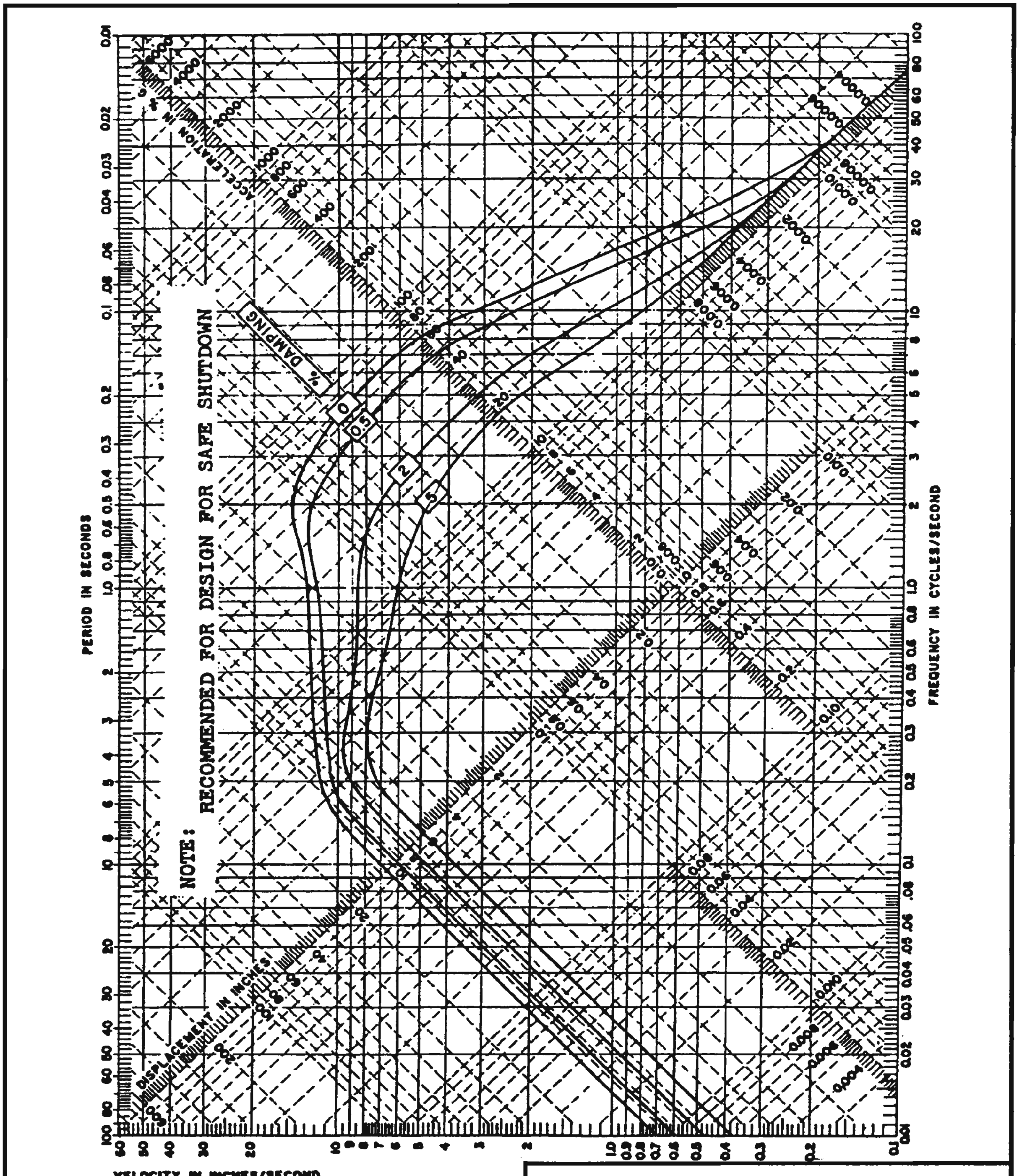
SITE GRADING PIAN			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG2.5-6_REV_3
	PAGE. NO.	DATE: 03-30-06	

Figure 2.5-7
SITE GRADING - SECTIONS AND DETAILS

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

Figure Withheld Under 10 CFR 2.390

SITE GRADING - SECTIONS AND DETAILS			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG2.5-7_REV_3
	PAGE. NO.	DATE: 03-30-06	



Rev 3 4/94

Figure 2.5-8

ISFSI DESIGN EARTHQUAKE RESPONSE SPECTRA

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

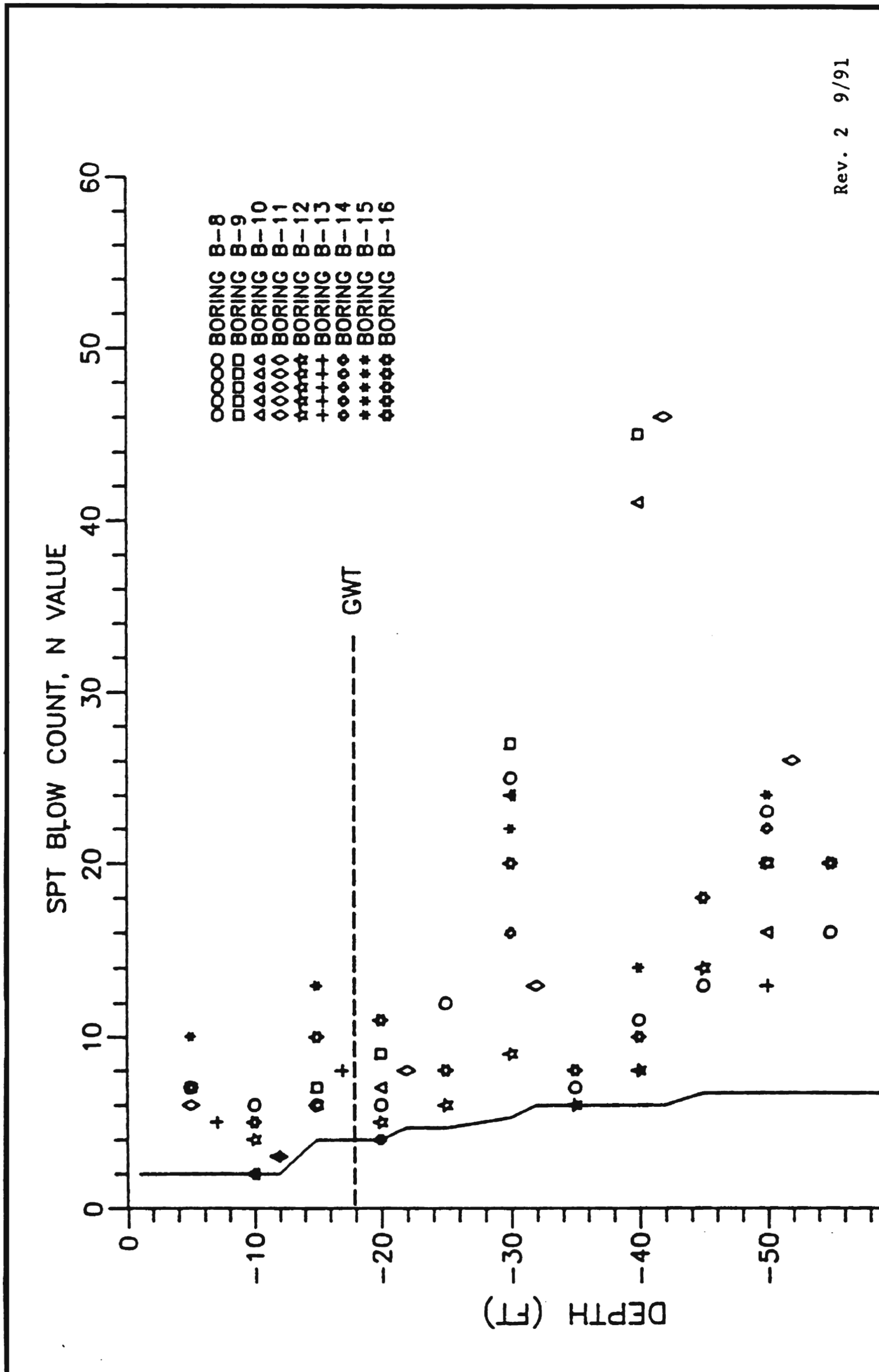


FIGURE 2.5-9

LIQUEFACTION ANALYSIS

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

LIQUEFACTION ANALYSIS

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 2	FIG2.5-9_REV_2
	PAGE. NO.	DATE: 03-30-06	

APPENDIX 2A

BORING LOGS

document #

FIELD TEST BORING RECORD COVER SHEET
STONE & WEBSTER ENGINEERING CORP.

		BORING NO. B-8
SITE : ISFSI Project Prairie Island (site 2)	J.O. NO. 12911.5401	SHEET 1 OF 3

COORDINATES N 593,482 E 2,354,298 GROUND ELEV. 695.0

INCLINATION Vertical BEARING NA INSPECTOR P. Jacobs

DATE: START/FINISH 6-4-91 / 6-4-91 CONTRACTOR/DRILLER Twin City Testing

STATIC G.W. DEPTH/DATE 19.2 (FT) / 6-4-91 DRILL RIG TYPE CME 55

DEPTH TO BEDROCK NA (FT) TOTAL DEPTH DRILLED 61.0 (FT)

METHODS:

DRILLING SOIL 6" Hollow Stem Augur Above The Water Table, Then Mud Rotary With 2 15/16" Tricone Bit Below The Water Table

SAMPLING SOIL Standard Split Spoon

DRILLING ROCK

SPECIAL TESTING OR INSTRUMENTATION

COMMENTS:

SUMMARY

SOIL DRILLED 61.0 (FT) ROCK DRILLED NA (FT)

NUMBER SPLIT BARREL SAMPLES 13

NO. & TYPE UNDISTURBED SAMPLES 0

LIST ALL TEST, INSPECTION AND CALIBRATION RECORDS ATTACHED:

GROUNDWATER (DEPTH BELOW GROUND SURFACE)

DEPTH	DATE	DEPTH	DATE
19.2	6-4-91		

NOTES

1. DATUM IS M.S.L. UNLESS OTHERWISE INDICATED

LEGEND FOR FIELD BORING LOGS

DISTURBED SAMPLES
- W - WASH
- S - SPLIT BARREL (ASTM - D1586 UNLESS OTHERWISE NOTED)
- E - ENVIRONMENTAL
- Z - OTHER, DESCRIBE:

UNDISTURBED SAMPLES
- US - SHELBY TUBE
- UF - STATIONARY (FIXED PISTON)
- UO - OSTERBERG
- UP - PITCHER
- UD - DENISON
- X - OTHER , DESCRIBE:

N - STD PENETRATION RESISTANCE BLOWS/FT (140 LB. HAMMER, UNLESS OTHERWISE NOTED)

REC - RECOVERY

RQD - ROCK QUALITY DESIGNATION

( ) - INCHES OF SAMPLE RECOVERY

PREPARED BY Paul J. Jacobs	DATE 6-19-91
CHECKED BY M. Gene Yow	DATE 6/19/91

document #

FIELD BORING LOG
STONE & WEBSTER ENGINEERING CORP.

SITE ISFSI Project Prairie Island (site 2)	J.O. NO. 12911.5401	BORING NO. B-8 / SHEET 2 OF 3
DRILLER Tom White	INSPECTOR P. Jacobs	

SAMPLE DESCRIPTION — PREPARED BY Paul J. Jacobs — CHECKED BY M. Gene Yow 6/19/91

DEPTH (FEET)	SAMPLE TYPE	SAMPLE NUMBER	BLOWS/OR REC/RQD	SPT N VALUE	GROUP SYMBOL	SAMPLE DESCRIPTION
	SS	1	1-2-4 (13")	6	SM	Silty Sand - Poorly Graded, Fine with silt, Loose, Moist, Reddish Brown
5	SS	2	1-2-3 (14")	5	SP	Sand - Poorly Graded, Fine with minor silt, Loose, Moist, Reddish Brown
10	SS	3	2-2-3 (15")	5	SP	Sand - Poorly Graded, Fine, Med Dense to Loose, Moist, Reddish Brown, sample layered with sands/silty sands
15	SS	4	2-3-3 (14")	6	SP	Sand - Poorly Graded, Fine to Medium, Med Dense to Loose, Moist, Light Brown
20	SS	5	3-3-3 (16")	6	SP	same as above - with a 2" thick gravely sand seam
25	SS	6	4-8-5 (13")	13	SP	same as above - no gravel
30	SS	7	8-13-15 (16")	28	SP	same as above
35	SS	8	2-4-4 (12")	8	SP	same as above - with one 1" piece of gravel SPT blow count value may not be representative because the rig ran out of gas 1/2 way through the 18" blow count test. Test was resumed after refueling
40	SS	9	4-6-7 (13")	13	SP	same as above - no gravel
	SS	10	16-7-9 (9")	16	SP	same as above - with sand becoming coarser at the bottom 1" of the sample barrel

document #

FIELD BORING LOG
STONE & WEBSTER ENGINEERING CORP.

BORING NO. B-8	
SITE ISFSI Project Prairie Island (site 2)	J.O. NO. 12911.5401 — SHEET 3 OF 3
DRILLER Tom White	INSPECTOR P. Jacobs

SAMPLE DESCRIPTION

PREPARED BY Paul A. Jacobs CHECKED BY M. Gene Yow 6/19/91

DEPTH (FEET)	SAMPLE TYPE	SAMPLE NUMBER	BLOWS/OR REC/RQD	SPT N VALUE	GROUP SYMBOL	SAMPLE DESCRIPTION
						encountered gravel while drilling at 46.5' (rough drilling)
50	SS	11	12-16-13 (12")	29	SW-GW	Gravely Sand - Well Graded, Fine to 1" gravel, Dense, wet, Light Brown with Black Gravel
55	SS	12	8-11-10 (11")	21	SW-GW	same as above - with a 1/2" thick lense of silty sand
60	SS	13	9-15-13 (14")	28	SW-GW	same as above - no silt T.D. 61.0'
65						
70						
75						
80						
85						

document #

FIELD TEST BORING RECORD COVER SHEET
STONE & WEBSTER ENGINEERING CORP.

SITE : ISFSI Project Prairie Island (site 2)	J.O. NO. 12911.5401	BORING NO. B-9 SHEET 1 OF 3

COORDINATES N 593,538 E 2,354,406 GROUND ELEV. 694.6

INCLINATION Vertical BEARING NA INSPECTOR P. Jacobs

DATE: START/FINISH 6-4-91 / 6-5-91 CONTRACTOR/DRILLER Twin City Testing

STATIC G.W. DEPTH/DATE 18.2 (FT) / 6-5-91 DRILL RIG TYPE CME 55

DEPTH TO BEDROCK NA (FT) TOTAL DEPTH DRILLED 51.0 (FT)

METHODS:

DRILLING SOIL 6" Hollow Stem Auger Above The Water Table, Then Mud Rotary With 2 15/16" Tricone Bit Below The Water Table

SAMPLING SOIL Standard Split Spoon

DRILLING ROCK

SPECIAL TESTING OR INSTRUMENTATION

COMMENTS:

SUMMARY

SOIL DRILLED 51.0 (FT) ROCK DRILLED NA (FT)

NUMBER SPLIT BARREL SAMPLES 6

NO. & TYPE UNDISTURBED SAMPLES 0

LIST ALL TEST, INSPECTION AND CALIBRATION RECORDS ATTACHED:

GROUNDWATER (DEPTH BELOW GROUND SURFACE)

DEPTH	DATE	DEPTH	DATE
18.2	6-5-91		

NOTES

1. DATUM IS M.S.L. UNLESS OTHERWISE INDICATED

LEGEND FOR FIELD BORING LOGS

DISTURBED SAMPLES
- W - WASH
- S - SPLIT BARREL (ASTM - D1586 UNLESS OTHERWISE NOTED)
- E - ENVIRONMENTAL
- Z - OTHER, DESCRIBE:

UNDISTURBED SAMPLES
- US - SHELBY TUBE
- UF - STATIONARY (FIXED PISTON)
- UO - OSTERBERG
- UP - PITCHER
- UD - DENISON
- X - OTHER , DESCRIBE:

N - STD PENETRATION RESISTANCE BLOWS/FT (140 LB. HAMMER, UNLESS OTHERWISE NOTED)

REC - RECOVERY

RQD - ROCK QUALITY DESIGNATION

( ) - INCHES OF SAMPLE RECOVERY

PREPARED BY Paul J. Jacobs DATE 6-19-91

CHECKED BY M. Gane Yow DATE 6/19/91

FIELD BORING LOG
STONE & WEBSTER ENGINEERING CORP.

BORING NO. B-9

SITE ISFSI Project Prairie Island (site 2)

J.O. NO. 12911.5401

SHEET 2 **OF** 3

DRILLER Tom White

INSPECTOR P. Jacobs

SAMPLE DESCRIPTION

PREPARED BY Paul J. Jacobs **CHECKED BY** M Dane Yow 6/19/91

DEPTH (FEET)	SAMPLE TYPE	SAMPLE NUMBER	BLOWS/OR REC/RQD	SPT N VALUE	GROUP SYMBOL	SAMPLE DESCRIPTION
5	SS	1	1-2-3 (18")	5	SP	Sand - Poorly Graded, Fine, Loose, Moist Reddish Brown
10						
15	SS	2	2-4-3 (18")	7	SM	Silty Sand - Poorly Graded, Fine with laminations of silt in fine sand, Loose, Moist, Reddish Brown
20	SS	3	3-4-5 (11")	9	SP	Sand - Poorly Graded, Fine with 1/4-1/2" gravel, Med Dense to Loose, Wet, Light Brown with Black Gravel
25						
30	SS	4	11-14-16 (16")	30	SP	same as above - with no gravel
35						
40	SS	5	15-24-30 (15")	54	SP	same as above
						encountered gravel while drilling at 44.5'

document #

FIELD BORING LOG
STONE & WEBSTER ENGINEERING CORP.

BORING NO. B-9

SITE ISFSI Project Prairie Island (site 2)	J.O. NO. 12911.5401	SHEET 3 OF 3
DRILLER Tom White	INSPECTOR P. Jacobs	

SAMPLE DESCRIPTION

PREPARED BY Paul J. Jacobs CHECKED BY M. Gene Yow 6/19/91

DEPTH (FEET)	SAMPLE TYPE	SAMPLE NUMBER	BLOWS/OR REC/RQD	SPT N VALUE	GROUP SYMBOL	SAMPLE DESCRIPTION
50	SS	6	10-12-14 (11")	26	SW-GW	Gravely Sand - Well Graded, Fine to 3/4" gravel, Dense, Wet, Tan Brown with Black Gravel
						T.D. 51.0'
55						
60						
65						
70						
75						
80						
85						

document #

FIELD TEST BORING RECORD COVER SHEET
STONE & WEBSTER ENGINEERING CORP.

SITE : ISFSI Project Prairie Island (site 2)	J.O. NO. 12911.5401	BORING NO. B-10 SHEET 1 OF 3

COORDINATES N 593,482 E 2,354,406 GROUND ELEV. 694.6

INCLINATION Vertical BEARING NA INSPECTOR P. Jacobs

DATE: START/FINISH 6-4-91 / 6-4-91 CONTRACTOR/DRILLER Twin City Testing

STATIC G.W. DEPTH/DATE 18.6 (FT) / 6-4-91 DRILL RIG TYPE CME 55

DEPTH TO BEDROCK NA (FT) TOTAL DEPTH DRILLED 61.0 (FT)

METHODS:

DRILLING SOIL 6" Hollow Stem Auger Above The Water Table, Then Mud Rotary With 2 15/16" Tricone Bit Below The Water Table

SAMPLING SOIL Standard Split Spoon

DRILLING ROCK

SPECIAL TESTING OR INSTRUMENTATION

COMMENTS:

SUMMARY

SOIL DRILLED 61.0 (FT) ROCK DRILLED NA (FT)

NUMBER SPLIT BARREL SAMPLES 7

NO. & TYPE UNDISTURBED SAMPLES

LIST ALL TEST, INSPECTION AND CALIBRATION RECORDS ATTACHED:

GROUNDWATER (DEPTH BELOW GROUND SURFACE)

DEPTH	DATE	DEPTH	DATE
18.6	6-4-91		

NOTES

1. DATUM IS M.S.L. UNLESS OTHERWISE INDICATED

LEGEND FOR FIELD BORING LOGS

DISTURBED SAMPLES
- W - WASH
- S - SPLIT BARREL (ASTM - D1586 UNLESS OTHERWISE NOTED)
- E - ENVIRONMENTAL
- Z - OTHER, DESCRIBE:

UNDISTURBED SAMPLES
- US - SHELBY TUBE
- UF - STATIONARY (FIXED PISTON)
- UO - OSTERBERG
- UP - PITCHER
- UD - DENISON
- X - OTHER, DESCRIBE:

N - STD PENETRATION RESISTANCE BLOWS/FT (140 LB. HAMMER, UNLESS OTHERWISE NOTED)

REC - RECOVERY

RQD - ROCK QUALITY DESIGNATION

( ) - INCHES OF SAMPLE RECOVERY

PREPARED BY Paul J. Jacobs	DATE 6-19-91
CHECKED BY M. Gene Yow	DATE 6/19/91

document #

FIELD BORING LOG
STONE & WEBSTER ENGINEERING CORP.

SITE ISFSI Project Prairie Island (site 2)	J.O. NO. 12911.5401	BORING NO. B-10 SHEET 2 OF 3
DRILLER Tom White	INSPECTOR P. Jacobs	

SAMPLE DESCRIPTION

PREPARED BY Paul J. Jacobs CHECKED BY M. Gene Yow 6/19/91

DEPTH (FEET)	SAMPLE TYPE	SAMPLE NUMBER	BLOWS/OR REC/RQD	SPT N VALUE	GROUP SYMBOL	SAMPLE DESCRIPTION
	SS	1	1-2-4 (18")	6	SM	Silty Sand - Poorly Graded, Fine with silt, Loose, Moist, Reddish Brown
5						
10	SS	2	1-1-1 (18")	2	SP	Sand - Poorly Graded, Fine with little silt, Loose, Moist, Reddish Brown, Light Brown sand at bottom 1" of sampling barrel
15						
20	SS	3	3-3-4 (18")	7	SP	Sand - Poorly Graded, Fine, Loose, Wet, Light Brown
25						
30	SS	4	10-13-14 (16 1/2")	27	SP	same as above
35						
40	SS	5	16-23-26 (15")	49	SP	same as above
						encountered gravel while drilling at 44.0'

document #

FIELD BORING LOG
STONE & WEBSTER ENGINEERING CORP.

BORING NO. B-10

SITE ISFSI Project Prairie Island (site 2)	J.O. NO. 12911.5401	SHEET 3 OF 3
	DRILLER Tom White	INSPECTOR P. Jacobs

SAMPLE DESCRIPTION

PREPARED BY Paul J. Jacobs CHECKED BY M. Gene Yow 6/19/91

DEPTH (FEET)	SAMPLE TYPE	SAMPLE NUMBER	BLOWS/OR REC/RQD	SPT N VALUE	GROUP SYMBOL	SAMPLE DESCRIPTION
50	SS	6	10-9-11 (12")	20	SW-GW	Gravely Sand - Well Graded, Fine to 1 1/2" gravel, Dense, Wet, Light Brown with Black Gravel
55						
60	SS	7	6-10-13 (9 1/2")	23	SW-GW	same as above - with bottom 4" of sample composed of fine sand T.D. 61.0'
65						
70						
75						
80						
85						

FIELD TEST BORING RECORD COVER SHEET
STONE & WEBSTER ENGINEERING CORP.

		BORING NO. B-11
SITE : ISFSI Project Prairie Island (site 2)	J.O. NO. 12911.5401	SHEET 1 OF 3

COORDINATES N 593,432 E 2,354,406 GROUND ELEV. 695.2

INCLINATION Vertical BEARING NA INSPECTOR P. Jacobs

DATE: START/FINISH 6-5-91 / 6-5-91 CONTRACTOR/DRILLER Twin City Testing

STATIC G.W. DEPTH/DATE 20.7 (FT) / 6-5-91 DRILL RIG TYPE CME 55

DEPTH TO BEDROCK NA (FT) TOTAL DEPTH DRILLED 53.5 (FT)

METHODS:

DRILLING SOIL 6" Hollow Stem Auger Above The Water Table, Then Mud Rotary With 2 15/16" Tricone Bit Below The Water Table

SAMPLING SOIL Standard Split Spoon

DRILLING ROCK

SPECIAL TESTING OR INSTRUMENTATION

COMMENTS:

SUMMARY

SOIL DRILLED 53.5 (FT) ROCK DRILLED NA (FT)

NUMBER SPLIT BARREL SAMPLES 6

NO. & TYPE UNDISTURBED SAMPLES 0

LIST ALL TEST, INSPECTION AND CALIBRATION RECORDS ATTACHED:

GROUNDWATER (DEPTH BELOW GROUND SURFACE)

DEPTH	DATE	DEPTH	DATE
20.7	6-5-91		

NOTES

1. DATUM IS M.S.L. UNLESS OTHERWISE INDICATED

LEGEND FOR FIELD BORING LOGS

DISTURBED SAMPLES
- W - WASH
- S - SPLIT BARREL (ASTM - D1586 UNLESS OTHERWISE NOTED)
- E - ENVIRONMENTAL
- Z - OTHER, DESCRIBE:

UNDISTURBED SAMPLES
- US - SHELBY TUBE
- UF - STATIONARY (FIXED PISTON)
- UO - OSTERBERG
- UP - PITCHER
- UD - DENISON
- X - OTHER , DESCRIBE:

N - STD PENETRATION RESISTANCE BLOWS/FT (140 LB. HAMMER, UNLESS OTHERWISE NOTED)

REC - RECOVERY

RQD - ROCK QUALITY DESIGNATION

( ) - INCHES OF SAMPLE RECOVERY

PREPARED BY Paul A. Jacobs	DATE 6-19-91
CHECKED BY M. Gene Yow	DATE 6/19/91

document #

FIELD BORING LOG
STONE & WEBSTER ENGINEERING CORP.

SITE ISFSI Project Prairie Island (site 2)	J.O. NO. 12911.5401	BORING NO. B-11 SHEET 2 OF 3
DRILLER Tom White	INSPECTOR P. Jacobs	

SAMPLE DESCRIPTION

PREPARED BY Paul A. Jacobs CHECKED BY M. [illegible] 6/19/91

DEPTH (FEET)	SAMPLE TYPE	SAMPLE NUMBER	BLOWS/OR REC/RQD	SPT N VALUE	GROUP SYMBOL	SAMPLE DESCRIPTION
5	SS	1	1-2-2 (18")	4	SP	<u>Sand</u> - Poorly Graded, Fine with minor silt, Loose, Moist, Reddish Brown
10						
	SS	2	1-1-2 (16")	3	SM	<u>Silty Sand</u> - Poorly Graded, Fine with Silt, Loose, Moist, Reddish Brown
15						
						thin gravel layer encountered while drilling at 17.5'
20						
	SS	3	1-2-7 (18")	9	SP	<u>Sand</u> - Poorly Graded, Fine, Med Dense to Loose, Wet, Light Brown
25						
30						
	SS	4	5-7-8 (10")	15	SP	<u>same as above</u>
35						
40						
	SS	5	16-26-30 (13")	56	SP	<u>same as above</u> - with 8 thin horizontal layers of organic deposits

document #

FIELD BORING LOG
STONE & WEBSTER ENGINEERING CORP.

BORING NO. B-11

SITE ISFSI Project Prairie Island (site 2)	**J.O. NO.** 12911.5401	**SHEET** 3 **OF** 3
DRILLER Tom White	**INSPECTOR** P. Jacobs	

SAMPLE DESCRIPTION

PREPARED BY Paul A. Jacobs **CHECKED BY** M. [illegible] 6/19/91

DEPTH (FEET)	SAMPLE TYPE	SAMPLE NUMBER	BLOWS/OR REC/RQD	SPT N VALUE	GROUP SYMBOL	SAMPLE DESCRIPTION
						gravel encountered while drilling at 46.6'
50						
	SS	6	15-16-18 (11")	34	SW-GW	<u>Gravely Sand</u> - Well Graded, Fine to 1 1/4" Gravel, Dense, Wet, Light Brown with Black Gravel
						T.D. 53.3'
55						
60						
65						
70						
75						
80						
85						

document #

FIELD TEST BORING RECORD COVER SHEET
STONE & WEBSTER ENGINEERING CORP.

		BORING NO. B-12
SITE : ISFSI Project Prairie Island (site 2)	J.O. NO. 12911.5401	SHEET 1 OF 3

COORDINATES N 593,482 E 2,354,534 GROUND ELEV. 690.5

INCLINATION Vertical BEARING NA INSPECTOR P. Jacobs

DATE: START/FINISH 6-4-91 / 6-4-91 CONTRACTOR/DRILLER Twin City Testing

STATIC G.W. DEPTH/DATE 17.9 (FT) / 6-4-91 DRILL RIG TYPE CME 55

DEPTH TO BEDROCK NA (FT) TOTAL DEPTH DRILLED 61.0 (FT)

METHODS:

DRILLING SOIL 6" Hollow Stem Auger Above The Water Table, Then Mud Rotary With 2 15/16" Tricone Bit Below The Water Table

SAMPLING SOIL Standard Split Spoon

DRILLING ROCK

SPECIAL TESTING OR INSTRUMENTATION

COMMENTS:

SUMMARY

SOIL DRILLED 61.0 (FT) ROCK DRILLED NA (FT)

NUMBER SPLIT BARREL SAMPLES 13

NO. & TYPE UNDISTURBED SAMPLES 0

LIST ALL TEST, INSPECTION AND CALIBRATION RECORDS ATTACHED:

GROUNDWATER (DEPTH BELOW GROUND SURFACE)

DEPTH	DATE	DEPTH	DATE
17.9	6-4-91		

NOTES

1. DATUM IS M.S.L. UNLESS OTHERWISE INDICATED

LEGEND FOR FIELD BORING LOGS

DISTURBED SAMPLES
- W - WASH
- S - SPLIT BARREL (ASTM - D1586 UNLESS OTHERWISE NOTED)
- E - ENVIRONMENTAL
- Z - OTHER, DESCRIBE:

UNDISTURBED SAMPLES
- US - SHELBY TUBE
- UF - STATIONARY (FIXED PISTON)
- UO - OSTERBERG
- UP - PITCHER
- UD - DENISON
- X - OTHER , DESCRIBE:

N - STD PENETRATION RESISTANCE BLOWS/FT (140 LB. HAMMER, UNLESS OTHERWISE NOTED)

REC - RECOVERY

RQD - ROCK QUALITY DESIGNATION

( ) - INCHES OF SAMPLE RECOVERY

PREPARED BY Paul K. Jacobs	DATE 6-19-91
CHECKED BY M. GaneYow	DATE 6/19/91

document #

FIELD BORING LOG
STONE & WEBSTER ENGINEERING CORP.

BORING NO. B-12

SITE ISFSI Project Prairie Island (site 2)

J.O. NO. 12911.5401

SHEET 2 **OF** 3

DRILLER Tom White

INSPECTOR P. Jacobs

PREPARED BY Paul J. Jacobs **CHECKED BY** M. Gene Yow 6/19/91

DEPTH (FEET)	SAMPLE TYPE	SAMPLE NUMBER	BLOWS/OR REC/RQD	SPT N VALUE	GROUP SYMBOL	SAMPLE DESCRIPTION
	SS	1	1-2-4 (17")	6	SM	Silty Sand - Poorly Graded, Fine with silt, Loose, Moist, Reddish Brown
5	SS	2	1-2-3 (16")	5	SP	Sand - Poorly Graded, Fine with minor silt, Loose, Moist, Reddish Brown
10	SS	3	2-2-1 (18")	3	SM	Silty Sand - Poorly Graded, Very Fine, Loose, Moist, Reddish Brown
15	SS	4	2-2-4 (18")	6	SP	Sand - Poorly Graded, Fine to Medium, Med Dense to Loose, Moist, Light Brown
20	SS	5	1-2-3 (18")	5	SP	same as above
25	SS	6	3-3-4 (12")	7	SP	same as above
30	SS	7	4-4-6 (14 1/2")	10	SP	same as above
35	SS	8	3-3-4 (11")	7	SP	same as above
40	SS	9	4-5-5 (13 1/2")	10	SP	same as above
						encountered gravel while drilling at 44.0'
	SS	10	13-10-7 (9")	17	SW-GW	Gravely Sand - Well Graded, Fine to 1" Gravel, Dense, Wet, Light Brown with Black Gravel

FIELD BORING LOG
STONE & WEBSTER ENGINEERING CORP.

BORING NO. B-12

SITE ISFSI Project Prairie Island (site 2) | J.O. NO. 12911.5401 | SHEET 3 OF 3

DRILLER Tom White | INSPECTOR P. Jacobs

SAMPLE DESCRIPTION

PREPARED BY Paul J. Jacobs CHECKED BY M. Gene Yow 6/19/91

DEPTH (FEET)	SAMPLE TYPE	SAMPLE NUMBER	BLOWS/OR REC/RQD	SPT N VALUE	GROUP SYMBOL	SAMPLE DESCRIPTION
50	SS	11	10-14-11 (10 1/2")	25	SW	Sand - Well Graded, Coarse with minor gravel, Dense, Wet, Light Brown with Black Gravel
55	SS	12	10-13-14 (12")	27	SW-GW	Gravely Sand - as above
60	SS	13	7-13-17 (14")	30	SW	Sand - Well Graded, Coarse with minor gravel, Dense, Light Brown with Black Gravel T.D. 61.0'
65						
70						
75						
80						
85						

document #

FIELD TEST BORING RECORD COVER SHEET
STONE & WEBSTER ENGINEERING CORP.

		BORING NO. B-13
SITE : ISFSI Project Prairie Island (site 2)	J.O. NO. 12911.5401	SHEET 1 OF 3

COORDINATES N 593,532 E 2,354,662 GROUND ELEV. 690.2

INCLINATION Vertical BEARING NA INSPECTOR P. Jacobs

DATE: START/FINISH 6-6-91 / 6-6-91 CONTRACTOR/DRILLER Twin City Testing

STATIC G.W. DEPTH/DATE 16.0 (FT) / 6-6-91 DRILL RIG TYPE CME 55

DEPTH TO BEDROCK NA (FT) TOTAL DEPTH DRILLED 51.0 (FT)

METHODS:

DRILLING SOIL 6" Hollow Stem Auger Above The Water Table, Then Mud Rotary With 2 15/16" Tricone Bit Below The Water Table

SAMPLING SOIL Standard Split Spoon

DRILLING ROCK

SPECIAL TESTING OR INSTRUMENTATION

COMMENTS:

SUMMARY

SOIL DRILLED 51.0 (FT) ROCK DRILLED NA (FT)

NUMBER SPLIT BARREL SAMPLES 7

NO. & TYPE UNDISTURBED SAMPLES 0

LIST ALL TEST, INSPECTION AND CALIBRATION RECORDS ATTACHED:

GROUNDWATER (DEPTH BELOW GROUND SURFACE)

DEPTH	DATE	DEPTH	DATE
16.0	6-6-91		

NOTES

1. DATUM IS M.S.L. UNLESS OTHERWISE INDICATED

LEGEND FOR FIELD BORING LOGS

DISTURBED SAMPLES
- W - WASH
- S - SPLIT BARREL (ASTM - D1586 UNLESS OTHERWISE NOTED)
- E - ENVIRONMENTAL
- Z - OTHER, DESCRIBE:

UNDISTURBED SAMPLES
- US - SHELBY TUBE
- UF - STATIONARY (FIXED PISTON)
- UO - OSTERBERG
- UP - PITCHER
- UD - DENISON
- X - OTHER , DESCRIBE:

N - STD PENETRATION RESISTANCE BLOWS/FT (140 LB. HAMMER, UNLESS OTHERWISE NOTED)

REC - RECOVERY

RQD - ROCK QUALITY DESIGNATION

( ) - INCHES OF SAMPLE RECOVERY

PREPARED BY Paul J. Jacobs	DATE 6-19-91
CHECKED BY M. Gene Yow	DATE 6/19/91

document #

FIELD BORING LOG
STONE & WEBSTER ENGINEERING CORP.

BORING NO. B-13

SITE ISFSI Project Prairie Island (site 2) | J.O. NO. 12911.5401 | SHEET 2 OF 3

DRILLER Tom White | INSPECTOR P. Jacobs

SAMPLE DESCRIPTION

PREPARED BY Paul J. Jacobs CHECKED BY M. Gene Yow 6/19/91

DEPTH (FEET)	SAMPLE TYPE	SAMPLE NUMBER	BLOWS/OR REC/RQD	SPT N VALUE	GROUP SYMBOL	SAMPLE DESCRIPTION
	SS	1	5-6-8 (18")	14	SM	Silty Sand - Poorly Graded, Fine with silt, Loose, Moist, Reddish Brown
5						
	SS	2	2-2-2 (18")	4	SP	Sand - Poorly Graded, Fine with minor silt, Very Loose, Moist, Reddish Brown
10						
	SS	3	1-1-2 (15")	3	SM	Silty Sand - Poorly Graded, Fine with silt, Loose, Moist, Reddish Brown
15						
	SS	4	2-3-5 (18")	8	SP	Sand - Poorly Graded, Fine with traces of gravel, Loose to Med Dense, Wet, Light Brown
20						
25						
30	SS	5	5-6-5 (11 1/2")	11	SP	same as above - no gravel
35						
40	SS	6	4-5-5 (12")	10	SP	same as above
						encountered gravel while drilling at 44.0'

document #

FIELD BORING LOG
STONE & WEBSTER ENGINEERING CORP.

SITE ISFSI Project Prairie Island (site 2)						J.O. NO. 12911.5401	SHEET 3 OF 3
							BORING NO. B-13
						DRILLER Tom White	INSPECTOR P. Jacobs
DEPTH (FEET)	SAMPLE TYPE	SAMPLE NUMBER	BLOWS/OR REC/RQD	SPT N VALUE	GROUP SYMBOL	SAMPLE DESCRIPTION PREPARED BY Paul L. Jacobs	CHECKED BY M. Gene Yow 6/19/91
50	SS	7	11-10-7 (12")	17	SW-GW	Gravely Sand - Well Graded, Fine to 1 1/4" gravel, Dense, Wet, Light Brown with Black Gravel	
						T.D. 51.0'	
55							
60							
65							
70							
75							
80							
85							

document #

FIELD TEST BORING RECORD COVER SHEET
STONE & WEBSTER ENGINEERING CORP.

		BORING NO. B-14
SITE : ISFSI Project Prairie Island (site 2)	J.O. NO. 12911.5401	SHEET 1 OF 3

COORDINATES N 593,482 E 2,354,662 GROUND ELEV. 690.3

INCLINATION Vertical BEARING NA INSPECTOR P. Jacobs

DATE: START/FINISH 6-5-91 / 6-5-91 CONTRACTOR/DRILLER Twin City Testing

STATIC G.W. DEPTH/DATE 18.0 (FT) / 6-5-91 DRILL RIG TYPE CME 55

DEPTH TO BEDROCK NA (FT) TOTAL DEPTH DRILLED 61.0 (FT)

METHODS:

DRILLING SOIL 6" Hollow Stem Auger Above The Water Table, Then Mud Rotary With 2 15/16" Tricone Bit Below The Water Table

SAMPLING SOIL Standard Split Spoon

DRILLING ROCK

SPECIAL TESTING OR INSTRUMENTATION

COMMENTS:

SUMMARY

SOIL DRILLED 61.0 (FT) ROCK DRILLED NA (FT)

NUMBER SPLIT BARREL SAMPLES 7

NO. & TYPE UNDISTURBED SAMPLES 0

LIST ALL TEST, INSPECTION AND CALIBRATION RECORDS ATTACHED:

GROUNDWATER (DEPTH BELOW GROUND SURFACE)

DEPTH	DATE	DEPTH	DATE
18.0	6-5-91		

NOTES

1. DATUM IS M.S.L. UNLESS OTHERWISE INDICATED

LEGEND FOR FIELD BORING LOGS

DISTURBED SAMPLES
- W - WASH
- S - SPLIT BARREL (ASTM - D1586 UNLESS OTHERWISE NOTED)
- E - ENVIRONMENTAL
- Z - OTHER, DESCRIBE:

UNDISTURBED SAMPLES
- US - SHELBY TUBE
- UF - STATIONARY (FIXED PISTON)
- UO - OSTERBERG
- UP - PITCHER
- UD - DENISON
- X - OTHER , DESCRIBE:

N - STD PENETRATION RESISTANCE BLOWS/FT (140 LB. HAMMER, UNLESS OTHERWISE NOTED)

REC - RECOVERY

RQD - ROCK QUALITY DESIGNATION

( ) - INCHES OF SAMPLE RECOVERY

PREPARED BY Paul J. Jacobs	DATE 6-19-91
CHECKED BY M. Gene Yow	DATE 6/19/91

FIELD BORING LOG
STONE & WEBSTER ENGINEERING CORP.

BORING NO. B-14

SITE ISFSI Project Prairie Island (site 2)

J.O. NO. 12911.5401

SHEET 2 **OF** 3

DRILLER Tom White

INSPECTOR P. Jacobs

SAMPLE DESCRIPTION

PREPARED BY Paul J. Jacobs **CHECKED BY** M. Gene Yow 6/19/91

DEPTH (FEET)	SAMPLE TYPE	SAMPLE NUMBER	BLOWS/OR REC/RQD	SPT N VALUE	GROUP SYMBOL	SAMPLE DESCRIPTION
	SS	1	1-3-6 (17")	9	SP	Sand - Poorly Graded, Fine with minor silt, Loose, Moist, Reddish Brown
5						
10	SS	2	1-1-1 (18")	2	SP	same as above - slightly lighter in color
15						
20	SS	3	1-1-3 (18")	4	SP	Sand - Poorly Graded, Fine, Loose, Wet, Light Brown
25						
30	SS	4	7-9-9 (14")	18	SP	same as above
35						
40	SS	5	4-6-6 (12")	12	SP	same as above

document #

FIELD BORING LOG
STONE & WEBSTER ENGINEERING CORP.

BORING NO. B-14

SITE ISFSI Project Prairie Island (site 2)

J.O. NO. 12911.5401

SHEET 3 **OF** 3

DRILLER Tom White

INSPECTOR P. Jacobs

SAMPLE DESCRIPTION

PREPARED BY Paul A. Jacobs **CHECKED BY** M. Gene Yow 6/19/91

DEPTH (FEET)	SAMPLE TYPE	SAMPLE NUMBER	BLOWS/OR REC/RQD	SPT N VALUE	GROUP SYMBOL	SAMPLE DESCRIPTION
						encountered gravel while drilling at 46.0'
50	SS	6	17-16-12 (9")	28	SW-GW	Gravely Sand - Well Graded, Fine to 1 3/4" gravel, Dense, Wet, Light Brown with Black Gravel
55						
60	SS	7	10-12-13 (9")	25	SW-GW	same as above - with 3/4" gravel max T.D. 61.0'
65						
70						
75						
80						
85						

document #

FIELD TEST BORING RECORD COVER SHEET
STONE & WEBSTER ENGINEERING CORP.

		BORING NO. B-15
SITE : ISFSI Project Prairie Island (site 2)	J.O. NO. 12911.5401	SHEET 1 OF 3

COORDINATES N 593,432 E 2,354,652 GROUND ELEV. 690.0

INCLINATION Vertical BEARING NA INSPECTOR P. Jacobs

DATE: START/FINISH 6-5-91 / 6-5-91 CONTRACTOR/DRILLER Twin City Testing

STATIC G.W. DEPTH/DATE 18.1 (FT) / 6-5-91 DRILL RIG TYPE CME 55

DEPTH TO BEDROCK NA (FT) TOTAL DEPTH DRILLED 51.0 (FT)

METHODS:

DRILLING SOIL 6" Hollow Stem Auger Above The Water Table, Then Mud Rotary With 2 15/16" Tricone Bit Below The Water Table

SAMPLING SOIL Standard Split Spoon

DRILLING ROCK

SPECIAL TESTING OR INSTRUMENTATION

COMMENTS:

SUMMARY

SOIL DRILLED 51.0 (FT) ROCK DRILLED NA (FT)

NUMBER SPLIT BARREL SAMPLES 6

NO. & TYPE UNDISTURBED SAMPLES

LIST ALL TEST, INSPECTION AND CALIBRATION RECORDS ATTACHED:

GROUNDWATER (DEPTH BELOW GROUND SURFACE)

DEPTH	DATE	DEPTH	DATE
18.1	6-5-91		

NOTES

1. DATUM IS M.S.L. UNLESS OTHERWISE INDICATED

LEGEND FOR FIELD BORING LOGS

DISTURBED SAMPLES
- W - WASH
- S - SPLIT BARREL (ASTM - D1586 UNLESS OTHERWISE NOTED)
- E - ENVIRONMENTAL
- Z - OTHER, DESCRIBE:

UNDISTURBED SAMPLES
- US - SHELBY TUBE
- UF - STATIONARY (FIXED PISTON)
- UO - OSTERBERG
- UP - PITCHER
- UD - DENISON
- X - OTHER , DESCRIBE:

N - STD PENETRATION RESISTANCE BLOWS/FT (140 LB. HAMMER, UNLESS OTHERWISE NOTED)

REC - RECOVERY

RQD - ROCK QUALITY DESIGNATION

( ) - INCHES OF SAMPLE RECOVERY

PREPARED BY Paul J. Jacobs	DATE 6-19-91
CHECKED BY M. Gene Yow	DATE 6/19/91

document #

FIELD BORING LOG
STONE & WEBSTER ENGINEERING CORP.

SITE ISFSI Project Prairie Island (site 2)	J.O. NO. 12911.5401	BORING NO. B-15 SHEET 2 OF 3
DRILLER Tom White	INSPECTOR P. Jacobs	

SAMPLE DESCRIPTION
PREPARED BY Paul J. Jacobs CHECKED BY M. Gene Yow 6/19/91

DEPTH (FEET)	SAMPLE TYPE	SAMPLE NUMBER	BLOWS/OR REC/RQD	SPT N VALUE	GROUP SYMBOL	SAMPLE DESCRIPTION
5	SS	1	3-4-3 (15")	7	SM	Silty Sand - Poorly Graded, Fine with silt, Loose, Moist, Reddish Brown
10						
						thin gravel layer encountered while drilling at 12.5'
15	SS	2	3-5-7 (16")	12	SP	Sand - Poorly Graded, Fine, Loose to Med Dense Moist, Light Brown
20	SS	3	1-2-2 (18")	4	SP	same as above - wet
25						
30	SS	4	6-12-13 (11 1/2")	25	SP	same as above
35						
40	SS	5	5-7-10 (9 1/2")	17	SP	same as above
						encountered gravel while drilling at 44.0'

Rev. 2 9/91

document #

FIELD BORING LOG
STONE & WEBSTER ENGINEERING CORP.

BORING NO. B-15	
SITE ISFSI Project Prairie Island (site 2)	J.O. NO. 12911.5401
SHEET 3 OF 3	
DRILLER Tom White	INSPECTOR P. Jacobs

SAMPLE DESCRIPTION

PREPARED BY Paul J. Jacobs CHECKED BY M. Gene Yow 6/19/91

DEPTH (FEET)	SAMPLE TYPE	SAMPLE NUMBER	BLOWS/OR REC/RQD	SPT N VALUE	GROUP SYMBOL	SAMPLE DESCRIPTION
50	SS	6	12-16-15 (9 1/2")	31	SW-GW	Gravely Sand - Well Graded, Fine to 1" gravel, Dense, Wet, Light Brown with Black Gravel
						T.D. 51.0'
55						
60						
65						
70						
75						
80						
85						

document #

FIELD TEST BORING RECORD COVER SHEET
STONE & WEBSTER ENGINEERING CORP.

SITE : ISFSI Project Prairie Island (site 2)	J.O. NO. 12911.5401	BORING NO. B-16 / SHEET 1 OF 3

COORDINATES N 593,482 E 2,354,770 GROUND ELEV. 693.0

INCLINATION Vertical BEARING NA INSPECTOR P. Jacobs

DATE: START/FINISH 6-3-91 / 6-3-91 CONTRACTOR/DRILLER Twin City Testing

STATIC G.W. DEPTH/DATE 20.4 (FT) / 6-3-91 DRILL RIG TYPE CME 55

DEPTH TO BEDROCK NA (FT) TOTAL DEPTH DRILLED 61.0 (FT)

METHODS:

DRILLING SOIL 6" Hollow Stem Auger Above The Water Table, Then Mud Rotary With 2 15/16" Tricone Bit Below The Water Table

SAMPLING SOIL Standard Split Spoon

DRILLING ROCK

SPECIAL TESTING OR INSTRUMENTATION

COMMENTS:

SUMMARY

SOIL DRILLED 61.0 (FT) ROCK DRILLED NA (FT)

NUMBER SPLIT BARREL SAMPLES 13

NO. & TYPE UNDISTURBED SAMPLES 0

LIST ALL TEST, INSPECTION AND CALIBRATION RECORDS ATTACHED:

GROUNDWATER (DEPTH BELOW GROUND SURFACE)

DEPTH	DATE	DEPTH	DATE
20.4	6-3-91		

NOTES

1. DATUM IS M.S.L. UNLESS OTHERWISE INDICATED

LEGEND FOR FIELD BORING LOGS

DISTURBED SAMPLES
- W - WASH
- S - SPLIT BARREL (ASTM - D1586 UNLESS OTHERWISE NOTED)
- E - ENVIRONMENTAL
- Z - OTHER, DESCRIBE:

UNDISTURBED SAMPLES
- US - SHELBY TUBE
- UF - STATIONARY (FIXED PISTON)
- UO - OSTERBERG
- UP - PITCHER
- UD - DENISON
- X - OTHER , DESCRIBE:

N - STD PENETRATION RESISTANCE BLOWS/FT (140 LB. HAMMER, UNLESS OTHERWISE NOTED)

REC - RECOVERY

RQD - ROCK QUALITY DESIGNATION

( ) - INCHES OF SAMPLE RECOVERY

PREPARED BY Paul J. Jacobs DATE 6-19-91

CHECKED BY M. Genelow DATE 6/19/91

FIELD BORING LOG
STONE & WEBSTER ENGINEERING CORP.

BORING NO. B-16

SITE ISFSI Project Prairie Island (site 2) | J.O. NO. 12911.5401 | SHEET 2 OF 3

DRILLER Tom White | INSPECTOR P. Jacobs

SAMPLE DESCRIPTION

PREPARED BY Paul J. Jacobs CHECKED BY M. Gene Yow 6/19/91

DEPTH (FEET)	SAMPLE TYPE	SAMPLE NUMBER	BLOWS/OR REC/RQD	SPT N VALUE	GROUP SYMBOL	SAMPLE DESCRIPTION
	SS	1	2-7-12 (13")	19		Fill - Boring located in the middle of a gravel roadway
5	SS	2	1-2-3 (15")	5	SP	Sand - Poorly Graded, Fine, Loose, Moist Reddish Brown
10	SS	3	1-2-2 (13")	4	SP	same as above
15	SS	4	3-5-4 (14")	9	SP	same as above - with a thin gravel seam at 15.1'
20	SS	5	3-5-6 (14")	11	SP	same as above
25	SS	6	3-4-5 (7")	9	SP	Sand - Poorly Graded, Fine, Med Dense, Wet, Light Brown
30	SS	7	7-10-12 (13")	22	SP	same as above
35	SS	8	3-5-5 (8 1/2")	10	SP	same as above
40	SS	9	3-5-7 (9")	12	SP	same as above - with thin reddish brown sand layer at 40.5'
	SS	10	5-10-12 (11")	22	SP	same as above

FIELD BORING LOG
STONE & WEBSTER ENGINEERING CORP.

BORING NO. B-16		
SITE ISFSI Project Prairie Island (site 2)	J.O. NO. 12911.5401	SHEET 3 OF 3
DRILLER Tom White	INSPECTOR P. Jacobs	

SAMPLE DESCRIPTION
PREPARED BY Paul A. Jacobs CHECKED BY M. Gene Yow 6/19/91

DEPTH (FEET)	SAMPLE TYPE	SAMPLE NUMBER	BLOWS/OR REC/RQD	SPT N VALUE	GROUP SYMBOL	SAMPLE DESCRIPTION
50	SS	11	22-18-15 (5")	33	SW-GW	<u>Gravely Sand</u> - Well Graded, Fine to 1" gravel, Dense, Wet, Light Brown with Black Gravel
55	SS	12	10-14-13 (11")	27	SW-GW	<u>same as above</u>
60	SS	13	10-14-17 (9 1/2")	31	SW-GW	<u>same as above</u> T.D. 61.0'
65						
70						
75						
80						
85						

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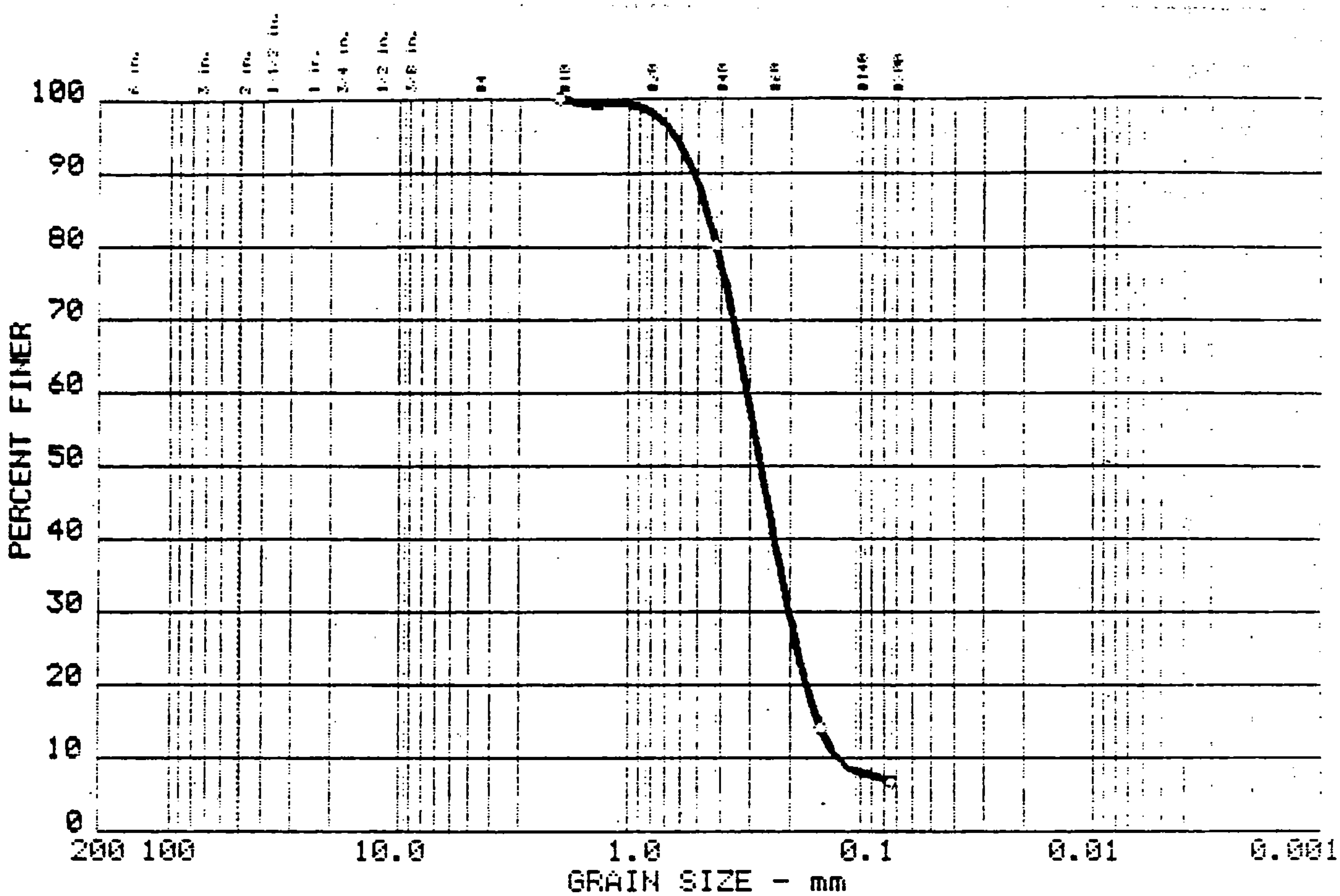
APPENDIX 2B

GRAIN SIZE DISTRIBUTION TEST REPORTS

GRAIN SIZE DISTRIBUTION TEST REPORT



Test	%+75mm	% GRAVEL	% SAND	% SILT	% CLAY
○ 1	0.0	0.0	93.2	6.8	

	LL	PI	D_{85}	D_{60}	D_{50}	D_{30}	D_{15}	D_{10}	C_c	C_u
○			0.46	0.31	0.27	0.203	0.1531	0.1259	1.07	2.4

MATERIAL DESCRIPTION	USCS	AASHTO
○ SAND W/SILT, FINE GRAINED	SP-SM	

Project No.: 4220 91-1181
Project: SPENT FUEL ROD VAULT
○ Location: PRAIRIE ISLAND NUCLEAR PLANT

Date: 06/21/91

Remarks:
BORING NO.: 8
SAMPLE NO.: 1
DEPTH (ft): 0 - 1.5

GRAIN SIZE DISTRIBUTION TEST REPORT
TWIN CITY TESTING CORPORATION

Figure No.

GRAIN SIZE DISTRIBUTION TEST DATA	Test No.: 1

ate: 06/21/91
ct No.: 4220 91-1181
Pr ct: SPENT FUEL ROD VAULT

Sample Data

ocation of Sample: PRAIRIE ISLAND NUCLEAR PLANT
Sample Description: SAND W/SILT, FINE GRAINED
SCS Class: SP-SM   Liquid limit:
SHTO Class: Plasticity index:

Notes

emarks: BORING NO.: 8   SAMPLE NO.: 1
DEPTH (ft): 0 - 1.

g. No.:

Mechanical Analysis Data

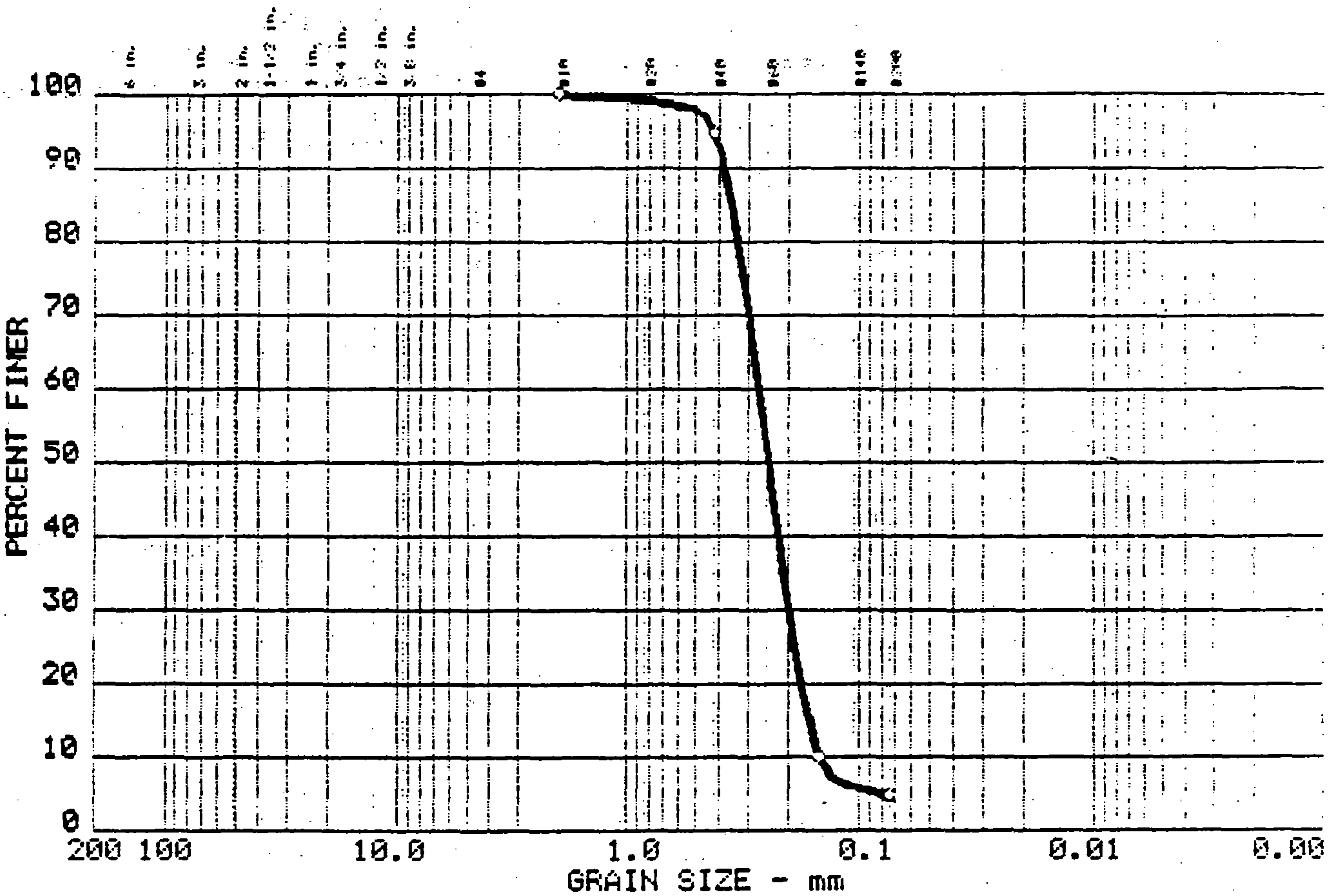
eve	Size, mm	Percent finer
= 10	2.000	100.0
= 40	0.420	80.3
100	0.149	14.0
200	0.074	6.8

Fractional Components

% + 3 in. = 0.0   % GRAVEL = 0.0   % SAND = 93.2
% FINES = 6.8

35= 0.46   D60= 0.307   D50= 0.269
D30= 0.2030   D15= 0.15311   D10= 0.12589
= 1.0654   Cu = 2.4406

GRAIN SIZE DISTRIBUTION TEST REPORT



Test	%+75mm	% GRAVEL	% SAND	% SILT	% CLAY
○ 2	0.0	0.0	95.2	4.8	

	LL	PI	D_{85}	D_{60}	D_{50}	D_{30}	D_{15}	D_{10}	C_c	C_u
○			0.35	0.27	0.24	0.200	0.1646	0.1484	1.00	1.8

MATERIAL DESCRIPTION	USCS	AASHTO
○ SAND, FINE GRAINED	SP	

Project No.: 4220 91-1181
Project: SPENT FUEL ROD VAULT
○ Location: PRAIRIE ISLAND NUCLEAR PLANT

Date: 06/21/91

Remarks:
BORING NO.: 9
SAMPLE NO.: 4
DEPTH (ft): 29.5 - 31

GRAIN SIZE DISTRIBUTION TEST REPORT
TWIN CITY TESTING CORPORATION

Figure No.

GRAIN SIZE DISTRIBUTION TEST DATA   Test No.: 2

ate: 06/21/91
r t No.: 4220 91-1181
rc t: SPENT FUEL ROD VAULT

Sample Data

ocation of Sample: PRAIRIE ISLAND NUCLEAR PLANT
mple Description: SAND, FINE GRAINED
CS Class: SP   Liquid limit:
SHTO Class: Plasticity index:

Notes

marks: BORING NO.: 9 SAMPLE NO.: 4
DEPTH (ft): 29.5 - 31
g. No.:

Mechanical Analysis Data

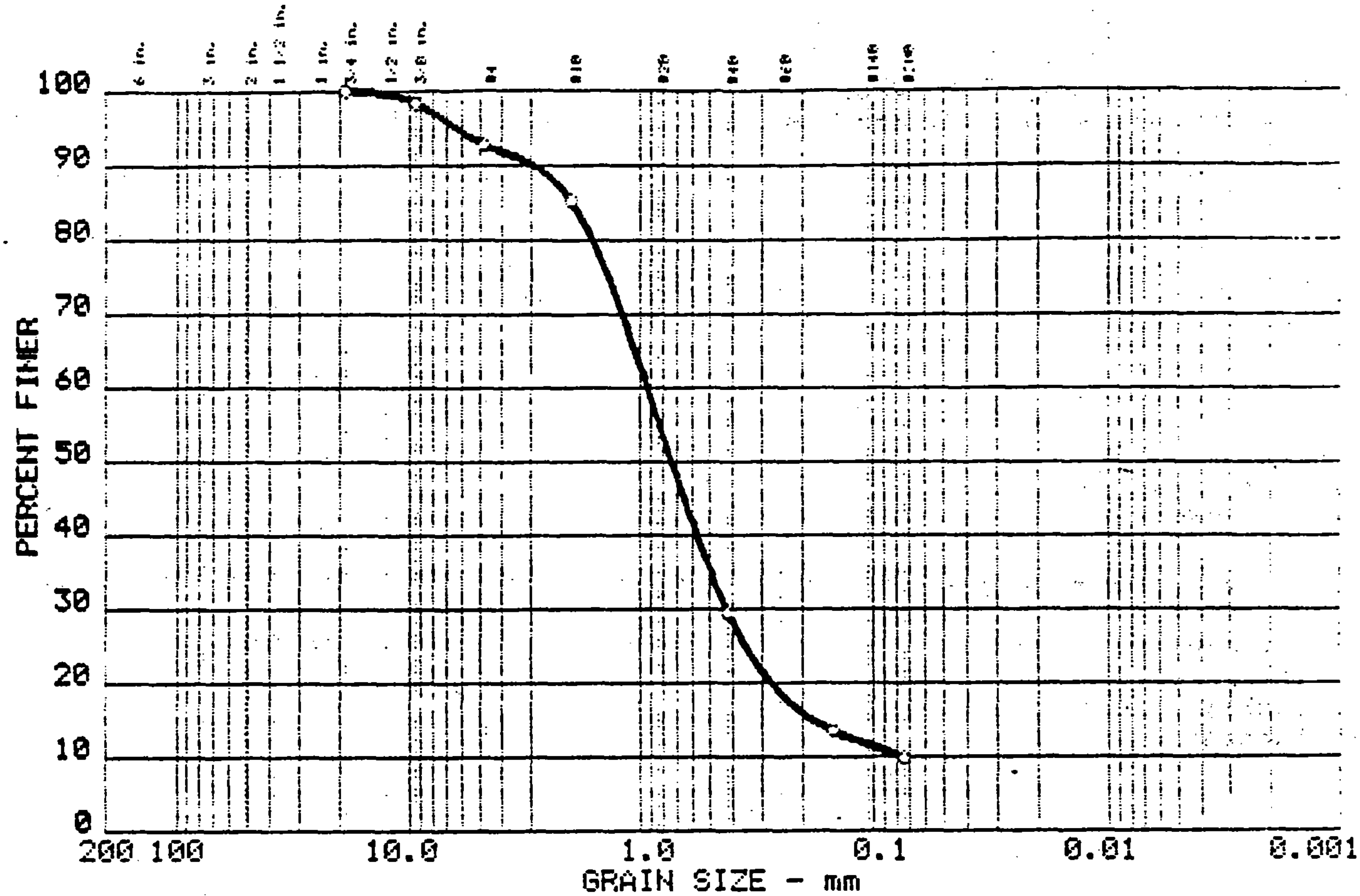
eve	Size, mm	Percent finer
10	2.000	100.0
40	0.420	94.8
100	0.149	10.1
200	0.074	4.8

Fractional Components

+ 3 in. = 0.0   % GRAVEL = 0.0   % SAND = 95.2
FINES = 4.8

5= 0.35 D60= 0.268 D50= 0.244
0= 0.1995 D15= 0.16463 D10= 0.14842
= 1.0000 Cu = 1.8072

GRAIN SIZE DISTRIBUTION TEST REPORT



	Test	%+75mm	% GRAVEL	% SAND	% SILT	% CLAY
○	3	0.0	7.3	82.8	9.9	

	LL	PI	D_{85}	D_{60}	D_{50}	D_{30}	D_{15}	D_{10}	C_c	C_u
○			1.97	0.94	0.74	0.429	0.1807	0.0745	2.63	12.6

MATERIAL DESCRIPTION	USCS	AASHTO
○ SAND W/SILT & W/A LITTLE GRAVEL, MEDIUM GRAINED	SP-SM	

Project No.: 4220 91-1181
Project: SPENT FUEL ROD VAULT
○ Location: PRAIRIE ISLAND NUCLEAR PLANT

Date: 06/21/91

Remarks:
BORING NO.: 9
SAMPLE NO.: 6
DEPTH (ft): 49.5 - 51

GRAIN SIZE DISTRIBUTION TEST REPORT
TWIN CITY TESTING CORPORATION

Figure No.

GRAIN SIZE DISTRIBUTION TEST DATA Test No.: 3

ate: 06/21/91
r ct No.: 4220 91-1181
roject: SPENT FUEL ROD VAULT

Sample Data

ocation of Sample: PRAIRIE ISLAND NUCLEAR PLANT
ample Description: SAND W/SILT & W/A LITTLE GRAVEL, MEDIUM GRAINED
SCS Class: SP-SM Liquid limit:
ASHTO Class: Plasticity index:

Notes

emarks: BORING NO.: 9 SAMPLE NO.: 6
DEPTH (ft): 49.5 - 51
g. No.:

Mechanical Analysis Data

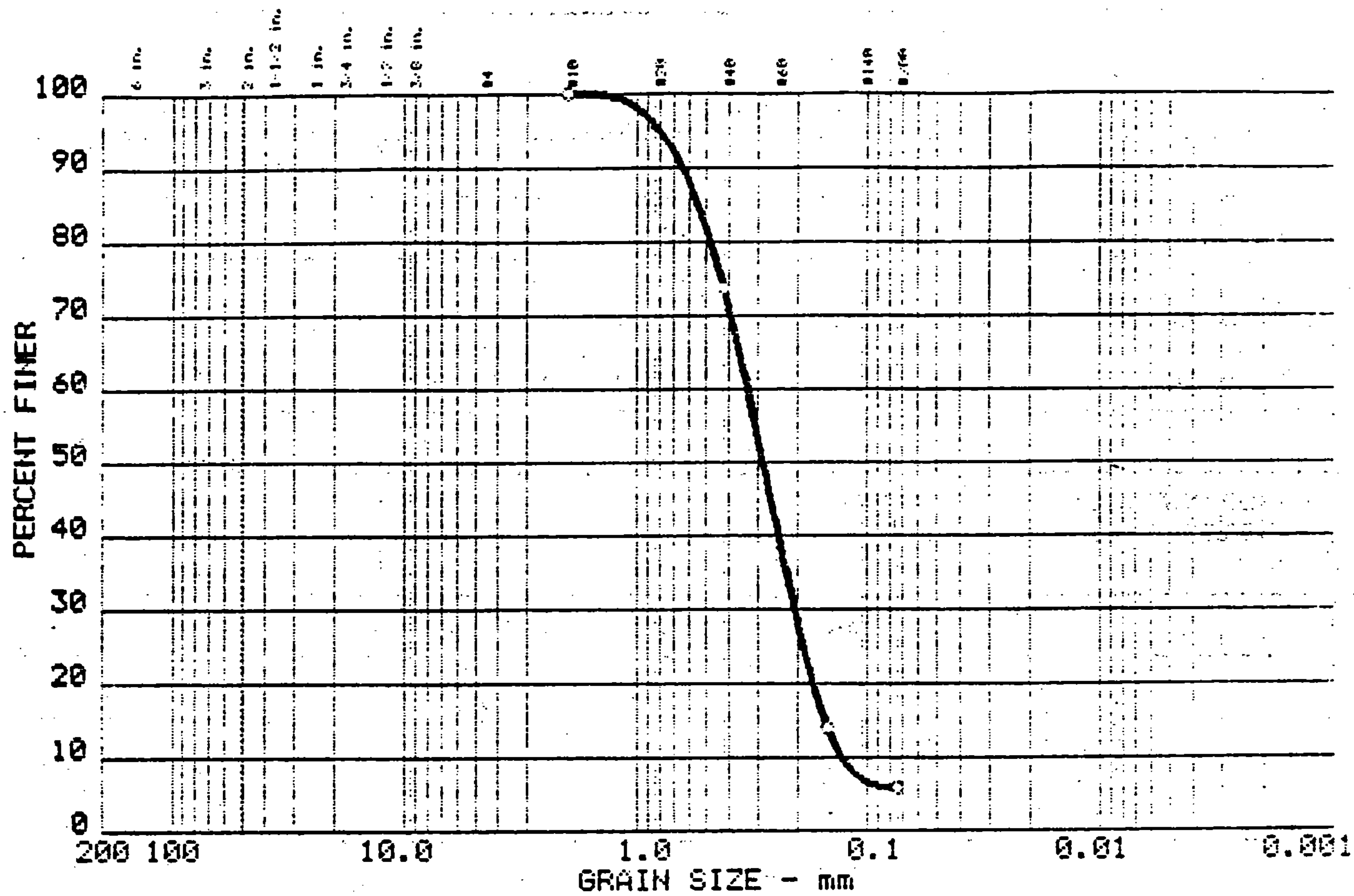
eve	Size, mm	Percent finer
75 inches	19.05	100.0
375 inches	9.53	98.4
4	4.760	92.7
10	2.000	85.2
40	0.420	29.4
	0.149	13.5
200	0.074	9.9

Fractional Components

+ 3 in. = 0.0 % GRAVEL = 7.3 % SAND = 82.8
FINES = 9.9

5= 1.97 D60= 0.938 D50= 0.736
0= 0.4285 D15= 0.18072 D10= 0.07447
= 2.6303 Cu = 12.5893

GRAIN SIZE DISTRIBUTION TEST REPORT



Test	%+75 mm	% GRAVEL	% SAND	% SILT	% CLAY
○ 4	0.0	0.0	94.1	5.9	

	LL	PI	D_{85}	D_{60}	D_{50}	D_{30}	D_{15}	D_{10}	C_c	C_u
○			0.54	0.33	0.28	0.207	0.1531	0.1288	1.01	2.6

MATERIAL DESCRIPTION	USCS	AASHTO
○ SAND W/SILT, FINE GRAINED	SP-SM	

Project No.: 4220 91-1181
Project: SPENT FUEL ROD VAULT
○ Location: PRAIRIE ISLAND NUCLEAR PLANT

Date: 06/21/91

GRAIN SIZE DISTRIBUTION TEST REPORT
TWIN CITY TESTING CORPORATION

Remarks:
BORING NO.: 10
SAMPLE NO.: 1
DEPTH (ft): 0 - 1.5

Figure No.

GRAIN SIZE DISTRIBUTION TEST DATA Test No.: 4

ate: 06/21/91
r ct No.: 4220 91-1181
ct: SPENT FUEL ROD VAULT

Sample Data

ocation of Sample: PRAIRIE ISLAND NUCLEAR PLANT
ample Description: SAND W/SILT, FINE GRAINED
SCS Class: SP-SM Liquid limit:
ASHTO Class: Plasticity index:

Notes

emarks: BORING NO.: 10 SAMPLE NO.: 1
DEPTH (ft): 0 - 1.5
ig. No.:

Mechanical Analysis Data

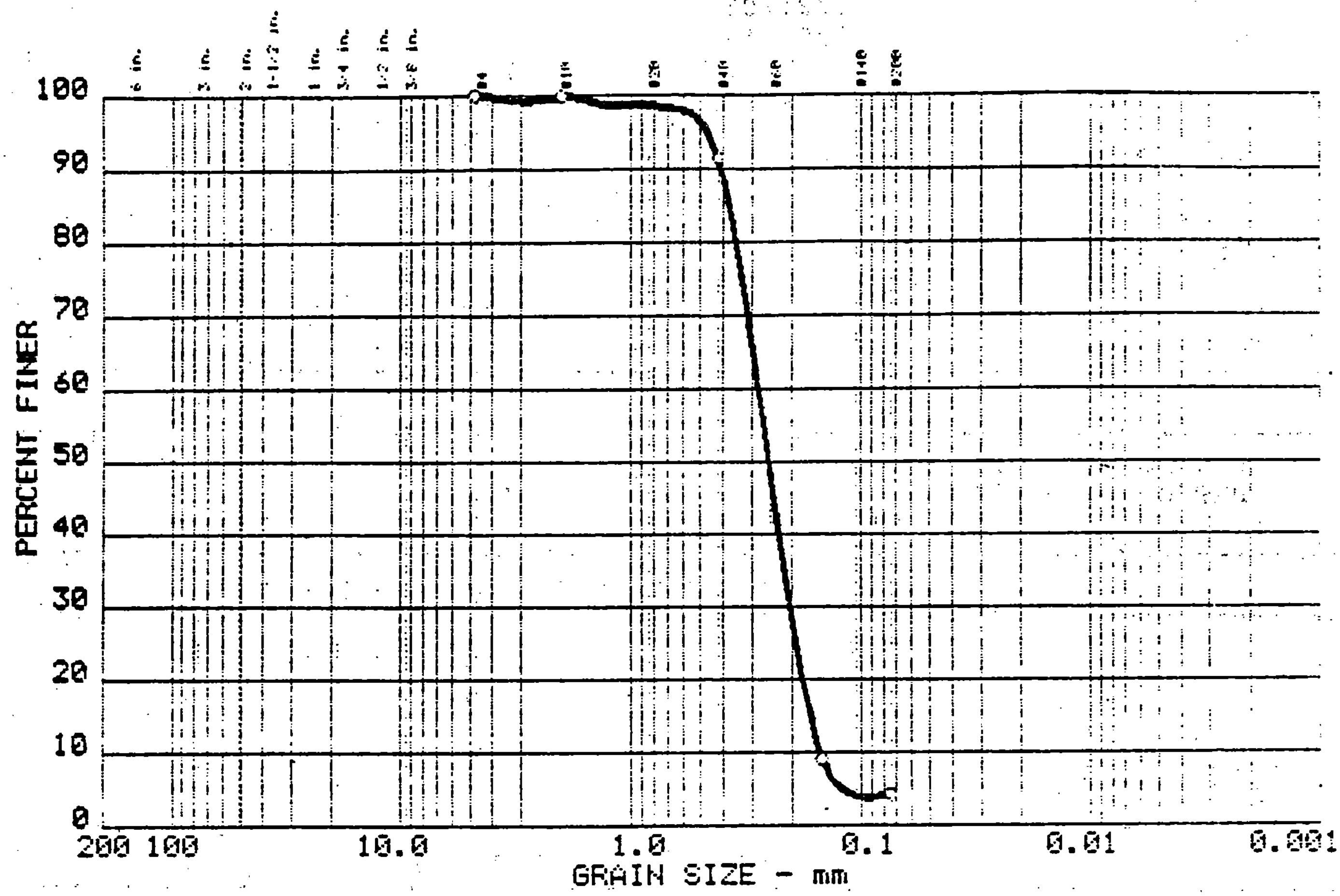
ieve	Size, mm	Percent finer
10	2.000	100.0
40	0.420	73.7
100	0.149	14.0
200	0.074	5.9

Fractional Components

+ 3 in. = 0.0 % GRAVEL = 0.0 % SAND = 94.1
FINES = 5.9

35= 0.54 D60= 0.331 D50= 0.284
30= 0.2073 D15= 0.15311 D10= 0.12882
= 1.0069 Cu = 2.5704

GRAIN SIZE DISTRIBUTION TEST REPORT



Test	%+75mm	% GRAVEL	% SAND	% SILT	% CLAY
○ 5	0.0	0.0	95.7	4.3	

	LL	PI	D_{85}	D_{60}	D_{50}	D_{30}	D_{15}	D_{10}	C_c	C_u
○			0.37	0.28	0.25	0.204	0.1679	0.1528	0.98	1.8

MATERIAL DESCRIPTION	USCS	AASHTO
○ SAND, FINE GRAINED	SP	

Project No.: 4220 91-1181
Project: SPENT FUEL ROD VAULT
○ Location: PRAIRIE ISLAND NUCLEAR PLANT

Date: 06/21/91

GRAIN SIZE DISTRIBUTION TEST REPORT
TWIN CITY TESTING CORPORATION

Remarks:
BORING NO.: 11
SAMPLE NO.: 5
DEPTH (ft): 22 - 23.5

Figure No.

GRAIN SIZE DISTRIBUTION TEST DATA Test No.: 5

ıte: 06/21/91
r ct No.: 4220 91-1181
rc ct: SPENT FUEL ROD VAULT

Sample Data

ocation of Sample: PRAIRIE ISLAND NUCLEAR PLANT
ample Description: SAND, FINE GRAINED
SCS Class: SP Liquid limit:
ASHTO Class: Plasticity index:

Notes

emarks: BORING NO.: 11 SAMPLE NO.: 5
DEPTH (ft): 22 - 23.5
g. No.:

Mechanical Analysis Data

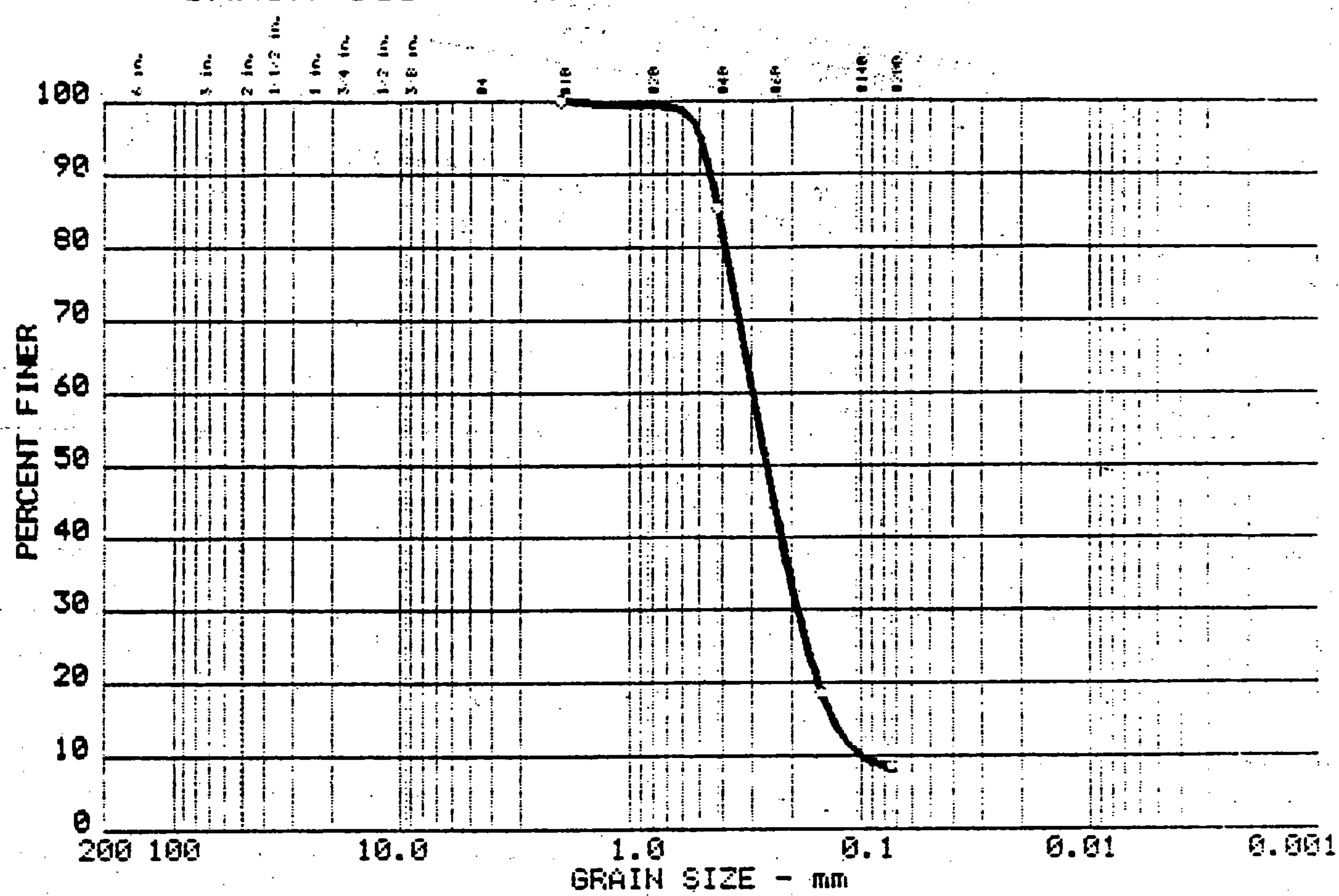
ieve	Size, mm	Percent finer
4	4.760	100.0
10	2.000	99.9
40	0.420	91.4
100	0.149	9.0
200	0.074	4.3

Fractional Components

+ 3 in. = 0.0 % GRAVEL = 0.0 % SAND = 95.7
FINES = 4.3

85= 0.37 D60= 0.279 D50= 0.252
30= 0.2044 D15= 0.16788 D10= 0.15276
c = 0.9795 Cu = 1.8281

GRAIN SIZE DISTRIBUTION TEST REPORT



Test	%+75mm	% GRAVEL	% SAND	% SILT	% CLAY
○ 6	0.0	0.0	91.5	8.5	

	LL	PI	D85	D60	D50	D30	D15	D10	Cc	Cu
○			0.42	0.29	0.26	0.189	0.1338	0.1003	1.21	2.9

MATERIAL DESCRIPTION	USCS	AASHTO
○ SAND W/SILT, FINE GRAINED	SP-SM	

Project No.: 4220 91-1181
Project: SPENT FUEL ROD VAULT
○ Location: PRAIRIE ISLAND NUCLEAR PLANT

Date: 06/21/91

GRAIN SIZE DISTRIBUTION TEST REPORT
TWIN CITY TESTING CORPORATION

Remarks:
BORING NO.: 12
SAMPLE NO.: 1
DEPTH (ft): 0 - 1.5

Figure No.

GRAIN SIZE DISTRIBUTION TEST DATA

Test No.: 6

ate: 06/21/91
r ct No.: 4220 91-1181
Pro ct: SPENT FUEL ROD VAULT

Sample Data

ocation of Sample: PRAIRIE ISLAND NUCLEAR PLANT
ample Description: SAND W/SILT, FINE GRAINED
CS Class: SP-SM   Liquid limit:
ASHTO Class: Plasticity index:

Notes

emarks: BORING NO.: 12 SAMPLE NO.: 1
DEPTH (ft): 0 - 1.5
g. No.:

Mechanical Analysis Data

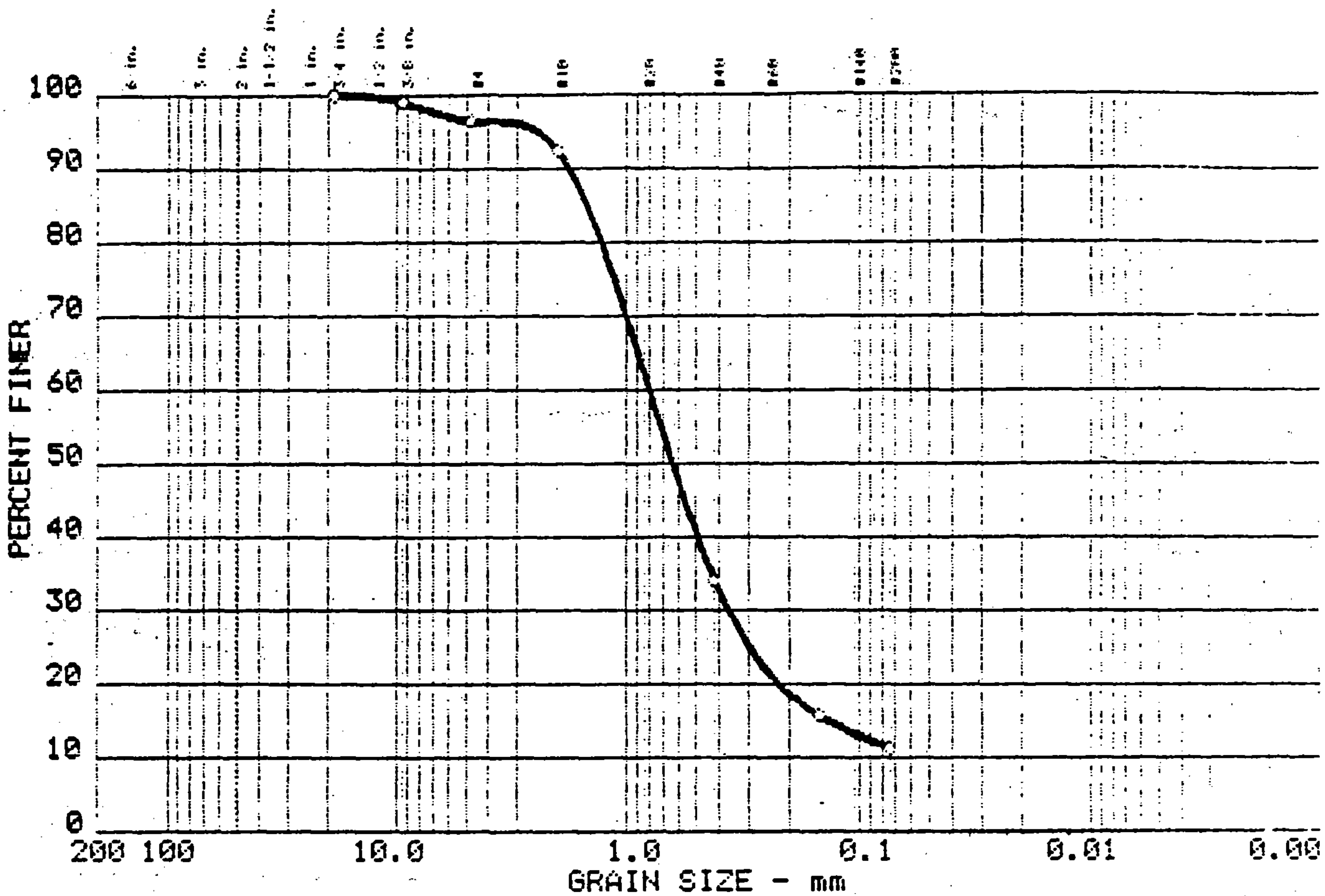
eve	Size, mm	Percent finer
10	2.000	100.0
40	0.420	85.5
100	0.149	18.5
200	0.074	8.5

Fractional Components

% + 3 in. = 0.0   % GRAVEL = 0.0   % SAND = 91.5
FINES = 8.5

D85= 0.42 D60= 0.294 D50= 0.256
D30= 0.1890 D15= 0.13381 D10= 0.10035
 = 1.2106 Cu = 2.9309

GRAIN SIZE DISTRIBUTION TEST REPORT



Test	%+75mm	% GRAVEL	% SAND	% SILT	% CLAY
○ 7	0.0	3.6	85.2	11.2	

	LL	PI	D_{85}	D_{60}	D_{50}	D_{30}	D_{15}	D_{10}	C_c	C_u
○			1.50	0.80	0.64	0.366	0.1358			

MATERIAL DESCRIPTION	USCS	AASHTO
○ SAND W/SILT & W/A LITTLE GRAVEL, MEDIUM GRAINED	SP-SM	

Project No.: 4220 91-1181
Project: SPENT FUEL ROD VAULT
○ Location: PRAIRIE ISLAND NUCLEAR PLANT

Date: 06/21/91

Remarks:
BORING NO.: 12
SAMPLE NO.: 12
DEPTH (ft): 54.5 - 56

GRAIN SIZE DISTRIBUTION TEST REPORT
TWIN CITY TESTING CORPORATION

Figure No.

GRAIN SIZE DISTRIBUTION TEST DATA

Test No.: 7

ıte: 06/21/91
r ct No.: 4220 91-1181
Pr. .ct: SPENT FUEL ROD VAULT

Sample Data

ocation of Sample: PRAIRIE ISLAND NUCLEAR PLANT
ample Description: SAND W/SILT & W/A LITTLE GRAVEL, MEDIUM GRAINED
CS Class: SP-SM   Liquid limit:
ASHTO Class: Plasticity index:

Notes

emarks: BORING NO.: 12 SAMPLE NO.: 12
DEPTH (ft): 54.5 - 56
.g. No.:

Mechanical Analysis Data

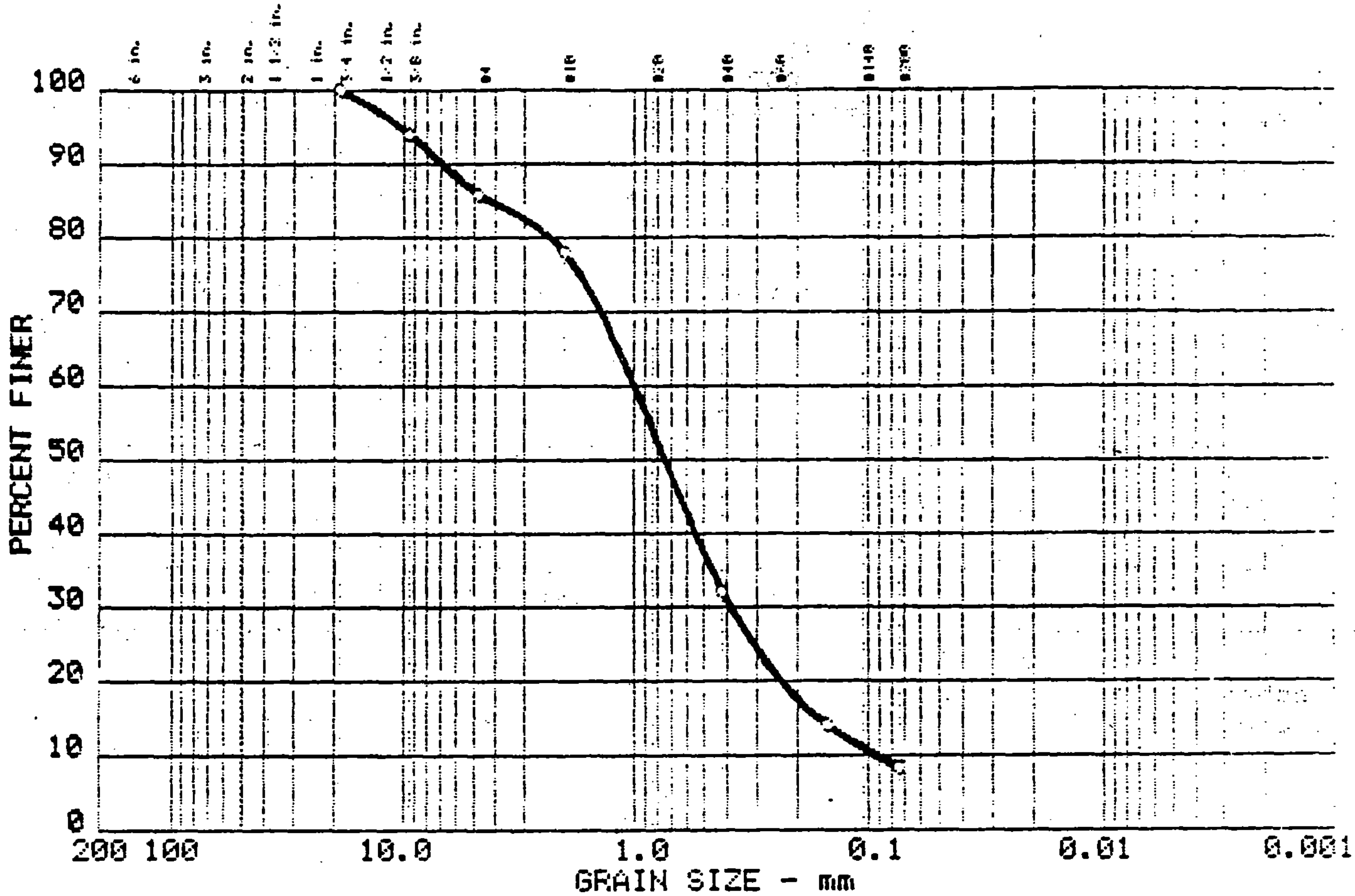
.eve	Size, mm	Percent finer
.75 inches	19.05	100.0
375 inches	9.53	99.0
4	4.760	96.4
10	2.000	92.4
40	0.420	34.1
	0.149	15.7
	0.074	11.2

Fractional Components

% + 3 in. = 0.0 % GRAVEL = 3.6 % SAND = 85.2
% FINES = 11.2

.35= 1.50 D60= 0.800 D50= 0.635
D30= 0.3656 D15= 0.13583

GRAIN SIZE DISTRIBUTION TEST REPORT



	Test	%+75 mm	% GRAVEL	% SAND	% SILT	% CLAY
○	8	0.0	14.1	77.5	8.4	

	LL	PI	D_{85}	D_{60}	D_{50}	D_{30}	D_{15}	D_{10}	C_c	C_u
○			4.22	1.00	0.74	0.385	0.1622	0.0923	1.60	10.8

MATERIAL DESCRIPTION	USCS	AASHTO
○ SAND W/SILT & W/A LITTLE GRAVEL, MEDIUM GRAINED	SP-SM	

Project No.: 4220 91-1181
Project: SPENT FUEL ROD VAULT
○ Location: PRAIRIE ISLAND NUCLEAR PLANT

Date: 06/21/91

Remarks:
BORING NO.: 13
SAMPLE NO.: 7
DEPTH (ft): 49 - 51

GRAIN SIZE DISTRIBUTION TEST REPORT
TWIN CITY TESTING CORPORATION

Figure No.

GRAIN SIZE DISTRIBUTION TEST DATA

Test No.: 8

te: 06/21/91
ct No.: 4220 91-1181
ct: SPENT FUEL ROD VAULT

Sample Data

cation of Sample: PRAIRIE ISLAND NUCLEAR PLANT
mple Description: SAND W/SILT & W/A LITTLE GRAVEL, MEDIUM GRAINED
CS Class: SP-SM Liquid limit:
SHTO Class: Plasticity index:

Notes

emarks: BORING NO.: 13 SAMPLE NO.: 7
DEPTH (ft): 49 - 51
g. No.:

Mechanical Analysis Data

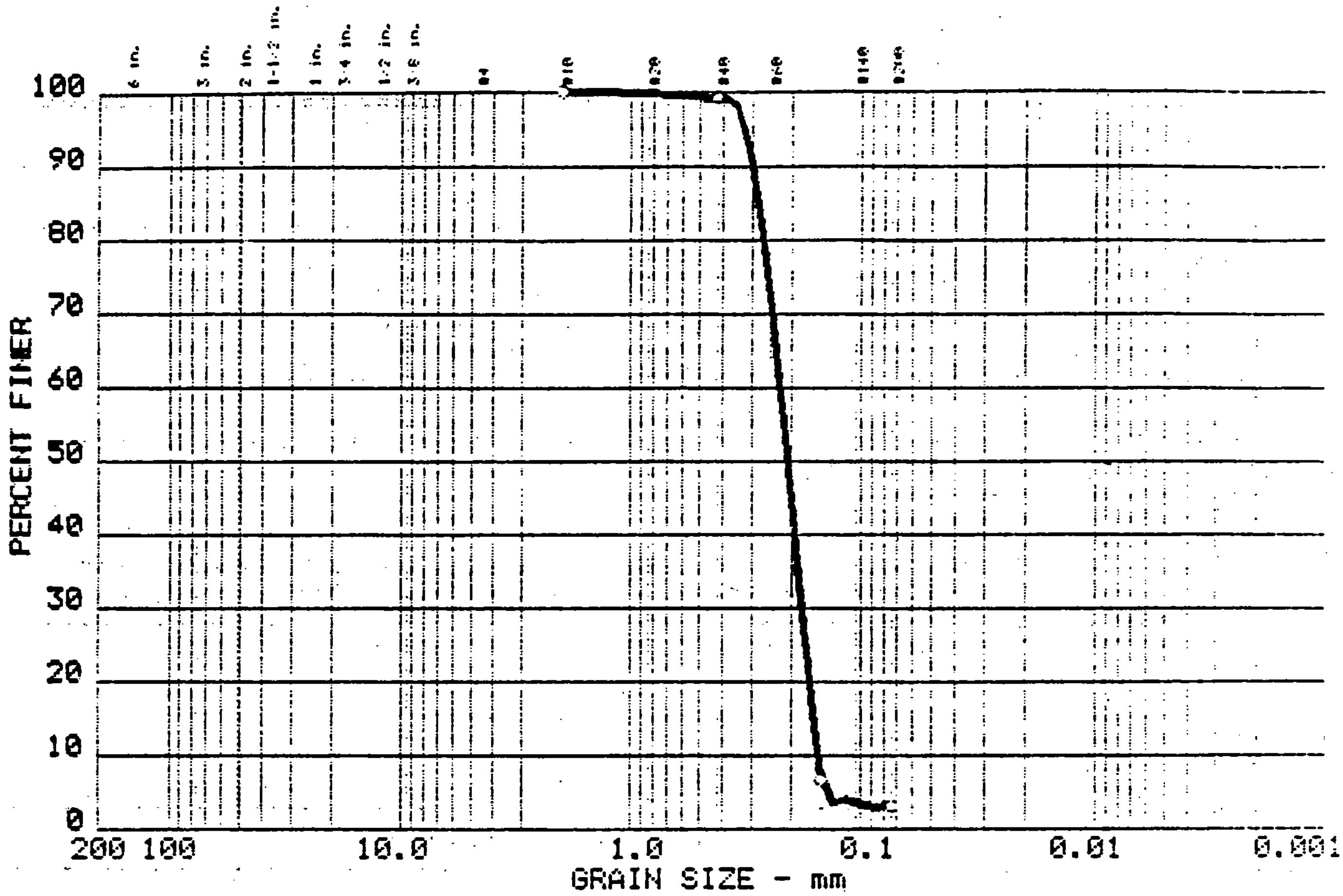
eve	Size, mm	Percent finer
75 inches	19.05	100.0
375 inches	9.53	94.0
4	4.760	85.9
10	2.000	77.9
40	0.420	32.2
	0.149	14.0
	0.074	8.4

Fractional Components

+ 3 in. = 0.0 % GRAVEL = 14.1 % SAND = 77.5
FINES = 8.4

5= 4.22 D60= 1.000 D50= 0.741
0= 0.3846 D15= 0.16218 D10= 0.09226
= 1.6032 Cu = 10.8393

GRAIN SIZE DISTRIBUTION TEST REPORT



	Test	%+75mm	% GRAVEL	% SAND	% SILT	% CLAY
○	9	0.0	0.0	96.8	3.2	

	LL	PI	D_{85}	D_{60}	D_{50}	D_{30}	D_{15}	D_{10}	C_c	C_u
○			0.28	0.22	0.21	0.180	0.1607	0.1542	0.93	1.5

MATERIAL DESCRIPTION	USCS	AASHTO
○ SAND, FINE GRAINED	SP	

Project No.: 4220 91-1181
Project: SPENT FUEL ROD VAULT
○ Location: PRAIRIE ISLAND NUCLEAR PLANT

Date: 06/21/91

Remarks:
BORING NO.: 14
SAMPLE NO.: 5
DEPTH (ft): 29.5 - 31

GRAIN SIZE DISTRIBUTION TEST REPORT
TWIN CITY TESTING CORPORATION

Figure No.

GRAIN SIZE DISTRIBUTION TEST DATA   Test No.: 9

te: 06/21/91
ct No.: 4220 91-1181
. .ct.: SPENT FUEL ROD VAULT

Sample Data

cation of Sample: PRAIRIE ISLAND NUCLEAR PLANT
mple Description: SAND, FINE GRAINED
CS Class: SP   Liquid limit:
SHTO Class: Plasticity index:

Notes

marks: BORING NO.: 14 SAMPLE NO.: 5
DEPTH (ft): 29.5 - 31
g. No.:

Mechanical Analysis Data

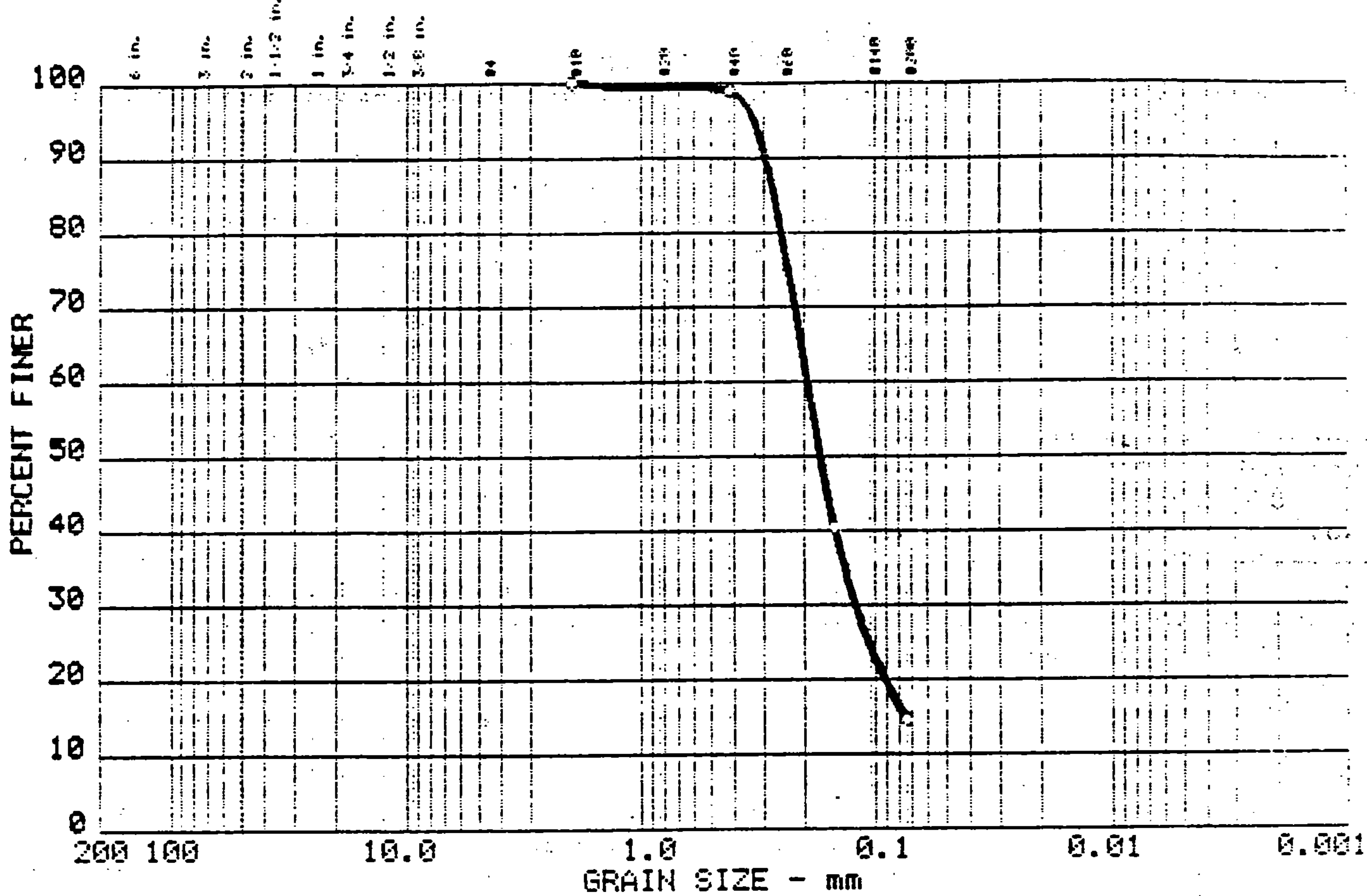
eve	Size, mm	Percent finer
10	2.000	100.0
40	0.420	99.3
100	0.149	6.7
200	0.074	3.2

Fractional Components

+ 3 in. = 0.0   % GRAVEL = 0.0   % SAND = 96.8
FINES = 3.2

5= 0.28 D60= 0.224 D50= 0.208
0= 0.1797 D15= 0.16069 D10= 0.15417
= 0.9343 Cu = 1.4538

GRAIN SIZE DISTRIBUTION TEST REPORT



	Test	%+75mm	% GRAVEL	% SAND	% SILT	% CLAY
○	10	0.0	0.0	85.3	14.7	

	LL	PI	D_{85}	D_{60}	D_{50}	D_{30}	D_{15}	D_{10}	C_c	C_u
○			0.27	0.19	0.17	0.122	0.0745			

MATERIAL DESCRIPTION	USCS	AASHTO
○ SILTY SAND, FINE GRAINED	SM	

Project No.: 4220 91-1181
Project: SPENT FUEL ROD VAULT
○ Location: PRAIRIE ISLAND NUCLEAR PLANT

Date: 06/21/91

Remarks:
BORING NO.: 15
SAMPLE NO.: 4
DEPTH (ft): 4.5 - 6

GRAIN SIZE DISTRIBUTION TEST REPORT
TWIN CITY TESTING CORPORATION

Figure No.

GRAIN SIZE DISTRIBUTION TEST DATA Test No.: 10

:t·· 06/21/91
r ct No.: 4220 91-1181
ʼoʝect: SPENT FUEL ROD VAULT

Sample Data

ocation of Sample: PRAIRIE ISLAND NUCLEAR PLANT
ımple Description: SILTY SAND, FINE GRAINED
;CS Class: SM Liquid limit:
ASHTO Class: Plasticity index:

Notes

emarks: BORING NO.: 15 SAMPLE NO.: 4
DEPTH (ft): 4.5 - 6
.g. No.:

Mechanical Analysis Data

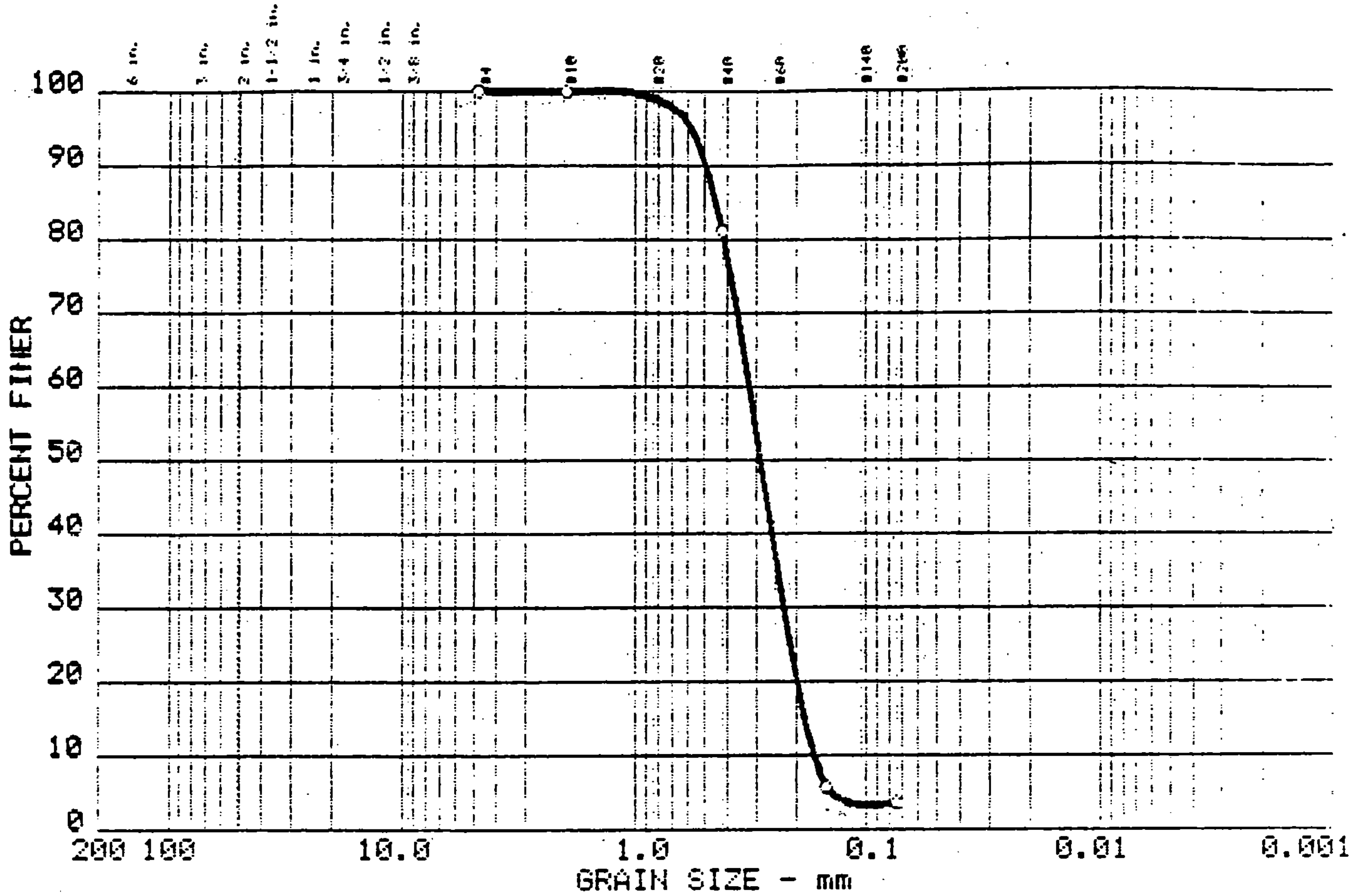
ıeve	Size, mm	Percent finer
10	2.000	100.0
40	0.420	98.9
100	0.149	40.5
200	0.074	14.7

Fractional Components

+ 3 in. = 0.0 % GRAVEL = 0.0 % SAND = 85.3
FINES = 14.7

85= 0.27 D60= 0.194 D50= 0.171
30= 0.1222 D15= 0.07447

GRAIN SIZE DISTRIBUTION TEST REPORT



	Test	%+75mm	% GRAVEL	% SAND	% SILT	% CLAY
○	11	0.0	0.0	96.4	3.6	

	LL	PI	D_{85}	D_{60}	D_{50}	D_{30}	D_{15}	D_{10}	C_c	C_u
○			0.45	0.32	0.29	0.227	0.1843	0.1681	0.96	1.9

MATERIAL DESCRIPTION	USCS	AASHTO
○ SAND, FINE GRAINED	SP	

Project No.: 4220 91-1181
Project: SPENT FUEL ROD VAULT
○ Location: PRAIRIE ISLAND NUCLEAR PLANT

Date: 06/21/91

GRAIN SIZE DISTRIBUTION TEST REPORT
TWIN CITY TESTING CORPORATION

Remarks:
BORING NO.: 15
SAMPLE NO.: 5
DEPTH (ft): 14.5 - 16

Figure No.

GRAIN SIZE DISTRIBUTION TEST DATA Test No.: 11

ate: 06/21/91
r ct No.: 4220 91-1181
Pr ct: SPENT FUEL ROD VAULT

Sample Data

ocation of Sample: PRAIRIE ISLAND NUCLEAR PLANT
ample Description: SAND, FINE GRAINED
CS Class: SP Liquid limit:
SHTO Class: Plasticity index:

Notes

emarks: BORING NO.: 15 SAMPLE NO.: 5
DEPTH (ft): 14.5 - 16
g. No.:

Mechanical Analysis Data

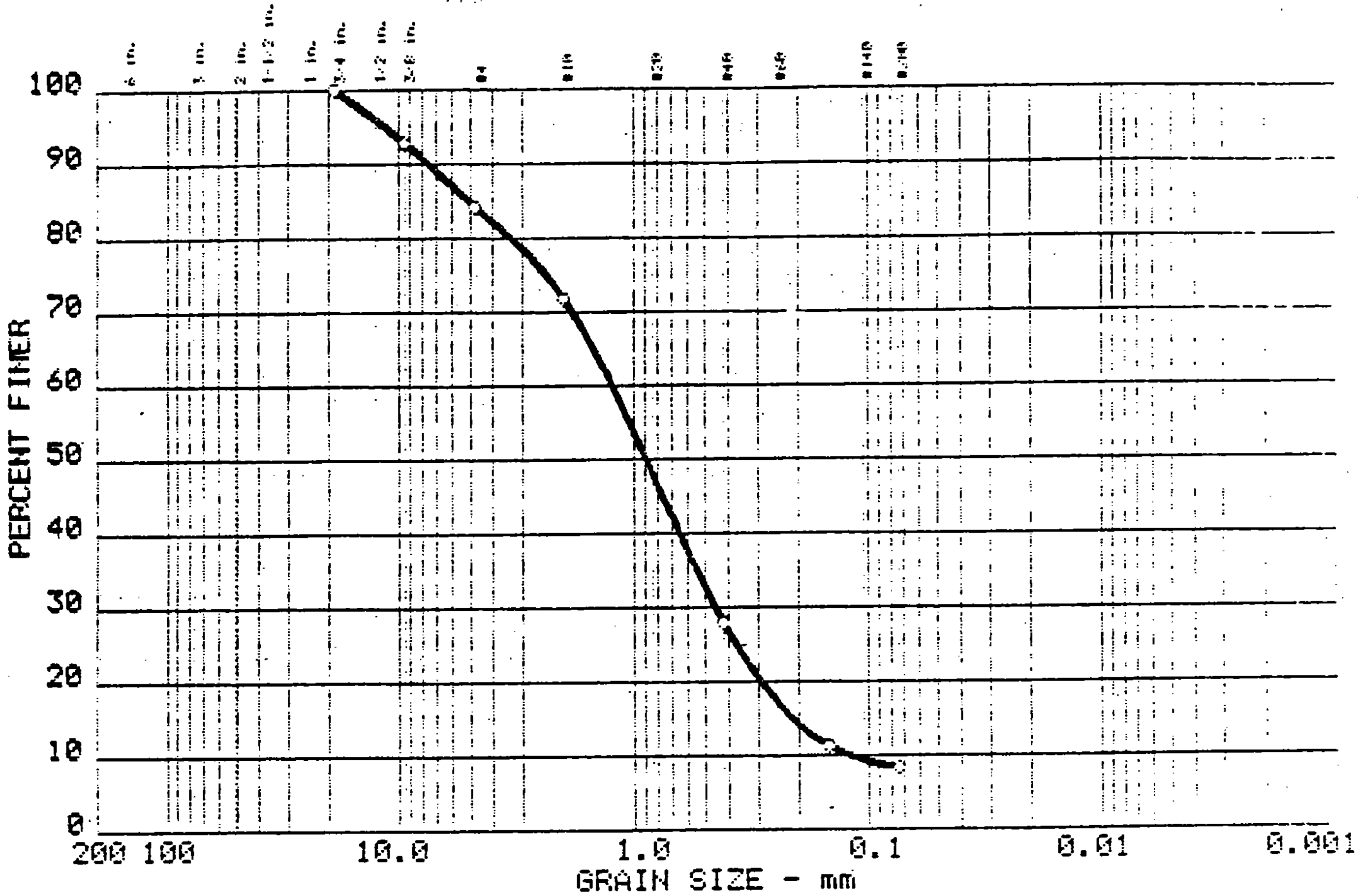
eve	Size, mm	Percent finer
4	4.760	100.0
10	2.000	99.8
40	0.420	81.3
100	0.149	5.7
200	0.074	3.6

Fractional Components

+ 3 in. = 0.0 % GRAVEL = 0.0 % SAND = 96.4
FINES = 3.6

85= 0.45 D60= 0.319 D50= 0.285
.0= 0.2267 D15= 0.18429 D10= 0.16807
: = 0.9583 Cu = 1.8989

GRAIN SIZE DISTRIBUTION TEST REPORT



Test	%+75mm	% GRAVEL	% SAND	% SILT	% CLAY
○ 12	0.0	15.8	75.9	8.3	

	LL	PI	D_{85}	D_{60}	D_{50}	D_{30}	D_{15}	D_{10}	C_c	C_u
○			5.07	1.24	0.89	0.456	0.2106	0.1171	1.42	10.6

MATERIAL DESCRIPTION	USCS	AASHTO
○ SAND W/SILT AND W/GRAVEL, MEDIUM GRAINED	SP-SM	

Project No.: 4220 91-1181
Project: SPENT FUEL ROD VAULT
○ Location: PRAIRIE ISLAND NUCLEAR PLANT

Date: 06/21/91

Remarks:
BORING NO.: 15
SAMPLE NO.: 9
DEPTH (ft): 49.5 - 51

GRAIN SIZE DISTRIBUTION TEST REPORT
TWIN CITY TESTING CORPORATION

Figure No.

GRAIN SIZE DISTRIBUTION TEST DATA	Test No.: 12

ıte: 06/21/91
r ct No.: 4220 91-1181
roject: SPENT FUEL ROD VAULT

Sample Data

ocation of Sample: PRAIRIE ISLAND NUCLEAR PLANT
ample Description: SAND W/SILT AND W/GRAVEL, MEDIUM GRAINED
SCS Class: SP-SM Liquid limit:
ASHTO Class: Plasticity index:

Notes

emarks: BORING NO.: 15 SAMPLE NO.: 9
DEPTH (ft): 49.5 - 51
.g. No.:

Mechanical Analysis Data

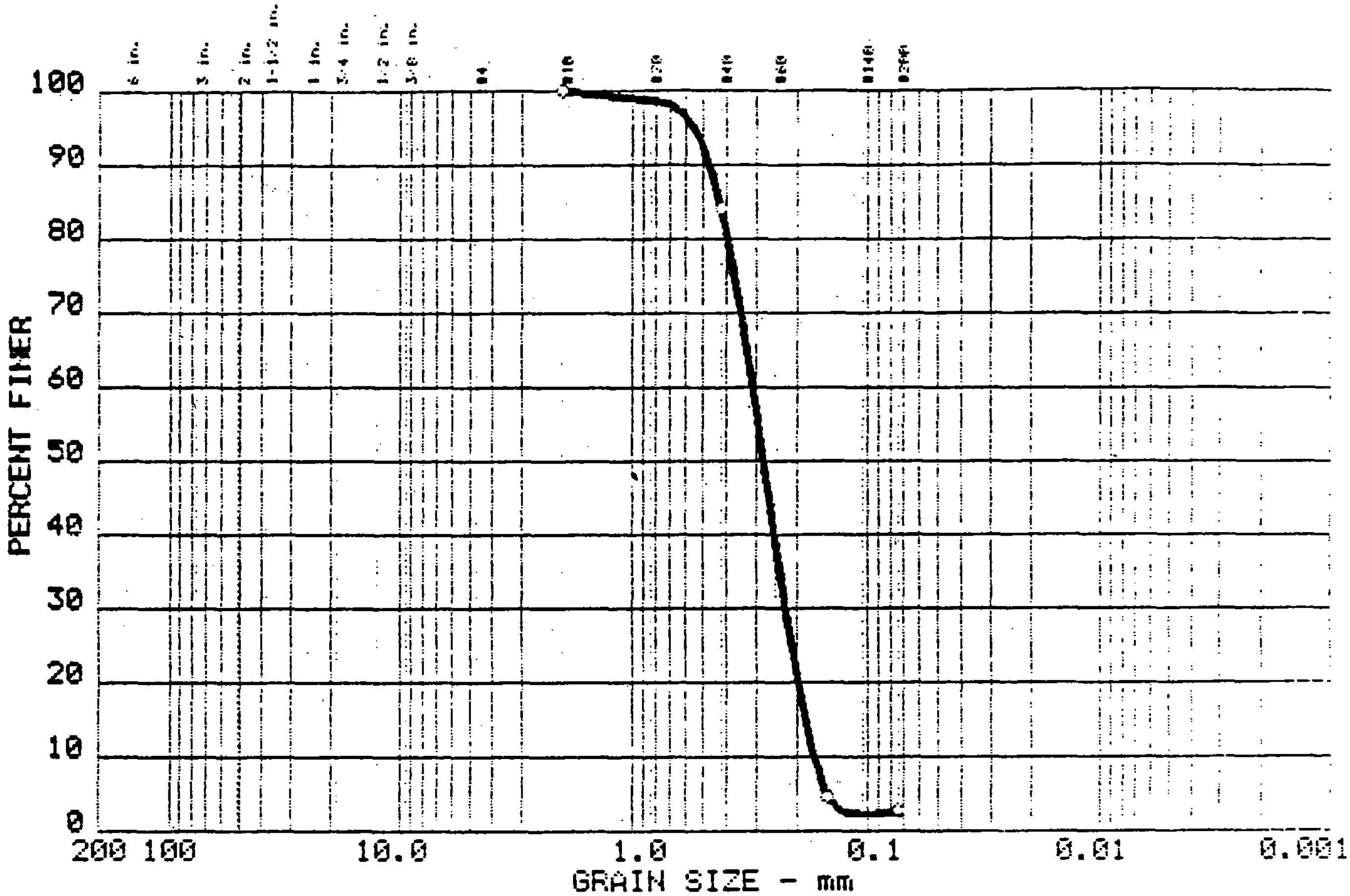
Ieve	Size, mm	Percent finer
.75 inches	19.05	100.0
375 inches	9.53	92.8
4	4.760	84.2
10	2.000	71.8
40	0.420	27.9
	0.149	11.3
[illegible]	0.074	8.3

Fractional Components

+ 3 in. = 0.0 % GRAVEL = 15.8 % SAND = 75.9
FINES = 8.3

85= 5.07 D60= 1.245 D50= 0.891
30= 0.4555 D15= 0.21062 D10= 0.11708
: = 1.4240 Cu = 10.6292

GRAIN SIZE DISTRIBUTION TEST REPORT



	Test	%+75mm	% GRAVEL	% SAND	% SILT	% CLAY
○	13	0.0	0.0	97.3	2.7	

	LL	PI	D_{85}	D_{60}	D_{50}	D_{30}	D_{15}	D_{10}	C_c	C_u
○			0.43	0.31	0.28	0.224	0.1851	0.1704	0.95	1.8

MATERIAL DESCRIPTION	USCS	AASHTO
○ SAND, FINE GRAINED	SP	

Project No.: 4220 91-1181
Project: SPENT FUEL ROD VAULT
○ Location: PRAIRIE ISLAND NUCLEAR PLANT

Date: 06/21/91

GRAIN SIZE DISTRIBUTION TEST REPORT
TWIN CITY TESTING CORPORATION

Remarks:
BORING NO.: 16
SAMPLE NO.: 9
DEPTH (ft): 34.5 - 36

Figure No.

GRAIN SIZE DISTRIBUTION TEST DATA	Test No.: 13

ate: 06/21/91
r ct No.: 4220 91-1181
r ct: SPENT FUEL ROD VAULT

Sample Data

ocation of Sample: PRAIRIE ISLAND NUCLEAR PLANT
ample Description: SAND, FINE GRAINED
SCS Class: SP Liquid limit:
ASHTO Class: Plasticity index:

Notes

emarks: BORING NO.: 16 SAMPLE NO.: 9
DEPTH (ft): 34.5 - 36
ig. No.:

Mechanical Analysis Data

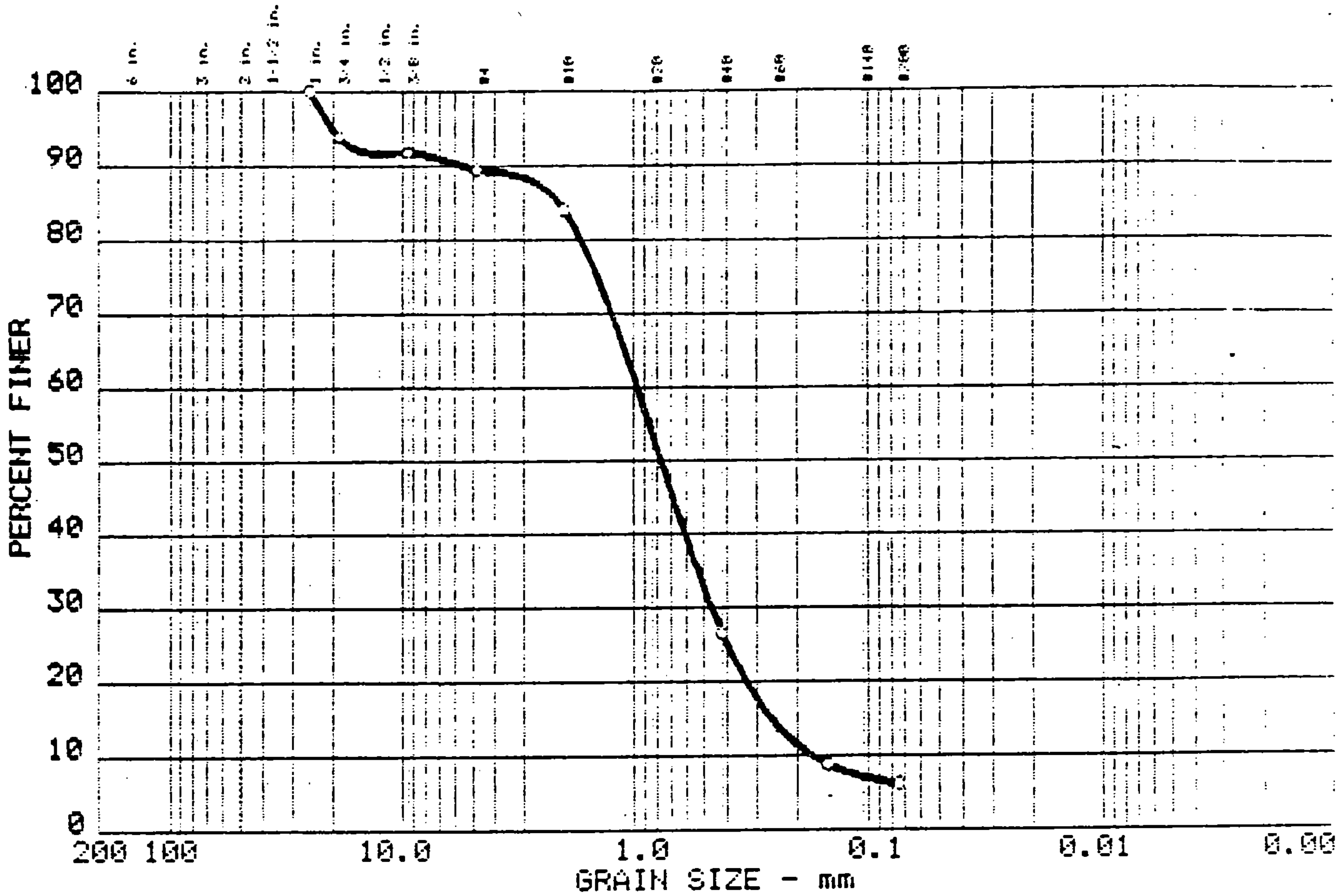
ieve	Size, mm	Percent finer
10	2.000	100.0
40	0.420	84.1
100	0.149	4.6
200	0.074	2.7

Fractional Components

+ 3 in. = 0.0 % GRAVEL = 0.0 % SAND = 97.3
FINES = 2.7

35= 0.43 D60= 0.309 D50= 0.278
30= 0.2241 D15= 0.18514 D10= 0.17041
: = 0.9550 Cu = 1.8113

GRAIN SIZE DISTRIBUTION TEST REPORT



	Test	%+75mm	% GRAVEL	% SAND	% SILT	% CLAY
○	14	0.0	10.5	83.3	6.2	

	LL	PI	D85	D60	D50	D30	D15	D10	Cc	Cu
○			2.11	0.97	0.77	0.466	0.2591	0.1732	1.30	5.6

MATERIAL DESCRIPTION	USCS	AASHTO
○ SAND W/SILT & W/A LITTLE GRAVEL, MEDIUM GRAINED	SP-SM	

Project No.: 4220 91-1181
Project: SPENT FUEL ROD VAULT
○ Location: PRAIRIE ISLAND NUCLEAR PLANT

Date: 06/21/91

GRAIN SIZE DISTRIBUTION TEST REPORT
TWIN CITY TESTING CORPORATION

Remarks:
BORING NO.: 16
SAMPLE NO.: 14
DEPTH (ft): 59.5 - 61

Figure No.

GRAIN SIZE DISTRIBUTION TEST DATA Test No.: 14

_ 06/21/91
r ct No.: 4220 91-1181
oject: SPENT FUEL ROD VAULT

Sample Data

ocation of Sample: PRAIRIE ISLAND NUCLEAR PLANT
.mple Description: SAND W/SILT & W/A LITTLE GRAVEL, MEDIUM GRAINED
CS Class: SP-SM Liquid limit:
ASHTO Class: Plasticity index:

Notes

emarks: BORING NO.: 16 SAMPLE NO.: 14
DEPTH (ft): 59.5 - 61
g. No.:

Mechanical Analysis Data

ieve	Size, mm	Percent finer
inches	25.40	100.0
75 inches	19.05	93.9
.375 inches	9.53	91.8
4	4.760	89.5
	2.000	84.0
	0.420	26.5
100	0.149	8.8
200	0.074	6.2

Fractional Components

+ 3 in. = 0.0 % GRAVEL = 10.5 % SAND = 83.3
FINES = 6.2

85= 2.11 D60= 0.966 D50= 0.766
0= 0.4661 D15= 0.25912 D10= 0.17318
_ = 1.2987 Cu = 5.5783

SECTION 3

PRINCIPAL CASK DESIGN CRITERIA

3.1 PURPOSES OF CASK

3.1.1 SPENT FUEL TO BE STORED

The ISFSI is designed to accommodate a total of 48 storage casks. Each of the casks is capable of accommodating 40 spent fuel assemblies. The total capacity of the fuel to be stored at the facility is 715.29 metric tons of uranium (MTU). This is based on storage of 482 Westinghouse standard assemblies (400 kgU/each), 481 Exxon's standard and TOPROD assemblies (370 kgU/each) and 957 Westinghouse optimized design assemblies (360 kgU/each).

Physical characteristics of the fuel to be stored in the ISFSI are described in detail in Chapter 3 of the Prairie Island USAR (Reference 1) and are summarized in Table 3.1-1.

The following fuel assembly characteristics constitute limiting parameters for storage of specific assemblies at the ISFSI:

- Initial Fuel Enrichment: shall be limited to a maximum of 3.85 weight percent U-235

- Fuel Burnup: shall be limited to a maximum burnup of 45,000 MWD/MTU

- Decay Time: shall be limited to a minimum of 10 years

- Physical Configuration/Condition: fuel assembly shall be intact, shall have no known cladding defects and shall not have physical damage which would inhibit insertion or removal from the cask fuel basket.

Only spent fuel which meets the requirements specified in Chapter 10 and the ISFSI Technical Specifications will be stored in the ISFSI.

The thermal and radiological characteristics for the spent fuel were generated using the ORIGEN2 computer code (Reference 2). These characteristics for the Westinghouse OFA 14x14 assembly are shown in Table 3.1-2. For the thermal and radiological characteristics, the Westinghouse 14x14 OFA assembly with an enrichment of 3.85 w/o U-235 was chosen. The OFA assembly contains the largest "structural" mass of all the assembly types. It was assumed to contain 380 kgU, which is the average of all assemblies. Analyses are included in Section 3.3.4.

Fuel with various combinations of burnup, specific power, enrichment and cooling time can be stored in the TN-40 cask as long as values for decay heat and gamma and neutron sources, including spectra, fall within the design limits specified in Table 3.1-2. For reference, Figure 3.1-1 shows the relationship between cooling time and decay heat for the design basis fuel assembly. The decay heat values were calculated by multiplying the ORIGEN2 values by 1.08 (Reference 3). Figures 3.1-2 and 3.1-3 show the total photon and neutron sources, respectively, as a function of cooling time for the design basis 14x14 PWR assembly. Figure 3.1-4 shows the dimensional data for the fuel assembly.

3.1.2 GENERAL OPERATING FUNCTIONS

The fuel assemblies will be stored unconsolidated and dry in sealed storage casks. The casks will rest on a reinforced concrete pad, and provide safe storage by ensuring a reliable decay heat path from the spent fuel to the environment and by providing appropriate shielding and containment of the fission product inventory. Storage of spent fuel in storage casks is a totally passive function, with no active systems required to function. Cooling of the casks is accomplished by radiant and convective cooling.

Each cask will be handled with a lifting yoke, the 125 ton capacity auxiliary building crane, a transport vehicle, or other appropriate equipment. The crane will lift the cask from the spent fuel pool, in the spent fuel pool enclosure, move the cask laterally through an access door, and lower the cask to ground level in the rail bay of the Auxiliary Building. The cask will then be picked up by the transport vehicle which will be pulled to the ISFSI by a tow vehicle. After the transport vehicle has been maneuvered to locate the cask in its storage position, the cask will be set down.

All the handling equipment to be used outside the Auxiliary Building will be designed according to appropriate commercial codes and standards, and will be operated, maintained, and inspected in accordance with the supplier's recommendations. Documentation will be maintained to substantiate conformance with all applicable standards.

3.2 DESIGN CRITERIA FOR ENVIRONMENTAL CONDITIONS AND NATURAL PHENOMENA

The storage casks are designed with the objectives of ensuring that fuel criticality is prevented, cask integrity is maintained, and fuel is not damaged so as to preclude its ultimate removal from the cask. The conditions under which these objectives must be met are described below.

The safe storage of the spent fuel assemblies must depend only on the capability of the storage casks to fulfill their design functions. The casks are self-contained, independent, passive systems, which do not rely on any other systems or components for their operation. Therefore, the casks are safety-related. The criteria used in the design of the casks ensure that their exposure to credible site hazards will not impair their safety functions. Because the ISFSI is located on the plant site, all ISFSI design criteria for environmental conditions and natural phenomena are the same as the plant design basis, as found in the USAR.

The design criteria satisfy the requirements of 10CFR72. They consider the effects of normal operation, natural phenomena and postulated man-made accidents. The criteria are defined in terms of loading conditions imposed on the storage cask. The loading conditions are evaluated to determine the type and magnitude of loads induced on the storage cask. The combinations of these loads are then established based on the number of conditions that can superimpose. The load combinations are then classified as Design and/or Service Conditions consistent with Section III of the ASME Boiler and Pressure Vessel Code (Reference 4). The stresses resulting from the application of these loads are then typically evaluated based on the rules for a Class 1 nuclear component in Subsection NB of the Code.

3.2.1 TORNADO AND WIND LOADINGS

Tornado loadings specified in Section 12.2.1.3.2 of the Prairie Island USAR were used in the design of the TN-40 cask. These loads consist of the following:

- A differential pressure equal to 3 psi. This pressure is assumed to build up from normal atmospheric pressure in 3 seconds.
- A lateral force caused by a funnel of wind having a peripheral tangential velocity of 300 mph and a forward progression of 60 mph.
- The design tornado driven missile was assumed equivalent to an airborne 4″ x 12″ x 12′0″ plank travelling end-on 300 mph, or a 4000 lb. automobile flying through the air at 50 mph and at not more than 25 ft. above ground level.

An analysis of impact on the cask of tornado missiles in accordance with Table 12.2-9 of the Prairie Island USAR is presented below. Wind loading is not significant in comparison to that due to tornados; therefore, the wind loading is conservatively taken to be the same as the tornado loading.

3.2.1.1 APPLICABLE DESIGN PARAMETERS

The external pressure drop of 3 psi associated with passing of the tornado is small and, when combined with the other internal pressure loads, is far exceeded by the design internal pressure (20 psig normal design rating, 100 psig analyzed) for the cask. Its effect is included in the effects of the other internal pressure loadings listed in Table 3.2-3.

3.2.1.2 DETERMINATION OF FORCES ON STRUCTURES

The 360 mph tornado wind was assumed based upon a summation of the assumed peripheral tangential velocity and forward progression of the design basis tornado. The wind loading was converted to a dynamic pressure (psf) acting on the cask by multiplying the square of the wind velocity (in mph) by a coefficient (0.002558 at ambient sea level condition) dependent on the air density. The result is a pressure of 332 psf. This is based on data presented in a paper by T.W. Singell (Reference 5). The net force acting on the cask is obtained by multiplying this pressure by the product of the area of the cask projected onto a plane normal to the direction of wind times a drag coefficient. A drag coefficient of 1 is used based on the geometric proportions of the cask (i.e. length to diameter ratio of approximately 2) and the conservative assumption that the cask surface is rough.

This results in a distributed load, w lb/in, acting on the cask in a vertical orientation over the length of 201.0 in. as shown in Figure 3.2-1. The load is calculated as follows:

$$\begin{aligned} w &= \frac{332}{144} \times \text{Outer Shell Diameter} \\ &= \frac{332}{144} \times 101.0 \\ &= 232.9 \text{ lb/in} \end{aligned}$$

An additional type of load on the structure is that created by the impact of tornado missiles on the cask. These impacts are analyzed for two types of missiles specified in the Prairie Island USAR. Missile A is a 4000 lb. automobile striking the cask horizontally at normal incidence at 50 mph. Missile B is a 4 in. x 12 in. x 144 in. plank weighing 200 lb. striking the cask:

- Horizontally at normal incidence at 300 mph
- Vertically at normal incidence at 80% of the horizontal velocity.

3.2.1.2.1 STABILITY OF THE CASK IN THE VERTICAL POSITION UNDER WIND LOADING

The cask stands in an upright position on a concrete pad. The coefficient of friction between the steel cask and the concrete is taken as 0.25 for dry concrete. This coefficient is conservatively low based on data in Marks Handbook (Reference 6) which gives a value of 0.29 for steel on sandstone. Steel on concrete would be similar. A wind velocity of 407 mph would then be required to cause the cask to slide.

This value is calculated below:

$$q = 0.002558\, V^2$$

$$q = \frac{F}{A}$$

F = force to overcome friction force, F_f

A = projected area of cask length, ft^2

$F = F_f$ = cask weight × 0.25

$= 240{,}000 \times 0.25$ *(The weight used in this evaluation is slightly low. This is conservative for stability evaluation under wind loading.)*

$= 60{,}000$ lb.

$A = 201.0 \times 101.0/144$

$= 141\ ft^2$

Therefore: $0.002558\,V^2 = \frac{60{,}000}{141}; V = 407\ mph$

If the cask does not slide, a constant wind velocity of 549 mph would be required to tip the TN-40 cask, with an outer diameter of 91.0 in. This is calculated as follows:

M_c = The tipping moment about the bottom edge of the cask

$$M_c = \frac{91.0}{2} W - w \times \frac{(201.0)^2}{2}$$

W = cask weight = 240,000 lb

w = distributed load to tip cask, lb/in

Therefore

$$w = 91.0W/2 \times 2/(201.0)^2 = 541\ lb/in.$$

The corresponding pressure load, q, is:

$$q = \frac{541}{101.0} \times 144 = 771 psf$$

The corresponding wind speed (v) is:

$$v = \sqrt{\frac{771}{0.002558}}$$

$$= 549 \ mph$$

3.2.1.2.2 STABILITY OF THE CASK IN THE VERTICAL POSITION UNDER MISSILE IMPACT

It is assumed that both Missile A and Missile B impact inelastically on the cask as shown in Figure 3.2-2. Both missile A (the automobile) and missile B (the wood plank) are assumed to crush. The cask will tend to slide if the missile strikes it below the center of gravity (unless it is blocked in position) or tilt if the missile strikes it above the center of gravity. Conservation of momentum is assumed for both sliding and tipping with a coefficient of restitution of zero. The energy transferred to the cask is dissipated by friction in the sliding case or transformed into potential energy as the cask center of gravity lifts in the tipping case. When a missile strikes the side of the cask at an elevation near the center of gravity, the following velocities are calculated:

In the sliding case:

$$V = \frac{mv_0}{M + m}$$

In the tipping case:

$$\omega_p = \frac{md_{CG} v_0}{m(d_{CG})^2 + I_p}$$

Where:

V = cask translational velocity after impact
V_o = missile initial velocity
ω_p = cask angular velocity about P after impact
m = mass of Missile A
M = cask mass
d_{CG} = distance from center of gravity to pivot point P
I_p = moment of inertia of cask about pivot point P

When the appropriate substitutions are performed for Missile A (automobile) impact, the cask velocity after impact, V, in the sliding case is found to be 1.20 ft/sec. The rotational velocity about P, ω_p, is found to be 0.21 rad/sec for impact at the top. Missile B impact produces lower cask translational and rotational velocities because of its lower initial momentum. Vertical impact of Missile B has no effect on cask stability.

If the cask slides on the concrete pad, the cask kinetic energy after impact is absorbed by friction. The friction work can be equated to the kinetic energy. Assuming a coefficient of friction of 0.25:

$$E_{friction} = \mu \; g(M + m)\, l$$
$$= 1/2\, (M + m)\, v^2$$

Where

$E_{friction}$ = *energy absorbed in friction*
μ = *coefficient of friction*
g = *acceleration of gravity*
l = *distance cask slides on concrete pad*

When a solution is performed to determine l, the sliding distance is determined to be 1.1 in. When the cask tips or pivots about point P after impact, the kinetic energy is transformed into potential energy as the center of gravity rises:

$$E_{tipping} = \text{Increase in Potential Energy} = \text{Kinetic Energy}$$

$$E_{tipping} = Mg\, d_{CG}\, [\cos(\beta+\alpha- \pi/2\,)-\cos\beta] = 1/2\;\, I_p\, \omega_p{}^2$$

Where:

$E_{tipping}$ = *The increase in potential energy of the cask since the center of gravity rises as the cask pivots about the corner.*

β, α are indicated in Figure 3.2-2.

The angle α is determined to be 89° for impact at the top of the cask (the cask tilts 1° and the center of gravity lifts about 0.8 in.). The cask is stable and will not tip over since it will return to the vertical position as long as α is greater than about 63.7°.

Even at this 89° angle, wind will not tip over the cask (if the wind force occurs simultaneously with Missile A impact). At an 89° angle the tipping moment about Point P due to the 360 mph wind is less than 45% of the restoring moment due to the weight. Therefore the cask is stable in the vertical orientation under simultaneous tornado wind and tornado missile loadings.

The impact forces applied to the cask as it is struck by the missiles are determined as follows:

- Missile A (automobile) is assumed to crush 1.5 ft. (18 in.) under a constant force during the impact. The loss of kinetic energy is assumed to be dissipated by crushing of the missile.

$$F_a \times 1.5\,ft. = \frac{1}{2}\left[m_a v_0^{\,2} - (M + m_a)V^2\right]$$

The frontal area of the automobile is assumed to be

$$3\,ft.\ x\ 6\,ft.\ = 18\,ft.^2$$

$$p_a = F_a/18\,ft.^2$$

where:

F_a = *Impact force on cask by Missile A*

p_a = *Impact pressure on cask by Missile A*

The impact force, F_a, is determined to be 222,682 lb., and the crush pressure on the frontal area of the automobile, p_a, is 85.9 psi.

- Missile B (Wood Plank) will crush under impact. The highest crush strength S_{crush} of any wood listed in Reference 6 is 9210 psi (hickory). The contact force required for the plank to start crushing is:

$$F_c = S_{crush} \times A = 9210 \times 48 = 442{,}080 \quad lb.$$

The kinetic energy of the plank is dissipated as the plank crushes against the cask wall. The kinetic energy (KE) of the plank is:

$$KE = \frac{1}{2}Mv_0^{\,2} = \frac{12}{2} \times \frac{200}{32.2} \times \left(\frac{300 \times 5280}{3600}\right)^2 = 7.215 \times 10^6\,in.lb.$$

The crush length (L) of the plank, for a constant force, is evaluated as:

$$L = \frac{KE}{F_c} = \frac{7.215 \times 10^6}{442{,}080} = 16.3 \ \ in.$$

For the case of the plank striking the top of the cask, the velocity is 80% of the horizontal case. Hence, the kinetic energy would be 4.618×10^6 in.lb. and the crush length of the plank 10.4 in.

3.2.1.3 LOCAL EFFECT ON CONTAINMENT OF MISSILE IMPACT

Both Missiles A and B deform and are crushed during the impact. The local pressure on the cask structure is a small fraction of the body yield strength. Therefore, no local penetration occurs.

3.2.2 WATER LEVEL (FLOOD) DESIGN

The design basis flood used for the ISFSI is the same as that used for Class 1 (safe shutdown) structures of the Prairie Island Nuclear Generating Plant, and is described in Section 2.4.1. The probable maximum flood level calculated to occur at the ISFSI is 706.7 ft. above msl with a water velocity of 6.2 ft/sec. This includes wave run-up. The ISFSI is sited and designed such that the lowest point of potential leakage into the cask is above the level of the probable maximum flood.

The casks are designed to withstand loads from forces developed by the probable maximum flood including hydrostatic effects and dynamic phenomena such as momentum and drag.

The storage cask is designed for an external pressure of 25 psi which would be equivalent to a static head of water of approximately 56 ft. This is greater than would be anticipated due to the probable maximum flood.

The drag force (F) exerted on the cask by the flowing water is:

$$F = C_D A \rho \frac{V^2}{2g}$$

where: F = *Drag force*

C_D = *Drag coefficient = 1.0*

A = *Projected area = 141* ft^2

ρ = *density of water = 62.4* lb/ft^3

V = *water velocity = 6.2 ft/sec*

Therefore $F = 5{,}250$ *lb.*

The drag force is applied to the cask in the same manner as the wind force discussed in Section 3.2.1. Since the wind force did not move the cask, and the drag force is less than 20% of the wind force, the postulated flood will not cause the cask to slide or tip.

As mentioned previously, the maximum flood height including wave run-up does not exceed the height of the cask seals. Therefore no inleakage of water can occur. Also, the interspace between the containment seals and the containment vessel cavity are pressurized to approximately 6 atm and 2 atm, respectively, to further preclude any possibility of water inleakage.

3.2.3 SEISMIC DESIGN

Seismic design criteria requirements are set forth in 10CFR72.102. The design earthquake for use in the design of the casks and ISFSI structures must be equivalent to the safe shutdown earthquake (SSE) for a collocated nuclear power plant, the site of which has been evaluated under the criteria of 10CFR100, Appendix A.

Section 2.6 of the Prairie Island USAR discusses site seismology and development of the SSE. An SSE of 0.12g horizontal and 0.08g vertical has been established as the design criteria for the Prairie Island Nuclear Generating Plant. The recommended response spectra developed for the SSE as set forth in the Prairie Island USAR, Appendix E and Figure 2.5-8 of this SAR, was used in cask design.

3.2.3.1 SEISMIC ACCELERATION LEVEL

The TN-40 storage cask is a stiff structure having high natural frequencies of vibration. The cask standing vertically on its pad has a frequency above the minimum required (33 cps) to treat a structure as a rigid body and to ignore amplification of free field seismic motion. Therefore the cask is evaluated using an equivalent static seismic loading method, and there is no need to specify a design response spectrum or its associated time history. The equivalent static load used is 1.5 times (per NUREG-0800, Reference 27) the basic g level specified earlier. These loads are assumed to be distributed over the cask length and cask bottom areas as shown in Figure 3.2-1.

3.2.3.2 SEISMIC-SYSTEM ANALYSIS

The only significant effects of seismic loading that might occur would be sliding and/or tipping (overturning) of the cask.

For a circular cask the horizontal g value necessary to tip the cask is calculated below:

$$M_{tip} = g \times W \times l_v + \frac{2}{3} g \times W \times l_r$$

Where:

M_{tip} = *Moment necessary to tip the cask, in lb.*

g = *Acceleration value necessary to tip the cask*

W = *Weight of cask on pad = 240,000 lb.*

l_v = vertical distance to center of gravity = 92 in.

l_r = radial distance to center of gravity from pivot point = 45.5 in.

$M_{stab} = W \times l_r$

Where:

M_{stab} = stabilizing moment on cask, in lb.

W = 240,000 lb.

l_r = 45.5 in.

Therefore the g value necessary to tip the cask is found by equating M_{tip} to M_{stab}:

$$(g \times W \times l_r) + \left(\frac{2}{3} g \times W \times l_r\right) = W l_r$$

$$g = \frac{45.5}{92 + 0.66 \times 45.5} = 0.37$$

The two horizontal components of seismic load are combined as indicated in Section 2.1 of Reference 7. At 45° to either horizontal component the response due to a N-S earthquake is sin 45°x N-S response and likewise for an E-W earthquake is sin 45°x E-W response. If both components are equal, the combined response is:

$(\sin^2 45° + \sin^2 45°)^{1/2} \times$ *response = response in either axis.*

For this evaluation, the horizontal response is 0.12gx1.5 = 0.18g and the vertical is 0.08x1.5 = 0.12g.

Since the applied horizontal acceleration of 0.18g (0.12x1.5) is less than the 0.37 g required to tip the cask, the cask will not tip over. A dynamic analysis of the potential for a cask tip has also been performed, taking into account the design of the pad. This analysis is described in Section 4.2.1.

If the cask were to slide due to seismic loading, the horizontal component of the seismic load must overcome the normal force acting at the cask/ground interface multiplied by the coefficient of friction. The vertical seismic force is applied upward (0.08Wx1.5 = 0.12W) so as to decrease the normal forces and hence the sliding resistance force. When this is done the downward load is 0.88W and the friction force that must be overcome to slide the cask is 0.22W (0.25x0.88W). The maximum side load is 0.12Wx1.5 = 0.18W; therefore, the cask will not slide.

3.2.4 SNOW AND ICE LOADINGS

The decay heat of the contained fuel will maintain the storage cask outer surface temperature above 32°F throughout its service life, including the end of life, with an ambient temperature of -20°F. Therefore, snow or ice will melt when it comes in contact with the cask so that snow and ice loadings need not be considered for the storage cask.

The temperature of the protective cover attached to the top of the cask above the lid could fall below 32°F under certain conditions and a layer of snow or ice might build up. A 50 psf (0.35 psi) snow or ice load as specified in the Prairie Island USAR corresponds to approximately 6 ft of snow or 1 ft of ice. However, this load is insignificant to the TN-40 since the cover is a 0.38 in. thick toruspherical steel head which can withstand an external pressure over 20 psi. Therefore, the cover will maintain its intended protective function under these snow or ice loading conditions.

Another possible influence on the TN-40 is a thermal shock when the warm cask is suddenly cooled by cold rain (conservatively assumed at 32°F). A number of such cycles is considered in the thermal fatigue analysis in Section 4.

3.2.5 COMBINED LOAD CRITERIA

3.2.5.1 INTRODUCTION

Sections 3.2.1 through 3.2.4, above, describe the most severe natural phenomena considered in the design of the TN-40. These natural phenomena have been analyzed in those SAR sections where it has been shown that the cask is stable. It will not tip over under any condition or slide on its pad more than about an inch. In addition, the forces and pressures applied to the cask due to these phenomena have been determined.

It should be noted that all of the above phenomena are upper bound, low probability events. In most cases, however, there is a more regular and frequent similar phenomena of lower magnitude. For instance, some small wind load occurs often, but a tornado is unlikely. The forces and pressures determined for the severe phenomena can therefore be conservatively used as upper bound values for all of the similar events.

It has been conservatively assumed that these bounding forces and pressures, with a single exception, can occur at any time and their effects are combined with those due to normal operation. The sole exception is the loading(s) due to the tornado missiles as described in Section 3.2.1.2.2. The missile case is evaluated in combination with others as a low probability event which is postulated only because the consequences of cask penetration might result in severe impact on the immediate environs.

3.2.5.2 TN-40 CASK LOADING

A brief explanation of the cask loads due to events that will occur or can be expected to occur in the course of normal operation of the ISFSI follows in the next paragraph. Next, the cask loads due to the severe natural phenomena and accidents will be compared with those for similar but less severe normal events. Then loads equal to or higher than the upper bound values selected for design and analysis of the TN-40, defined as Service Loads, will be described. Finally the Service Loads will be separated into two levels and superposition of simultaneous loadings (combined loads) will be discussed

3.2.5.2.1 NORMAL OPERATION

During normal storage on the ISFSI pad the cask is subjected to loading due to its own dead weight and that of its contents (fuel and basket), assembly stresses due to the bolt preload required to seat the double metallic seals and react to the internal pressure, and internal pressure due to initial pressurization and any fuel clad failure resulting in fission gas release. Additional normal loads include wind loading which produces a distributed lateral load on one side of the cask and can also result in slight external pressure drop on other portions of the cask. Also lifting loads are applied to the cask through the trunnions and the cask dead weight is reacted through the trunnions during lifting operations. Finally, an increased external pressure will be applied to all surfaces of the cask during fuel loading when the cask is at the bottom of the spent fuel pool. Snow and ice loads apply local external pressure loading to the top of the cask. The cask will, of course, be subjected to the full range of thermal conditions produced by ambient variations (including insolation) and decay heat.

3.2.5.2.2 LOADINGS DUE TO SEVERE NATURAL PHENOMENA AND ACCIDENTS

The cask is subjected to dead weight loading and assembly stresses due to bolt preload and seal compression under all conditions. If it becomes necessary to unload a recently loaded hot cask, cold water would be pumped into the cask to reduce the temperature before returning the cask to the pool. If proper controls are not maintained an internal pressure corresponding to saturated steam pressure at the cavity wall temperature could occur which would be higher than the normal internal pressure. The tornado wind loading described in Section 3.2.1 could produce higher lateral loading than any normal wind loading or flood water drag force. The external pressure drop due to the tornado wind is also more severe than due to any normal condition. Tornado missile impact described in Section 3.2.1.2.2 could apply a high local loading to the cask unlike any normal condition. External pressure loading of the cask could occur due to flooding (see Section 3.2.2), burial or nearby explosion. The full range of thermal conditions due to ambient variations, decay heat and minor fire in the vicinity of the cask apply.

3.2.5.2.3 THERMAL CONDITIONS

The TN-40 component temperatures and thermal gradients are affected by the following thermal conditions:

- Fuel loading
- Decay heat
- Insolation
- Beginning of life unloading
- Ambient variations
- Lightning
- Minor fire

The thermal conditions which are of concern structurally are the temperature distributions in the cask and the differential thermal expansions of interfacing cask components.

3.2.5.2.4 FUEL LOADING

The cask is loaded in a spent fuel pool under water. The cask is cooled by pool water; therefore, the thermal gradients established during fuel loading will be negligible.

3.2.5.2.5 DECAY HEAT/SOLAR LOAD

After the cask is loaded and removed from the pool, the body temperature will gradually reach steady state conditions. Since the mass of the cask is large, the time to reach equilibrium will be approximately 1 to 2 days. The temperature gradients in the cask body have an insignificant effect on the structural integrity of the body.

Several thermal analysis calculations were made for different ambient conditions. The methods used to obtain these results are discussed in Section 3.3.2.2. The temperature distribution in the cask resulting from the thermal analysis for the off-normal condition was used for the structural analysis.

3.2.5.2.6 BEGINNING OF LIFE UNLOADING

This condition would occur if it were necessary to place the cask back in the pool at the beginning of life after it had been loaded and reached thermal equilibrium. Prior to placing the cask in the pool, the cask and fuel would have to be cooled by circulating water through the cask. Therefore, cold water would contact the hotter cask inside surfaces. This condition has been evaluated and it has been shown that the thermal gradients in the cask body are small and would have an insignificant effect on the cask body and the fuel cladding

3.2.5.2.7 AMBIENT VARIATIONS

Because the cask thermal inertia is large, the cask temperature response to changes in atmospheric conditions will be relatively slow. Ambient temperature variations due to changes in atmospheric conditions i.e., sun, ice, snow, rain and wind will not affect the performance of the cask. Snow or ice will melt as it contacts the cask because the outer surface will be above 32°F for ambient temperatures above -40°F. The cyclical variation of insolation during a day will also create insignificant thermal gradients. One condition which is evaluated structurally is cold rain on a hot cask. The conservative assumption is made that the thin outer skin on the outer surface of the cask is restrained from contracting by the mass of the cask.

The thermal effects due to ambient variations and conditions are discussed in further detail in Section 3.3.2.2.

3.2.5.2.8 LIGHTNING

Lightning will not cause a significant thermal effect. If struck by lightning on the lid, the electrical charge will be conducted by paths provided by the lid bolts to the body.

The lid metallic O-ring seals can withstand temperatures of up to 600°F without loss of sealing capability. It is not anticipated that lightning could result in the seals reaching temperatures above these values.

3.2.5.2.9 FIRE

The only real source of fuel which could cause a fire in the vicinity of the cask is the fuel tank of the tow vehicle which transports the cask to the storage pad. An evaluation was made to determine the thermal response of the cask assuming this minor fire is an engulfing fire. The details of this analysis are given in Section 3.3.2.2.2 It was concluded that the cask will maintain its containment integrity during and after this bounding hypothetical fire accident.

3.2.5.2.10 BURIED CASK

An evaluation was made to determine the increase in cask temperature with time assuming the cask was completely buried by dirt and debris with very low thermal conductivity. The details of this analysis are given in Section 3.3.2.2. The analysis shows that the cask will maintain its containment integrity up to a maximum burial period of 55 hours.

3.2.5.3 BOUNDING LOADS FOR DESIGN AND SERVICE CONDITIONS

3.2.5.3.1 DEAD (WEIGHT) LOADS

The only dead loads (hereafter referred to as weight loads) on the cask are the cask weight including the contents. The calculated weights of the individual components of the cask and the total weights are given in Table 3.2-1. The weight of the cask assembly is reacted as a contact force between cask and storage pad except when the cask is supported (lifted) by the pair of trunnions at the top of the cask during handling prior to fuel loading.

3.2.5.3.2 LIFTING LOADS

The cask is provided with two trunnions at the top spaced 180 degrees apart for non-redundant lifting. The smaller trunnions at the bottom of the cask are for rotation of the cask.

The upper trunnions themselves are considered to be lifting devices and are evaluated for lifting for g levels equivalent to 6 times and 10 times the upper bound weight of the cask. These values are based on ANSI N14.6 (Reference 8), which requires that lifting devices be capable of lifting 6 times and 10 times the cask weight without exceeding the yield and ultimate strengths of the material. The trunnion loads for the ANSI N14.6 analysis are shown in Figure 3.2-3 and listed in Table 3.2-2.

The cask body is conservatively evaluated for a vertical load of 3 g (i.e., 3 times the weight of the cask) which is reacted at the trunnions involved in the handling operation. The factor of 3 provides ample allowance for sudden load application during lifting. The loads used in the cask body analysis during lifting operations are shown in Figure 3.2-4. The weight of the cask used for these analyses is a conservatively assumed maximum loaded weight of 250,000 lb. The attachment loads are calculated in terms of forces and moments acting locally on the cask body.

3.2.5.3.3 INTERNAL PRESSURE

The pressure inside the cavity of the storage cask results from several sources. Initially the cavity is pressurized with helium such that the cavity pressure is 2.2 atm at thermal equilibrium. The purpose of pressurizing the cavity above atmospheric pressure is to prevent in-leakage of air. The initial pressure is determined on the basis that a 1 atm pressure must exist in the cavity on the coldest day at the end of life. Pressure variations due to daily and seasonal changes in ambient temperature conditions will be small due to the large thermal capacity of the cask.

Fuel clad failure results in the release of fission gas which increases cavity pressure. Under normal storage conditions a 10% fission gas release is assumed due to fuel clad failure. This results in an increase in cavity pressure of 3.6 psi.

Another condition when internal pressure could increase is the cooldown prior to unloading. This could occur at the beginning or end of life. Unloading of fuel at the beginning of life would only be necessary due to excessive leakage past the lid seals or a severe accident, e.g. cask drop. The cask cavity wall temperature at the beginning of life is just below 303°F. Therefore, before returning the cask into a pool, cold water would be pumped into the cavity to reduce the temperature. When the water hits the cavity surface, steam might be produced and the resulting pressure inside the cavity could reach the saturated steam pressure of 71 psia (4.82 atm) corresponding to the cavity wall temperature of 303°F.

Table 3.2-3 presents a summary of internal pressures for the conditions identified. A pressure of 100 psig was chosen as the design internal pressure, since this value exceeds that of all conditions producing an internal pressure. In a response to questions from the NRC Staff, NSP provided justification in Reference 33 for the adequacy of the hydrostatic testing performed on the TN-40 casks during fabrication. In a Safety Assessment dated May 11, 1995 (Reference 31), the NRC concluded that the hydrostatic testing performed on the TN-40 casks was adequate and that TN-40 casks need to be tested with a 25 psig hydrostatic pressure on the inner containment vessel only.

3.2.5.3.4 EXTERNAL PRESSURE

There are several conditions which can result in external pressure on the cask. The external pressure due to flood level is less than 7 psi at the bottom of the cask which corresponds to the 14.7 ft. head of water as discussed in Section 3.2.2.

During fuel loading or unloading the cask is at the bottom of the spent fuel pool, nominally 40 ft. deep. This results in an external hydrostatic pressure of approximately 20 psi.

An explosion on a barge in the vicinity of the Prairie Island plant has been shown to produce an overpressure of less than 2.25 psi at the ISFSI location.

An earth pressure loading would occur if the cask were to be buried under dirt. This is similar to a hydrostatic pressure head of water. The density of loose dirt or earth is approximately 100 lb/ft^3 compared to 62.4 lb/ft^3 for that of water. Therefore 36 ft. of earth is equivalent to a 56 ft. head of water and an external pressure of 25 psi.

The various external pressures are also summarized in Table 3.2-3. The cask is designed and evaluated for an external pressure of 25 psi. This value was selected because it exceeds the maximum external pressure which would be anticipated for any of the loading conditions considered above including floodwater discussed in Section 3.2.2 and snow and ice in Section 3.2.4.

3.2.5.3.5 CASK BODY LOADS

Global distributed loads may be applied to the cask by wind (tornado is upper bound case), flood water and seismic excitation. These loads are explained in detail and calculated in Sections 3.2.1 through 3.2.3. Table 3.2-4 lists the numerical values of these forces as calculated in the various sections. Note that bounding loads equal to the weight of the cask (1g load) in each direction (lateral and vertical) applied as inertial loads for stress analysis purposes envelop all of these distributed loads with a great deal of margin. The local loads due to the tornado missile impact loading are unique. The calculated values from Section 3.2.1 are directly used in the cask analysis since there are no other cases to bound.

3.2.5.4 DESIGN AND SERVICE LOADS

The various cask loading conditions are listed in Table 3.2-5. These loading conditions include those described in 10CFR72, which are categorized as normal, man-made and natural phenomena. The applied loads acting on the different cask components due to these loading conditions have been determined and are discussed in the preceding sections and are listed in Tables 3.2-1 through 3.2-4. This section describes the bases which are used to combine the loads for each cask component. The specific stress criteria against which each load combination will be compared are described in Section 4.2.3.

The bounding pressures and loads described above are used in the load combinations. Certain combinations therefore are conservative evaluations of several events (e.g. one load combination conservatively represents stresses due to tornado wind, hurricane wind, normal wind, flood water, etc.). Several loads are always present and are included in all evaluations. These are the assembly stresses due to bolt preload and metallic seal compression. Lifting loads are always reacted by the cask weight (supported on trunnions - not the storage pad). Lifting loads are not combined with those due to extreme natural phenomena since ISFSI operations would be halted during a flood, hurricane, etc. Dead weight loads are reacted at the bottom of the cask by the storage pad for all cases except the lifting cases.

3.2.5.4.1 CASK BODY

The loading conditions for the cask body including the containment vessel are categorized based on the rules of the ASME Boiler and Pressure Vessel Code Section III, Subsection NB, for a Class 1 nuclear component. The ASME code categorizes component loadings into five service loading conditions. They include Design Conditions (same as the Primary Service) and Levels A, B, C and D Service Loadings. The code also provides different stress limits for each of these service loadings.

For each of these service loading conditions there are several applied loads which are acting on the cask. The Design Loads are listed in Table 3.2-6. They include internal and external pressure; lid bolt preload including the effect of the gasket reactions; distributed loads due to weight, wind, and handling; and attachment loads applied by the trunnion to the cask body.

The inertia g loads are quasistatically applied loads which are multiples of the weight of the cask and/or contents. The magnitude of the Design Loads envelop the maximum Level A Service Loads. Thermal effects are excluded, except for their influence on the preload of the lid bolts (if any) because the ASME Code does not consider these as Design (i.e. primary) Loads.

The Level A Service loads are listed in Table 3.2-7 and are basically the same as the Design Loadings except that the thermal effects on the containment vessel are included. The thermal effects consist of secondary (thermal) stresses caused by differential thermal expansion due to temperature differences caused by decay heat, solar insolation, ambient temperature variations and ambient conditions, e.g. ice, snow, wind, sun.

There are no Level B or C Service Loading Conditions. All loads are categorized as Design, Level A, or Level D loads. See the discussion below describing the bounding analysis where most of the Level D cases are evaluated using Design or Level A limits.

The loads due to Level D Service Loading Conditions, which are extremely unlikely conditions, are listed in Table 3.2-8.

Loading combinations for Design Conditions and Levels A and D Service Loadings which are evaluated are given in Table 3.2-9. The loads are listed across the top of the table and the Load Combinations are designated in the first column of the table. There are three Design Condition Load combinations listed, three Level A combinations and two Level D combinations. The loads which are acting simultaneously for each of these combinations are denoted by an "X" under the load column heading. For example, for Design Condition Load Combination Des (1), internal pressure due to cavity pressurization, fission gas release, distributed weight, wind, water or seismic load, and lid bolt preload are acting simultaneously.

Note that the bounding loads from Table 3.2-4 that envelop several other loads are used in the load combinations. When the superimposed bounding loads represent a Level D case as well as a Level A case (1g lateral load on cask conservatively bounds tornado wind load, flood water load and seismic load), the Level D case is effectively evaluated as a Design Case and the stress results will be later evaluated to Design Condition limits. Therefore, all of the combinations that would be Level D cases, except those including tornado missile loads, have been elevated to Design Conditions. These have also been combined with thermal effects as Level A combinations.

3.2.5.4.2 BASKET

Cask body internal and external pressures have no effect on the basket. External loads applied to the TN-40 cask do not result in basket loads unless the cask actually moves. Therefore, tornado wind and flood water produce no basket loads. Seismic loading, however, is an inertial loading since the cask and ISFSI pad experience both horizontal and vertical accelerations during an earthquake as discussed in Section 3.2.3 above. The seismic acceleration loading (much less than 1g acceleration) does combine with dead weight loading since these two effects occur simultaneously.

Temperature effects due to snow, cold rain, minor fire and even day/night cycles that can cause thermal transients on the outside of the cask body will not cause similar transients in the basket. The high heat capacity of the body slows the temperature response and effectively eliminates transients at the wall of the cask cavity. The steady state temperature and temperature differences throughout the basket are, however, affected by decay heat, solar insolation and ambient temperature variations.

The basket is important to control of criticality of the fuel assemblies stored in the cask. Therefore, since the basket loads described above are low, a conservatively high load level is assumed for the basket evaluation. Note that bounding lateral and vertical inertial loadings of the cask body equal to 1g (in each direction) have been shown to envelop the cask body loadings. For basket evaluation an even more conservative 3g loading is specified in both lateral and vertical directions.

The stresses in the 304 stainless steel portions of the basket due to the primary loading, 3g in any lateral direction combined with 3g vertical (including dead weight), are determined conservatively neglecting the tensile and bending strength of the aluminum thermal conductor plates between fuel compartment boxes. However, the through thickness strength of the aluminum plates which separate the boxes is considered. Thus the aluminum is conservatively neglected in the primary load analysis where it can react some of the load. These primary stresses in the steel are evaluated at the maximum metal temperature occurring under extreme ambient conditions.

The secondary (thermal) stresses in the stainless steel are calculated assuming elastic behavior of the steel but considering the actual strength of the aluminum. The local bearing stresses in the aluminum plates adjacent to the plugs are approximately five times the yield value when calculated elastically. The aluminum would therefore yield and creep resulting in lower thermal stresses in the stainless steel. The primary steel stresses calculated ignoring the aluminum (when it actually can react some of the load) are super-imposed on the secondary stresses calculated assuming the full strength of the aluminum is available to induce thermal stresses in the stainless steel. Therefore the primary plus secondary stresses determined for the 304 stainless steel fuel compartment boxes and their attachments in the basket are conservative. The basket design criteria described in Section 4.2.3.3.3 is based on Section III of the ASME Code for stress limits and buckling. The basket evaluation is also summarized in that section. The complete basket analysis is provided in Appendix 4B.

3.2.5.4.3 UPPER TRUNNIONS

The upper trunnions are considered to be lifting devices and they are evaluated to the ANSI N14.6 requirements during lifting operations. During lifting, the trunnions are evaluated for vertical lifting reactions applied on the centers of the lifting shoulders required to support 6 times or 10 times the maximum weight of a fully loaded cask. When the load is equal to 6 times the weight, the maximum tensile stresses shall not exceed the minimum yield strength of the trunnion material. For the load equal to 10 times the weight, the maximum tensile stresses shall not exceed the minimum ultimate tensile strength of the trunnion material.

In addition to the trunnions themselves, the welds that attach the trunnions to the cask body gamma shielding and the local region of the gamma shielding are analyzed under the same 6W and 10W reactions. The stresses in the welds and shielding shall not exceed the minimum yield strength of these components under the 6W loading nor the minimum ultimate strength under the 10W loading.

The loads acting on the trunnions are given in Table 3.2-2. The structural analysis of the trunnions is presented in Appendix 4A. A summary of the results and comparison with the design criteria are given in Section 4.2.3.

In a response to questions from the NRC Staff, NSP provided justification in Reference 30 for the adequacy of the load testing performed on the TN-40 cask trunnions during fabrication. In a Safety Evaluation dated May 11, 1995 (Reference 31) the NRC concluded that NSP had demonstrated that the trunnion-to-cask attachment welds have an acceptable level of quality and that load testing of the trunnions as described in Reference 30 provides an acceptable demonstration of the adequacy of the TN-40 cask trunnions.

3.2.5.4.4 OUTER SHELL

The outer shell is evaluated for the combined effects of inertia g loads due to lifting and internal pressure.

Outgassing from the resin between the cask body and outer shell may cause a slight pressure on the inside of the outer shell. A pressure relief valve is provided in the outer shell to assure any pressure build-up is small. The design is based on several years of neutron shield operating experience with this type of resin. The outer shell is completely supported by the resin when subjected to an external pressure. An internal pressure of 3 psi will occur due to the reduced external pressure during a tornado. However, since the cask body is designed for an external pressure of 25 psi, an internal pressure of 25 psi is conservatively used to evaluate the outer shell.

The structural analysis of the outer shell is presented in Appendix 4A. A summary of results and comparison with design criteria are given in Section 4.2.3.

The combined stress due to the inertia g loads and pressure is less than the minimum yield strength of the outer shell material.

3.3 SAFETY PROTECTION SYSTEMS

3.3.1 GENERAL

The TN-40 dry storage cask is designed to provide storage of spent fuel for at least 25 years (the ISFSI is licensed for 20 years). The cask cavity pressure is always above ambient during the storage period as a precaution against the in-leakage of air which might be harmful to the fuel. Since the containment vessel consists of a steel cylinder with an integrally-welded bottom closure, the cavity gas can escape only through the lid closure system. In order to ensure cask leak tightness, two systems are employed. A double barrier system for all potential lid leakage paths consisting of covers with multiple seals is utilized. Additionally, pressurization of monitored seal interspaces provides a continuous positive inward and outward pressure gradient which guards against a release of the cavity gas to the environment and the admission of air to the cavity.

3.3.2 PROTECTION BY MULTIPLE CONFINEMENT BARRIERS AND SYSTEMS

3.3.2.1 CONFINEMENT BARRIERS AND SYSTEMS

A combined cover-seal pressure monitoring system (Figure 3.3-1) always meets or exceeds the requirement of a double barrier closure which guarantees tight, permanent containment. There are two lid penetrations, one for a drain pipe and one for venting and pressurization. When the cask is placed in storage, a pressure greater than that of the cavity is set up in the gaps (interspaces) between the double metallic seals of the lid and the lid penetrations. A decrease in the pressure of the monitoring system would be signalled by a pressure transmitter mounted at the side of the cask (Figure 3.3-1). The system is pressurized through a fill valve mounted near the overpressure tank. Lead shielding will be provided to reduce radiation exposure to the transmitter to acceptable levels.

Connections to the overpressure tank are welded fittings. A quick connect coupling with a diaphragm valve is used to fill the tank.

The Helicoflex metallic face seals of the lid and lid penetrations possess long-term stability and have high corrosion resistance over the entire storage period. These high performance seals are comprised of two metal linings formed around a helically-wound spring. The sealing principle is based on plastically deforming the seal's outer lining. Permanent contact of the lining against the sealing surface is ensured by the outward force exerted by the helically-wound spring. Additionally, all metallic seal seating areas are stainless steel overlay for improved surface control. The overlay technique has been used for Transnuclear's transport casks.

For protection against the environment, a toruspherical protective cover equipped with an elastomer seal is provided above the lid. The lid and cover seals described above are contained in grooves. A high level of sealing over the storage period is assured by utilizing seals in a deformation-controlled design. The deformation of the seals is constant since bolt loads assure that the mating surfaces remain in contact. The seal deformation is set by its original diameter and the depth of the groove.

Metal gasket face seal fittings, diaphragm valves and Helicoflex metallic seals are all capable of limiting leak rates to less than 1 x 10^{-7} atm-cc/sec of helium.

The cask cavity's maximum operating pressure is 2.2 atm. This pressure assures a storage pressure of 1 atm with a minimum ambient temperature of -40°F. The maximum operating pressure of 2.2 atm corresponds to the maximum decay heat load of 0.675 kw per assembly, insolation, a 100°F ambient temperature and the storage of the casks in a 2 x 12 array.

During normal storage, cavity pressure variations due to changing ambient conditions will be small. However, fuel clad failure could result in an increase in cavity pressure due to free gas release of the fuel rods. Based on data from Reference 9, the Exxon assembly contains the most free gas, with 9.04 m^3 at standard temperature and pressure (40 assemblies). The TN-40 cask has a cavity free volume of 6.35 m^3. A 10% release of fission gas would cause an increase in cavity pressure of about 3.8 psi at an average cavity gas temperature of 439°F (Table 3.3-1).

The pressure assuming a 100% fuel failure and the off-normal cavity gas temperature is calculated to be 4.76 atm (70.0 psia).

The initial operating pressure of the monitoring system's overpressure tank is set at 5.5 atm minimum. Over the storage period, the pressure decreases as a result of leakage from the system and as a result of temperature reduction of the gas in the system. Since the level of permeation through the containment vessel is negligible and leakage past the higher pressure of the monitoring system is physically impossible, a decrease in cavity pressure during the storage period occurs only as a result of a reduction in the cavity gas temperature with time. As long as the cavity pressure is greater than ambient pressure and the pressure in the monitoring system is greater than that of the cavity, no in-leakage of air nor out-leakage of cavity gas is possible.

The calculations which follow define the monitoring system helium test leakage rate which ensures that no cavity gas can be released to the environment nor air admitted to the casks for the 25 year storage period. All seals are considered collectively in the analysis as the monitoring system pressure boundary.

The instantaneous leakage (L) from the monitoring system is equal to the leakage rate L_x during an infinitely small time period dt and L is also equal to the system volume V times the change in system pressure dp.

$$L = Vdp/dt$$

$$dt = \frac{-Vdp}{L} \qquad (1)$$

The leakage can either be choked or unchoked flow. From ANSI N14.5 (Reference 10), choked flow can be assumed if the ratio $P_d/P_u \leq r_c$.

where:

P_d = downstream pressure

P_u = upstream pressure

r_c = maximum ratio of P_d/P_u to permit choked gas flow

For helium, r_c = 0.485. It is conservatively assumed for this evaluation that P_d is always 1.0 atm; therefore, choked flow will occur if the pressure in the monitoring system is greater than 2.05 atm.

From Reference 10, for choked flow:

$$L_r = L\sqrt{\frac{0.583}{\frac{K}{k+1}}\frac{M}{29}\frac{298}{T}}\,\frac{1}{P_u}\,\frac{0.634}{\left(\frac{2}{k+1}\right)}\left(\frac{1}{k-1}\right)$$

For He, k=1.66 and M=4. Substituting and rearranging equation (1) becomes:

$$dt = \frac{-0.977V\sqrt{38.399/T}}{L_r}\frac{dP_u}{P_u}$$

After integration:

$$P = P_0\, exp-\left\{{tL_r}\Big/\left(0.977V\sqrt{38{,}399\big/T}\right)\right\} \qquad (2)$$

where:

P_0 = *pressure at time* t_0

L_r = *test leak rate (atm-cc/sec)*

V = *monitoring system volume (14,750 cc)*

T = *monitoring system gas temperature (°K)*

t = *time (sec)*

For unchoked flow, the simplified equation in Reference 10, example 21 can be used:

$$L_r = L \frac{\mu}{0.185} \frac{0.99}{P_u - P_d}$$

Substituting into equation (1) and integrating:

$$P = (P_0 - 1) \exp - \{ t L_r/(53.51\, V \mu)\} + 1 \qquad (3)$$

where: μ = *viscosity (cp)*

The integration of these equations accounts for the change in the monitoring system leakage rate which occurs during the storage period because of the change in the monitoring system pressure. Although the cask cavity pressure is initially 2.2 atm and remains above ambient during the storage period, it is conservatively assumed in this analysis that both the inner and outer seals of the monitoring system are subject to a constant 1.0 atm downstream pressure.

Since these equations only account for the monitoring system pressure loss due to leakage, a correction of pressure based on a decrease in system gas temperature was employed at the end of each three month time step at which the relation was evaluated.

In order to ensure that the monitoring system pressure is simultaneously greater than the ambient and the cavity gas pressures, both the monitoring system gas and the cavity gas temperatures were established as a function of time. The cavity gas and the monitoring system gas temperatures were calculated for both the beginning and end of storage conditions and they were assumed to decrease linearly during the storage period. The end of storage condition for this calculation was conservatively assumed to be 0.43 kw/assembly and an ambient temperature of 100°F. It was determined that the cavity pressure would decrease from an initial value of 2.2 atm to 1.93 atm in 20 years due to an average gas temperature reduction. The viscosity of the monitoring system gas was also corrected for temperature change for each successive time increment evaluated.

Equations 2 and 3 were used to calculate the monitoring system pressure as a function of time. A time step of three months was utilized. Choked flow conditions (Equation 2) was used until the calculated pressure decreased below 2.05 atm. Equation 3 was then used to calculate the pressure.

The determination of the maximum allowable test leakage rate is graphically depicted in Figure 3.3-2. According to this figure, the cavity pressure of 1.93 atm after twenty years is equivalent to the monitoring system pressure after twenty years when the helium test leakage rate is 1.0 x 10^{-5} atm-cc/sec. In Figure 3.3-3, the change in both the monitoring system gas and the cavity gas pressures are shown as a function of time for various test leak rates.

3.3.2.2 HEAT TRANSFER DESIGN

The TN-40 packaging is designed to passively reject decay heat under normal conditions of storage and hypothetical accident conditions while maintaining appropriate packaging temperatures and pressures within specified limits. An evaluation of the TN-40 thermal performance is presented in this section. Objectives of the thermal analyses performed for this evaluation include:

- Determination of maximum and minimum temperatures with respect to material limits.
- Determination of temperature distributions for analysis of thermal stresses.
- Determination of temperatures for containment pressurization.

Section 3.4 and Table 3.4-1 present the principal design bases for the TN-40 packaging.

A significant thermal design feature of the TN-40 is the basket described in Section 1.3. The basket consists of an assembly of 40 stainless steel fuel compartments with aluminum and boral plates sandwiched between them. The compartments are plug-welded together to form the basket. Aluminum plates are strategically welded to the basket periphery to enhance conduction heat transfer to the cavity wall. The design of the basket allows the heat from the fuel assemblies to be conducted along the plates to the periphery of the basket and dissipated to the cavity wall.

Another design feature is the conduction path created by the aluminum boxes in the neutron shielding layer described in Section 1.3. The neutron shielding is provided by a resin compound cast into long slender aluminum containers placed around the cask shell and enclosed within a smooth outer shell. By butting against the adjacent shell surfaces, the aluminum containers allow decay heat to be conducted across the neutron shield.

The TN-40 dry storage cask falls under the jurisdiction of 10CFR72 when used as a component of an ISFSI. To establish the heat removal capability, several thermal design criteria are established for the TN-40 cask. These are:

- Containment of radioactive material and gases is a major design requirement. Seal temperatures must be maintained within specified limits to satisfy the leak tight containment function during normal and accident conditions. A maximum temperature limit of 570°F (299°C) is set for the Helicoflex seals (double metallic O-rings) in the containment vessel closure lid.

- To maintain the stability of the neutron shield resin during normal storage conditions, a maximum temperature limit of 300°F (149°C) is set for the neutron shield.

- Maximum temperatures of the containment structural components must not adversely affect the containment function.

- Maintaining fuel cladding integrity during storage is another design consideration. To minimize creep deformation that can occur over the storage duration, the maximum initial storage fuel cladding temperature is determined as a function of the initial fuel age using the guidelines provided by the Commercial Spent Fuel Management Program (CSFM) (Reference 11). These temperature limits are reported in Section 3.3.7.1.

In general, all the thermal criteria are associated with maximum temperature limits and not minimum temperatures. All materials can be subjected to the minimum environment temperature of -40°F (-40°C) without adverse effects.

The TN-40 cask is analyzed based on a maximum heat load of 27 kw from 40 fuel assemblies. The thermal evaluation concludes that with this heat load all design criteria are satisfied for normal conditions. A summary of the results from the analyses performed for normal conditions is provided in Table 3.3-1.

Table 3.3-2 lists the thermal properties of materials used in the thermal analyses. The values are listed as given in the corresponding references. The analyses use interpolated values when appropriate for intermediate temperatures where the temperature dependency of a specific parameter is deemed significant. The interpolation assumes a linear relationship between the reported values.

Thermal radiation effects at the external surface of the packaging are considered. The external surfaces of the TN-40 cask are painted white (emissivity = 0.95, solar absorptivity = 0.18) (Reference 12). To account for dust and dirt, an emissivity of 0.9 and a solar absorptivity of 0.3 are used for exterior surfaces in the thermal models.

Normal ambient conditions of storage are site specific and are discussed in Section 2.3. The short term (OFF-NORMAL) environmental conditions are used for the calculation of maximum TN-40 component temperatures, containment pressure and thermal stresses. These steady state conditions correspond to the maximum daily averaged conditions, and are an ambient temperature of 100°F and a solar heat load of 135 Btu/hr-ft^2. The cladding temperatures, for the fuel cladding integrity evaluation, are based on long term (NORMAL) environmental conditions. These are a yearly averaged maximum ambient temperature of 50°F and a solar heat load of 108 Btu/hr-ft^2. The analyses include the effect of storing the TN-40 in an array that is 2 wide and infinitely long.

3.3.2.2.1 THERMAL MODEL

A three-dimensional finite element computer model of the TN-40 cask is used to simulate heat transfer in the packaging. The ANSYS computer program (Reference 13) is utilized for the analyses. This program is a large scale, general purpose finite element computer code which can perform steady state and transient three-dimensional thermal analyses.

The thermal model represents the TN-40 cask standing vertically on the concrete pad. The model includes the geometry and material properties (Table 3.3-2) of the basket, the cask shells, the neutron shield (resin in aluminum containers), the outer shell and the concrete pad. A quarter slice of the TN-40 cask is modeled with the appropriate symmetry boundary conditions. Figures 3.3-4 and 3.3-5 show sketches of radial and axial cross sections through the model.

The basket is composed of 40 stainless steel boxes (8.05 x 8.05 x 160 in.) with two 0.25 in. thick aluminum and one 0.075 in. thick boral plates placed between adjacent boxes. The boxes are held together by welded plugs which pass through the aluminum and boral plates. The plug welding design causes the aluminum and boral plates to be tightly sandwiched between adjacent box sides. The basket portion of the thermal model simulates the conduction paths provided by the aluminum plates. No credit is taken for the heat transfer paths provided by the stainless steel boxes and the boral plates. The heat flux from the fuel assemblies is applied directly to the aluminum plates. As shown in Figure 3.3-4, some aluminum plates are interrupted to allow other plates a direct conduction path to the basket periphery. As a conservative modeling approach, a 0.02 in. gap is used between the interrupted and continuous plates. This causes heat to be transferred across a gaseous medium (helium) between the two plates rather than along the more conductive stainless steel boxes sandwiching the gap, and provides conservative basket and cladding temperature results for all as-built gap sizes in the TN-40 casks.

Aluminum plates (0.38 in. thick)[(1)] are welded to the basket periphery to increase the surface area for heat dissipating while providing structural support for the basket. These peripheral plates are sized and formed to contact the cavity surface at thermal equilibrium. However, the thermal model assumes a nominal gap of 0.06 in. between these plates and the cavity surface. The aluminum rails bolted to the cavity wall are sized so that heat is conducted from the basket periphery across a 0.06 in. gap. These gaps are shown in Figure 3.3-4. The peripheral aluminum plates may be interrupted axially to allow for dimensional adjustment at fabrication or assembly. To factor in the loss of axial conduction along these plates, the thermal conductivity in the axial direction is assigned a value of zero. No heat can be dissipated from the basket to the cavity without passing across a gaseous gap (helium) by conduction. Other modes of heat transfer are conservatively neglected.

The plug holes in the aluminum basket plates reduce their thermal conductivity. To evaluate the effect of the holes, simple two-dimensional finite element models of the plates with and without holes were developed. It was concluded that for the same temperature difference across symmetry boundaries, the conductivity for the basket plates are effectively reduced 9.3% with plug holes. Accordingly, the thermal conductivity for the basket plates are reduced 10% in the thermal model.

The ANSYS three-dimensional isoparametric thermal shell element, STIF 57, is used to simulate heat transfer along the aluminum plates and across helium gaps. At the periphery, the three-dimensional isoparametric solid element, STIF 70, is used for gaseous conduction across gaps between the basket and the cavity wall

The cask body portion of the model consists of the cask bottom, the inner containment and outer gamma shield cask body shells, the neutron shield (resin in aluminum containers) and the outer shell. The model of the cask body extends 173.25 in. from the bottom. The cask components above and including the lid are not modeled and assumed to provide an adiabatic boundary. Figures 3.3-4 and 3.3-5 show the cask components in the thermal model.

The inner and outer cask body shells will be assembled with an interference fit. This will assure thermal contact at the shell interface. A contact conductance of 350 Btu/hr-ft^2 is used based on the data provided in Reference 14 for interface resistance between steel surfaces in contact (air gaps). In the bottom region of the cask, the two cask bottom plates are assumed to be separated by a 0.125 in. air gap. The cask bottom is also modeled with no thermal contact with the basket bottom.

The neutron shield consists of 60 long slender resin-filled aluminum containers placed between the cask body and outer steel shell. The aluminum containers butt against the shells. However, an air gap of 0.01 in. is used in the model.

[(1)] May be ground to less than 0.38 inch locally to fit-up in cask body. See Note 4 on proprietary version of Figure 1.3-7.

The concrete pad in the thermal model is 36 in. thick and extends 36 in. around the bottom of the cask. The bottom of the cask is assumed to be in perfect thermal contact with the concrete pad. The bottom of the pad, which is in contact with soil, is treated as a constant temperature boundary. For the off-normal ambient conditions, a soil temperature of 70°F is used. A soil temperature of 50°F is used for the yearly averaged normal ambient conditions.

The finite element model for the stainless steel shells, the resin, the air gaps and the concrete pad, are developed using STIF 70 elements. The aluminum containers for the resin are generated using STIF 57 elements.

Figure 3.3-6 shows the three-dimensional ANSYS finite element model developed.

Solar Heat Load

The maximum solar heat load is applied as a constant value to all external surfaces of the thermal model, without taking any credit for the night period when there is no insolation. A solar absorptivity of 0.3 is used for the painted surfaces of the cask and 0.9 for the concrete pad.

Decay Heat Load

The fuel assemblies are not included in the thermal model. Instead, an axially varying heat flux corresponding to decay heat load of 0.675 kw per assembly is applied to the basket surfaces forming the fuel compartments. The heat flux profile for a typical PWR fuel assembly with a peak power factor of 1.2 and an active length of 144 in. is used. Figure 3.3-7 shows the axial power profile assumed for the analyses.

Heat Dissipation to the Environment

Most of the heat from the TN-40 packaging is dissipated to the environment by radiation and natural convection. If the packaging is stored in an array, partial radiation "blockage" occurs which reduces the overall view factor from the packaging to the environment. The analyses assume that the packagings will be stored in an array that is two wide and infinitely long and placed at least 18 ft. (center to center) apart (Figure 3.3-8). Convection heat transfer is assumed to be unaffected. Heat transfer between packagings is neglected. An emissivity of 0.9 is used for all external surfaces.

To simplify the packaging environment view factor calculation, the TN-40 cask is assumed to be a cylinder of 8.4 ft. diameter and 12.7 ft. length. This represents the surface dimensions for the neutron shell. Based on its location in a two wide and infinitely long array, it is possible for 42.9% of the neutron shell surface area to be surrounded by other packagings. Radiation heat transfer for this "blocked" region can be approximated as that between two concentric cylinders with the inner radius corresponding to the cask diameter and the outer radius corresponding to the array spacing of 18 feet. The equation for the view factor F_{2-1} is obtained from Reference 15 for two concentric cylinders of finite length and is:

$$F_{2-1} = (1/R) - [1/(\pi R)] \cos^{-1}(B/A) + [1/(2\pi RL)]\left([(A+2)^2-(2R)^2]^{1/2}\cos^{-1}(B/RA)\right) - [B/(2\pi RL)][\sin^{-1}(1/R)] + [A/(4RL)]$$

where,

$A = L^2 + R^2 - 1$

$B = L^2 - R^2 + 1$

$L = h/r_1$

$R = r_2/r_1$

r_1 = radius of inner cylinder

r_2 = radius of outer cylinder

h = height of cylinders

From the reciprocity theorem,

$$F_{1-2} = F_{2-1}(r_2/r_1)$$

The view factor from the outer surface to the environment,

$$F_{1-amb} = 1 - F_{1-2}$$

Based on an array spacing of 18 ft. and a packaging diameter of 8.4 ft., an overall view factor of 0.8138 is calculated between the packaging and the ambient.

The heat transfer coefficient, H_r, for heat dissipation by radiation, is given by the equation

$$H_r = (0.1714E\text{-}8)(G_{12})(T_s+460)^4-(T_a+460)^4]/(T_s-T_a)$$

$$(Btu/hr\text{-}ft^2\text{-}°F)$$

where,

G_{12} = the gray body exchange coefficient

= (surface emissivity) (view factor)

For the horizontal surfaces,

G_{12} = (0.9)(1) = 0.9

For the vertical blocked surfaces,

G_{12} = (0.9) (0.8138) = 0.7324

T_a = ambient temperature, °F

T_s = surface temperature, °F

Heat dissipation by natural convection from vertical surfaces is described by the following equation (Reference 12).

$$Nu_L = 0.13(GR_L\,Pr)^{1/3} \text{ for } Gr_L > 10^9$$

where,

Nu_L = Average Nusselt number

Gr_L = Grashof number

Pr = Prandtl number

Simplifying in terms of H_c, the natural convection coefficient,

$$H_c = 0.13(k)\ [(T_s-T_a)(Pr)(\rho^2 g\beta/\mu^2)]^{1/3}$$

$$(Btu/hr\text{-}ft^2\text{-}°F)$$

where,

ρ = *density, lb/ft*^{3}

g = *acceleration due to gravity, ft/sec*^{2}

β = *temperature coefficient of volume expansion, 1/R*

μ = *absolute viscosity, lb/ft-sec*

k = *conductivity, btu/hr/ft.*

For horizontal surfaces,

$$Nu_L = 0.16\ (GR_L\ Pr)^{1/3} \qquad \textbf{(Reference 12)}$$

and

$$H_c = 0.16(k)\ [(T_s\text{-}T_a)(Pr)(\rho^2 g\beta/\mu^2)]^{1/3}$$

The total heat transfer coefficient $H_t = H_r + H_c$, is applied as a boundary condition on the outer surfaces of the finite element model.

<u>Maximum Fuel Cladding Temperature</u>

The finite element model does not include the fuel assemblies. Instead, the maximum fuel cladding temperature is evaluated after obtaining a steady state temperature distribution using the finite element model. For conservatism, the average temperature of the hottest fuel compartment is assumed to be the maximum basket temperature.

The Wooton-Epstein equation (Reference 16) is used to calculate the maximum fuel cladding temperature. This is a semi-empirical, semi-theoretical correlation which accounts for natural convection (in air) and radiation cooling of a spent fuel assembly in a horizontal cask. The TN-40 cask is stored vertically and is back-filled with helium. Based on experimental studies performed on the TN-24P (Reference 17), lower fuel cladding temperatures were reported for a helium (vertical cask) instead of a nitrogen (horizontal cask) back-fill gas medium. Hence it is conservative to use the Wooton-Epstein correlation for the fuel cladding temperature calculation. The correlation is:

$$q = 4WL_a\left[sC_1(1/E_r + 1/E_w - 1)(T_r^4 - T_w^4) + C_2(T_r - T_w)^{4/3}\right]$$

Note: A modified Wooton-Epstein correlation was used to evaluate the reduced thickness of the aluminum periphery plates. (See footnote on page 3.3-7 and Note 4 on the proprietary version of Figure 1.3-7.)

where,

q = heat dissipation from the fuel assembly, Btu/hr

W = width of the fuel assembly, ft

L_a= *active length of fuel assembly, ft*

s = Stefan-Boltzmann constant (0.171E-8 Btu/hr-ft^2- °R^4)

C_1= *4/(N+2) (geometric constant)*

E_r= *emissivity of fuel cladding*

E_w= *emissivity of compartment walls*

N = number of fuel rods per side

C_2= *0.118 (an experimentally determined constant)*

T_r= *maximum fuel cladding temperature, °R*

T_w= *maximum basket temperature, °R*

A Westinghouse 14x14 OFA, 3.85 w/o fuel assembly is selected as the basis assembly. An axial power peak factor of 1.2 was assumed in the heat load calculation. For a Westinghouse 14x14 assembly,

W = 0.647 ft

L_a= 12 ft

C_1= 0.25

E_r= 0.8 (from Reference 17)

E_w= 0.15 (fuel compartment emissivity)

q = (0.675 kw) $\times$ 1.2 $\times$ (3413 Btu/hr-kw) = 2764.5 Btu/hr

Average Cavity Gas Temperature

The cavity gas temperature is maximum at the hottest fuel cladding and minimum at the cooler surfaces in the lid region. For simplicity and conservatism, it is assumed that the average cavity gas temperature is the average value of the maximum fuel cladding and the lid region temperatures.

3.3.2.2.2 RESULTS OF THE THERMAL ANALYSES

Short-Term OFF-NORMAL Storage Conditions

A steady state thermal analysis is performed using the maximum decay heat load of 0.675 kw per assembly (27 kw total), 100°F ambient temperature and an insolation of 135 Btu/hr-ft^2. Figure 3.3-9 shows the temperature distribution predicted by the finite element model. The specific temperature distributions in the hottest cross section of the model, the hottest cross section of the basket, the top 5 in. of the basket, and the cask body are shown in Figures 3.3-10, -11, -12 and -13 respectively. The maximum fuel cladding and average cavity gas temperatures are calculated using the appropriate steady state component temperatures. A summary of the calculated packaging temperatures are listed in Table 3.3-1.

Long-Term NORMAL Storage Conditions

The NORMAL thermal analysis assumes a maximum decay heat of 0.675 kw per assembly, 50°F ambient temperature and a solar heat load of 108 Btu/hr-ft^2.

The steady state temperature distribution in the finite element model is shown in Figure 3.3-14. This is used to calculate the maximum fuel cladding and the average cavity gas temperature. Table 3.3-1 contains a summary of the calculated packaging temperatures.

Evaluation of Packaging Performance

The thermal analysis for normal storage concludes that the TN-40 cask design meets all applicable requirements. The maximum temperatures calculated using conservative assumptions are low. The maximum temperature of any containment structural component is less than 303°F (151°C) which has an insignificant effect on the mechanical properties of the containment materials used. The maximum seal temperature (242°F, 117°C) during normal storage is well below the 570°F long term limit specified for continued seal function. The maximum neutron shield temperature is below 300°F (149°C) and no degradation of the neutron shielding is expected during the 25 year storage life. The long-term maximum fuel cladding temperature is 602°F (317°C) and within allowable fuel temperature limits (Section 3.3.7.1). The minimum temperature of -40°F (-40°C) is also inconsequential to the packaging function.

Buried Cask Thermal Evaluation

The TN-40 cask dissipates heat to the environment by radiation and natural convection. If the packaging is accidentally buried in medium that will not provide the equivalent cooling of natural convection and unrestricted radiation to the environment, component temperatures will increase to a higher steady state condition after long-term burial. Of interest is the containment integrity which is assured as long as the metallic seals remain below 570° and the cavity pressure is less than 100 psig.

The temperature response of the TN-40 cask is evaluated using a cross-sectional slice of the finite element model developed for the normal storage analysis. For this analysis, the packaging is assumed to be completely buried in dry soil with such poor heat transfer characteristics that it effectively insulates the packaging. This conservative assumption eliminates any communication between the cask and a final heat sink. The resulting analysis therefore determines the time required to reach limiting temperatures for the containment integrity.

Initial conditions before burial are established by using steady state temperatures in the hottest cross section (Figure 3.3-10). The transient analysis is performed with a packaging heat load of 27 kw.

The results of the analysis show that if the packaging is not uncovered within 5 hours, the resin will start outgassing as its temperature exceeds 300°F (149°C). Thereafter, packaging component temperatures will increase by about 5°F (2.7°C) per hour. The cask body temperatures will reach 570°F about 60 hours after burial. At this time all temperatures in the packaging will be about 260°F (144°C) higher than those for initial conditions. The cavity pressure, if all fuel fails, will not exceed 100 psig. Thermal gradients in the packaging are small and less than those during normal storage. Figure 3.3-15 shows the maximum temperature/time history for the outer surface, neutron shielding, cavity wall and basket.

The ISFSI operating and emergency procedures will consider these time frames in planning for recovery from an accidental cask burial.

Hypothetical Fire Accident Thermal Evaluation

The hypothetical fire accident for the TN-40 cask is based on a fuel fire, the source of fuel being that from a ruptured fuel tank of the cask transporter tow vehicle. The bounding capacity of the fuel tank is 200 gallons and the bounding hypothetical fire is an engulfing fire around the cask.

From IAEA requirements (Reference 28), the "pool" of fuel is assumed to extend 1 meter beyond the cask surface. Based on an outer shell diameter of 101 inches, this gives a "pool" diameter of approximately 180 inches and a pool surface of 25,400 in^2. A fuel consumption rate of 0.15 in/min. was selected from a Sandia Report (Reference 29) concerning gasoline/tractor kerosene experimental burning rates. This translates into a fuel consumption rate of approximately 16.5 gal./min. Therefore, the 200 gallons of fuel will sustain a fire for about 12 minutes and hence a 15 minute fire is evaluated. The Sandia Report also reports an average flame temperature of 1550°F and an average convective heat transfer coefficient of 4.5 Btu/hr-ft^2-°F for a railroad tank car fire test. The same parameters are utilized for the fire accident evaluation.

The fire thermal evaluation is performed primarily to demonstrate the containment integrity of the TN-40. This is assured as long as the metallic lid seals remain below 570°F and the cavity pressure is less than 100 psig. Two models, a cross section model and a lid seal model, are used for the evaluation.

A. Cross Section Model

The cross section finite element model utilized for the buried cask thermal evaluation is reused here. The cask components and their geometry are shown in Figure 3.3-4. Initial temperatures before the fire condition are established by using steady state temperatures in the hottest cross-section (Figure 3.3-10) for short term OFF-NORMAL storage conditions. The analysis uses a cask heat load of 27 kw. The effects of insulation are neglected during the transient fire evaluation.

During the fire condition period (15 min.), heat absorption at the outer surface is by radiation and convection, and is given by the following equation:

$$q_{fire} = (H_c + H_r)(T_f - T_s)$$

where:

q_{fire} = heat flux into packaging from fire, Btu/hr-ft^2

T_f = flame temperature = 1550 °F

T_s = surface temperature of the packaging, °F

H_c = convection heat transfer coefficient = 4.5 Btu/hr-ft^2-°F

H_r = radiation heat transfer coefficient, Btu/hr-ft^2-°F

$$H_r = (0.1714E\text{-}8)(F_s)[(E)(T_f + 460)^4 - (T_s + 460)^4]/(T_f - T_s)$$

where:

F_s = outer surface absorptivity = 0.8 (Ref. 28)

E = flame emissivity = 0.9 (Ref. 28)

During the cooldown period after the fire condition, heat dissipation from the outer surface is by radiation and natural convection to an ambient temperature of 100°F (as in the short-term OFF-NORMAL conditions).

The results of the analysis show that no melting of the metallic cask components occurs. The peak transient temperatures in selected locations in the cask are listed in Table 3.3-8. The transient temperatures in the cask cavity increase by no more than 52°F (29°C) and the corresponding peak cavity pressure assuming 100% fuel failure is 74 psia.

B. Lid-Seal Region Model

To demonstrate the integrity of the seals in the lid during the fire accident, a finite element model of the top portion of the TN-40 is developed. The model is an axisymmetric two-dimensional model and includes the geometry and material properties of the lid, resin disk, protective cover and upper region of the cask body shells. Figure 3.3-20 shows the geometry of the model. Figure 3.3-21 is an element plot of the 2-D axisymmetric ANSYS model. Elements representing the same material have the same color.

The outer surfaces of the protective cover and the cask body are subjected to the heat flux from the 15 minute fire, and during the cooldown period, heat is dissipated from these surfaces to an ambient temperature of 100°F. Most of the heat transfer in the enclosure under the protective cover is by radiation in the fire condition. Hence, heat transfer in this enclosure is modeled by radiation with all surfaces being assigned an emissivity of 0.9. Near the seal region where the air gap between the lid and protective cover is small, heat conduction through air is assumed. The region where heat conduction through air is assumed is indicated in Figure 3.3-21. No other heat conduction through air is assumed under the protective cover. The initial temperature before the fire condition is 242°F, the short term OFF-NORMAL storage temperature for the lid region. The effects of insolation are neglected during the transient fire evaluation.

The results of the computer analysis show that no melting of the metallic components in the lid region occur. The maximum lid seal temperature peaks at 340°F (171°C).

C. Conclusion

Based on the thermal analyses for the fire accident conditions, the TN-40 packaging will withstand the hypothetical fire accident event without compromising the containment integrity of the TN-40. Peak seal temperature remains well below 570°F and the cavity pressure below 100 psig.

The maximum basket temperature peaks at 568°F (a 38°F increase). The transient maximum fuel temperature will increase no more than 38°F from 642°F (assumed initial short term OFF-NORMAL storage temperature) to 680°F (360°C). This transient temperature is well below the maximum operating fuel temperature in the reactor of 700°F (371°C).

The neutron shield will off-gas during the hypothetical accident. A pressure relief valve is provided on the outer shell to prevent the pressurization of the outer shell. Shielding analyses have been performed showing acceptable consequences even if all the resin disappears.

3.3.3 PROTECTION BY EQUIPMENT AND INSTRUMENTATION SELECTION

3.3.3.1 EQUIPMENT

Design criteria for the casks are described in Section 3.4 and summarized in Table 3.4-1.

3.3.3.2 INSTRUMENTATION

Due to the totally passive and inherently safe nature of the storage, safety-related instrumentation is not necessary. Instrumentation to monitor cask pressure is furnished. Appropriate capabilities to check and recalibrate these monitors is also provided. The pressure monitoring system is further described in Section 3.3.2.1.

3.3.4 NUCLEAR CRITICALITY SAFETY

3.3.4.1 CONTROL METHODS FOR PREVENTION OF CRITICALITY

The design criterion for criticality is that the effective neutron multiplication factor, k_{eff}, including statistical uncertainties, shall be less than 0.95 for all postulated arrangements of fuel within the cask.

The control methods used to prevent criticality are:

1. Incorporation of neutron absorbing material (boron) in the basket material.
2. Loading of the irradiated fuel assemblies in the fuel pool containing at least 1800 ppm boron.
3. revention of fresh water entering the loaded cask.

The basket has been designed to assure an ample margin of safety against criticality under the conditions of fresh fuel in a cask flooded with borated water. The methods of criticality control are in keeping with the requirements of 10CFR72.124.

Criticality analysis is performed using the KENO-V.A Monte Carlo code (Reference 18) along with data prepared using the NITAWL code (Reference 19) and the SCALE 27-group cross section library. These codes and cross-section library are part of the SCALE system prepared by Oak Ridge National Laboratory for the U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research (Reference 20). They are widely used for criticality analysis of shipping casks, fuel storage pools and storage casks. Benchmark problems are run to verify the codes, methodology and cross section library. Examples of computer input used for criticality evaluation are included in Appendix 3A (Proprietary).

In the criticality calculation, fuel assembly, basket, and cask wall geometries are modeled explicitly. Within each assembly, each fuel pin and each guide tube is represented. Materials are not homogenized.

Reactivity analyses (K-Infinite) were performed for the following Prairie Island fuel assemblies: Westinghouse Standard 3.41 w/o U235; Exxon Standard 3.47 w/o U235; Exxon TOPROD 3.82 w/o U235; and Westinghouse OFA 3.80 w/o U235. The calculated K-infinite were 1.3490, 1.3547, 1.3830, and 1.3930, respectively. These results indicate that the Westinghouse OFA type assembly is the most reactive. This assembly therefore was utilized for the criticality calculation at 3.85 w/o U235. The analysis assumes fresh fuel composition with 1,800 ppm borated water in the cavity, and the cask surrounded by a water reflector.

A three-dimensional criticality model is utilized. The plenum and end fitting areas are modeled as fresh water. The poison plates cover only the active fuel length. The 0.38 in. aluminum supports at the basket periphery are modeled as a 0.12 in. shell at the cavity wall. The geometric model for the fuel assembly is shown in Figure 3.3-16 and the cask model in Figure 3.3-17. The material compositions are given in Table 3.3-3.

The result of the calculation with 1800 ppm borated water and 3.85 w/o U235 is k_{eff}=0.8988$\pm$0.0033. Including the bias determined from benchmark calculations and 2 sigma yields k_{eff}=0.9162.

Criticality calculations were made to investigate the reactivity of the cask as the borated water is drained from the cavity. No significant reactivity peaks were found. Table 3.3-4 lists the results (not corrected for bias) of the calculations. A calculation was also performed with each of the assemblies moved to the inside corner of its fuel compartment, toward the cask centerline. The calculated k_{eff} for this configuration was 0.9032$\pm$0.0033 (no bias correction).

To approximate irradiated fuel placed into the cask, a calculation was performed using 1.9 w/o U235 and unborated water in the cask. The result of this calculation is k_{eff}=0.9306$\pm$0.0036, (uncorrected for bias). Calculations were also performed to evaluate the reactivity effect of low density water in the cask with this model. The results, Table 3.3-5, show that optimum moderation occurs with full density water.

Since the TN-40 cask will only contain water with at least 1,800 ppm boron, (seals are above the maximum hypothetical flood height), and irradiated fuel, criticality safety of the cask is assured.

3.3.4.2 ERROR CONTINGENCY CRITERIA

Provision for error contingency is built into the criterion used in Section 3.3.4.1 above. The criterion, used in conjunction with the KENO-V.A and NITAWL codes, is common practice for licensing submittals. Because conservative assumptions are made in modeling, it is not necessary to introduce additional contingency for error.

3.3.4.3 VERIFICATION ANALYSIS – BENCHMARKING

Five critical experiments by Pacific Northwest Laboratory (PNL) are used to validate the criticality analysis (References 21 and 22). The PNL experiments were designed to simulate conditions associated with both fuel element shipping packages and with fuel storage pools. The experiments selected are associated with critical separation between water flooded subcritical clusters of fuel rods with and without poison plates between clusters. Run A contained boral plates, Run B borated stainless steel plates, Run C SS304L plates, and Run D no poison plates. All four experiments have steel reflecting walls. The fifth experiment (E) contains no poison plates and no reflecting walls.

The NITAWL code is used to perform resonance calculations and to prepare the working library from the SCALIAS 27 group cross section library for input to KENO-V.a.

Fuel rods and dimensions are illustrated in Figure 3.3-18. The experimental geometry is shown in Figure 3.3-19. The poison plates, when present, are positioned at the outer cell boundary of the center fuel cluster. Dimensions are shown in Table 3.3-6. The KENO-V.a geometric representation includes the explicit model of each fuel rod and poison plate.

The results of the calculation are shown in Table 3.3-7. The results show a non-conservative average bias of 1.2%. Thus, the criticality results for the TN-40 are increased by a factor of 1.012.

3.3.5 RADIOLOGICAL PROTECTION

Provisions for radiological protection by confinement barriers and systems are described in Section 3.3.2.1.

3.3.5.1 ACCESS CONTROL

The ISFSI does not require the continuous presence of operators or maintenance personnel. In addition, it is located within a fenced-in area shared only with the Equipment Storage Building and Security Building which will be used for storage of cask handling and security related equipment and will not be continuously manned. Access to the fenced-in area is limited to personnel needed during operations at the ISFSI. Activities will include periodic inspections of these facilities, emplacement of storage casks, and security checks. These activities will be defined and controlled by the Radiation Protection and Security procedures manuals covering the ISFSI.

3.3.5.2 SHIELDING

The storage casks provide sufficient radiation shielding to allow handling of the loaded casks with as low as reasonably achievable (ALARA) doses to the operators and to comply with the radiation limits in 10CFR72. For specific dose estimates, see Section 7.

3.3.5.3 RADIOLOGICAL ALARM SYSTEMS

There are no credible events which could result in releases of radioactive products or unacceptable increases in direct radiation. In addition, the releases postulated as the result of the hypothetical accidents described in Section 8 are of a very small magnitude. Therefore, radiological alarm systems are not necessary. However, as described in Section 3.3.3.1, nonsafety-grade pressure monitors are provided. Procedures to be followed when these alarms are activated will be specified in the ISFSI operating procedures.

3.3.6 FIRE AND EXPLOSION PROTECTION

No hydrocarbon fuel of any sort will be stored in the ISFSI. The quantity of fuel carried in the tow vehicle will be limited so that only a small fire of short duration would be possible. There are no other significant combustible sources within the ISFSI security fence. Due to the large thermal mass of the casks any minor fires in the vicinity of the ISFSI would raise the cask temperature by only a few degrees and are not expected to affect cask integrity.

As indicated in Section 2.2, overpressures of 2.25 psi can be conservatively postulated to occur at the ISFSI as a result of accidents involving explosive materials which are stored or transported near the site. This impact is less than that postulated to result from the tornado wind loading and missile impact analysis, as described in Section 3.2.1, and is well within the design basis of the cask.

3.3.7 MATERIAL HANDLING AND STORAGE

3.3.7.1 SPENT FUEL HANDLING AND STORAGE

The handling of spent fuel within the Prairie Island Nuclear Generating Plant will be conducted in accordance with existing fuel handling procedures. Fuel that may be damaged as defined in Section 10.1.1 will not be considered for storage at the ISFSI.

Handling of the sealed casks outside of the Auxiliary Building in the process of emplacing them at the ISFSI will be done according to procedures that ensure that their safety functions and the power station capability for safe shutdown are not impaired. These operations are described in Section 5.4.

The design criteria for the TN-40 dry storage cask require that the maximum fuel cladding temperature of the hottest fuel rod in the cask shall not exceed the temperature limit calculated according to PNL-6189 (Reference 11). This temperature limit has been calculated as a function of fuel age to account for the effect of fuel age on creep deformation and fuel cladding rupture. As the age of fuel increases, its cooling rate rapidly decreases. If the initial fuel temperature is too high at loading, significant creep deformation can occur as a result of the decreasing cooling rates with fuel age. The Commercial Spent Fuel Management Program (CSFM) used the TN-24P packaging as one of its models for developing generic fuel cladding temperature limit curves for 40 year dry storage. The CSFM generic curves are used to establish the fuel cladding temperature limit for 10-year cooled fuel. The TN-40 has a storage life of 20 years and it is conservative to use the CSFM curves developed for 40 year storage.

From Reference 11, the midwall hoop-stress is given by the equation,

$$S_{mhoop}, T_2 = (PD_{mid}/2t)(a)(T_2/T_1)$$

where

S_{mhoop}, T_2 = *the midwall hoop=stress (psi) at temperature of interest* T_2 *(°K)*

P = *the internal pressure (psi) at the hot-volume average temperature,* T_1 *(°K)*

D_{mid} = *the midwall diameter (in.) accounting for cladding corrosion*

t = *the cladding thickness (in.)*

a = *0.95 for PWR fuel assemblies*

Using fuel data for the Exxon (ANF) and Westinghouse assemblies, it was determined that the Exxon fuel was the most conservative with a fuel rod pressure of 2,110 psia at 603°F (rod diameter=0.417 in., cladding thickness = .0295 in.) with a burnup of 43,000 MWD/MTU.

Substituting values and simplifying,

$S_{mhoop}, T_2 = 22.3\ T_2$ *psi/K for Exxon assemblies and*

$S_{mhoop}, T_2 = 14.6\ T_2$ *psi/K for Westinghouse 14 x 14 OFA lead rod at 50,000 MWD/MTU*

For conservatism, the Exxon fuel assembly equation is used to determine the fuel rod temperature limits. The temperature limits are determined graphically by plotting the midwall hoop-stress equation on the CSFM generic limit curves of Reference 11. The acceptable temperature limits obtained are 342°C and 335°C for 10 year and 15 year cooled fuel, respectively.

3.3.7.2 RADIOACTIVE WASTE TREATMENT

The ISFSI will not generate radioactive waste. However, cask loading and decontamination operations, while in the Auxiliary Building, may generate small amounts of waste. This waste is disposed of in accordance with the radioactive waste handling procedures described in Section 6, and is part of the 10CFR50 licensed activities. Waste storage facilities are neither required nor provided for the ISFSI.

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3.4 SUMMARY OF STORAGE CASK DESIGN CRITERIA

3.4.1 Cask Design Criteria 01204149

The principal design bases for the TN-40 cask are presented in Table 3.4-1. The TN-40 dry storage cask is designed to store 40 intact 14x14 PWR spent fuel assemblies, with a maximum assembly average burnup of 45,000 MWD/MTU and a minimum cooling time of 10 years.

The maximum total heat generation rate of the stored fuel is limited to 27 kw in order to keep the maximum fuel cladding temperature below the limit necessary to ensure cladding integrity for 40 years storage (Reference 11). The fuel cladding integrity is assured by the limited fuel cladding temperature and maintenance of a nonoxidizing environment in the cask (Reference 23).

The containment vessel (body and lid) is designed and fabricated to the maximum practicable extent as a Class I component in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Articles NB-3200. The cask design, fabrication and testing are covered by a Quality Assurance Program which conforms to the criteria in 10CFR72(G).

The cask is designed to maintain a subcritical configuration during loading, handling, storage and accident conditions. Poison materials in the fuel basket are employed to maintain $k_{eff} \leq 0.95$ including statistical uncertainties. The TN-40 cask is designed to withstand the effects of severe environmental conditions and natural phenomena such as earthquakes, tornados, lightning, hurricanes and floods. Section 8 describes the cask behavior under these environmental conditions.

3.4.2 Design Basis Limits for Fission Product Barriers (DBLFPBs)

The NRC has defined the design basis limit for a fission product barrier as the controlling numerical value for a parameter established during the license review as presented in the Safety Analysis Report for any parameter(s) used to determine the integrity of the barrier. The list of DBLFPBs for the TN-40 cask is listed in Table 3.4-2. 01204149

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TABLE 3.1-1
FUEL ASSEMBLY PARAMETERS

Maximum weight, without control component or BPRA	1,300 lb
Assembly dimensions:	
Without control component	7.763"x7.763"x161.3"
Fuel rod array	14 x 14
Number of fuel rods	179
Active fuel length	144"
Initial enrichment (maximum)	3.85 w/o U235
Burnup (maximum)	45 GWD/MTU
Cooling time (minimum)	10 YEARS
Initial uranium content:	
maximum	410 kgU
minimum	350 kgU

	Westinghouse Standard	OFA	Exxon Standard	TOPROD
Fuel Pellet O.D., in	0.3659	0.3444	0.3565	0.3505
Fuel Rod O.D., in	0.422	0.400	0.424/0.426	0.417
Clad Thickness, in	0.0243	0.0243	0.030/0.031	0.0295
Clad Material	Zr-4	Zr-4	Zr-4	Zr-4

TABLE 3.1-2
THERMAL, GAMMA AND NEUTRON SOURCES FOR
THE DESIGN BASIS 14 X 14 WESTINGHOUSE OFA FUEL ASSEMBLY

U235 Enrichment (w/o)	3.85
Burnup (MWD/MTU)	45,000
Specific Power (MW/MTU)	37.5
Cooling Time (yr)	10
Decay Heat (kw)	0.675
Gamma Source (photons/sec)	2.44E+15
Neutron Source (n/sec)	2.19E+8

TABLE 3.2-1
SUMMARY OF TN-40 WEIGHTS

COMPONENT	WEIGHT, LB.
Body	116,334
Bottom	18,866
Lid	13,907
Aluminum Boxes	1,991
Resin	10,578
Outer Shell	7,453
Polypropylene Top Neutron Shield (1)	1,692
Trunnions	563
Protective Cover (2)	1,465
Basket and Rails	15,841
Fuel Assemblies	52,000
Weight on Storage Pad (3)	240,690

Notes:

(1) Installed after cask moved to Auxiliary Building rail bay.

(2) Installed after cask moved to Auxiliary Building rail bay or ISFSI.

(3) Actual weight(s) will vary. Fully loaded cask weight is 244,000 lbs. max (Reference 32).

TABLE 3.2-2
SUMMARY OF LIFTING LOADS USED IN UPPER TRUNNION ANSI N14.6 ANALYSIS OF TN-40 CASK

Handling Condition	Load in g's at cask CG (1) vertical
Lifting-Cask Vertical	
Yield Evaluation	1.5×10^6 lb.
Ultimate Evaluation	2.5×10^6 lb.
	Load at each trunnion (2)
Yield Evaluation	0.75×10^6 lb.
Ultimate Evaluation	1.25×10^6 lb.
	Moment at trunnion/body interface (2)
Yield Evaluation	3.90×10^6 in-lb.
Ultimate Evaluation	6.50×10^6 in-lb.

Notes:
(1) Based on a cask weight of 250,000 lb.
(2) Load evenly divided between one pair of upper trunnions.

TABLE 3.2-3
SUMMARY OF INTERNAL AND EXTERNAL PRESSURES ACTING ON TN-40 CASK

Loading Condition	Maximum Pressure, psig
Internal Pressure:	
(a) Initial cavity pressurization	12
(b) With 10% fuel failure	15
(c) In minor fire	12
(d) Beginning of life unloading	56
(e) Tornado	3*
(f) Selected bounding pressure	100
External Pressure:	
(a) Flood	7
(b) Snow and ice loading	0.35
(c) Fuel loading & unloading	20
(d) Earth burial	25
(e) Selected bounding pressure	25

*This is due to a reduced external pressure

TABLE 3.2-4
SUMMARY OF LOADS ACTING ON TN-40 CASK DUE TO ENVIRONMENTAL AND NATURAL PHENOMENA

Distributed Loads		
Lateral Loading:		
(a) Maximum Wind or Water (external force on cask body)		46,750 lb.
(b) Seismic (inertial force throughout system)	0.18W=	45,000 lb.
Selected Bounding Load W_{max} x 1G=		250,000 lb.
Vertical Loading (1)		
(a) Seismic (inertial force throughout system)	0.12W=	30,000 lb.
Selected Bounding Load W_{max} x 1G=		250,000 lb.
Local Loads		
Tornado Missile Loading (external force on local area of body):		
(a) Lateral Load		442,080 lb.
(b) Vertical Load		442,080 lb.

Note:

(1) Does not include dead weight or lifting loads.

TABLE 3.2-5
TN-40 CASK LOADING CONDITIONS

Normal

Assembly Loads (bolt preload and seal compression)
Pressure (internal and external)
Weight
Lifting Loads
Handling
Wind
Thermal variations (e.g. insolation, decay heat, rain, snow, ice, ambient)

Man-Made

Fuel cladding failure
Minor fire
Explosion
Cask Burial

Natural Phenomena

Earthquakes
Tornados
Hurricane
Flood
Tsunami
Seiches
Lightning

TABLE 3.2-6
TN-40 CASK DESIGN LOADS

Applied Load	Loading Condition
Internal Pressure	(1) and (2)
External Pressure	(3)
Distributed Loads	Weight Cask Body Contents Snow Ice Wind (Tornado) Lifting
Attachment Loads	Lifting
Bolt Loads	Preload for 100 psi metallic seal compression

NOTES:

(1) Cask designed for 100 psi internal pressure which envelops all internal pressure effects.
(2) For normal conditions, the fission gas release should be less than 10%. However, for analysis purposes, 100% release is assumed.
(3) Cask designed for 25 psi external pressure which envelops all external pressure effects.

TABLE 3.2-7
LEVEL A SERVICE LOADS (TN-40 CASK)

Applied Load	Loading Condition (Cause)	
Internal Pressure (1)	Rated Design	20 psig
Operating (2)	Gas Release	15psig
External Pressure (3)	Flood/burial	25psig
Distributed Loads	Weight	Table 3.2-1
	Cask body	Table 3.2-1
	Contents	Table 3.2-1
	Snow	Nil
	Ice	Nil
	Wind (Tornado)	Nil
	Lifting	Table 3.2-1
Attachment Loads	Lifting	Table 4.2-3
Bolt Loads	Preload for 100 psi and metallic seal compression	25 ksi
Thermal Effects	Decay Heat/liner materials	300 Deg F
	Solar Insolation	108 Btu/hr-ft^2
	Cold Rain on Hot Cask	Nil

NOTES:

(1) Cask rated at 20 psig normal operating pressure and designed for 100 psig internal pressure which envelops all internal pressure effects.
(2) For normal conditions, the fission gas release should be less than 10%.
(3) Cask designed for 25 psig external pressure which envelops all external pressure effects.

TABLE 3.2-8
LEVEL D SERVICE LOADS (TN-40 CASK)

Load	Cause
Internal Pressure	(1) and (2)
External Pressure	(3)
Distributed Loads	Weight
	Cask body
	Contents
	Tornado Wind
	Flood Water
	Seismic
Local Loads	Tornado Wind Driven Missiles
Bolt Loads	Preload for 100 psig and metallic seal compression

NOTES:

(1) Cask design rating is 20 psig normal operating pressure and the cask is analyzed for 100 psig internal pressure which envelops all internal pressure effects.

(2) The fission gas release should be less than 10%. However, for analysis purposes, 100% release is assumed.

(3) Cask designed for 25 psig external pressure which envelops all external pressure effects including flood water level, cask burial and explosion. Explosions close to the cask are unexpected. Explosions at a significant distance from the cask would have a negligible effect.

TABLE 3.2-9
LOAD COMBINATIONS FOR TN-40 CASK BODY

	Bolt Preload	1 g Down	Internal Pressure	Ext. Pressure	Temp.	3 G On Trunnions	Trunnion Local	Seismic, Tornado or Flood	Tornado Missiles
Design Des (1)	X	X	X					X	
Des (2)	X		X			X	X		
Des (3)	X	X		X				X	
Level A A (1)	X	X	X		X			X	
A (2)	X		X		X	X	X		
A (3)	X	X		X	X			X	
Level D D (1)	X	X	X					X	X
D (2)	X	X		X				X	X

TABLE 3.3-1
SUMMARY OF THERMAL ANALYSES (TN-40 CASK)

<u>Short Term (OFF-NORMAL) Storage Conditions*</u>

Maximum Temperatures		
Outer Surface	223°F	(112°C)
Neutron Shield (resin/aluminum)	274°F	(134°C)
Seal/Lid	242°F	(117°C)
Cavity Wall	303°F	(151°C)
Basket Plate	537°F	(281°C)
Fuel Cladding	642°F	(339°C)
Average Cavity Gas Temperature	442°F	(228°C)

<u>Long Term (NORMAL) Conditions **</u>

Maximum Temperatures		
Outer Surface	191°F	(88°C)
Neutron Shield (resin/aluminum)	233°F	(111°C)
Seal/Lid	200°F	(93°C)
Cavity Wall	261°F	(127°C)
Basket Plate	496°F	(258°C)
Fuel Cladding	602°F	(317°C)
Average Cavity Gas Temperature	401°F	(205°C)

*100°F (38°C) ambient temperature, solar heat load of 135 btu/hr-ft^2.

**50°F (10°C) ambient temperature, solar heat load of 108 Btu/hr-ft^2.

TABLE 3.3-2
PR0PERTIES OF MATERIALS USED IN THERMAL ANALYSES (TN-40 CASK)

Material	Stainless Steel 18Cr-8Ni	Carbon Steel C-Mn-Si	Aluminum 6061 Alloy	Resin	Helium	Air	Concrete
Used for	Basket	Cask body shells and outer shells	Basket and neutron shield	Neutron shield	Fill gas	air gaps	storage pad
Density (lb/in.^{3})	0.29 (24)*	0.284 (24)	0.98 (24)	0.057 (25)	0	0	
Emissivity	0.15 (12)						0.94 (15)
Specific Heat (Btu/lb-^{0}F)	0.119@200°F (4) 0.123@400°F 0.130@600°F 0.132@700°F 0.136@1,000°F	0.110@100°F (4) 0.123@300°F 0.133@500°F 0.143@700°F 0.155@900°F 0.164@1000°F	0.215@100°F (4) 0.221@200°F 0.226@300°F 0.230@400°F	0.3107 (25)	0	0	
Thermal Conductivity (Btu/hr-ft-°F)	8.6@70°F (4) 8.7@100°F 9.3@200°F 9.8@300°F 10.4@400°F 10.9@500°F	23.6@70°F (4) 23.9@100°F 24.2@150°F 24.4@200°F 24.4@300°F 24.2@400°F 23.7@500°F 23.1@600°F 21.7@800°F 20.0@1,000°F	96.1@70°F (4) 96.9@100°F 98.0@150°F 99.0@200°F 100.6@300°F 101.9@400°F	0.1(17)	0.078@0°F(12) 0.098@200°F 0.114@400°F 0.130@600°F 0.145@800°F 0.159@981°F 0.172@116°F	0.0137@32°F (12) 0.0145@68°F 0.0153@104°F 0.0161@140°F 0.0169@176°F 0.0177@212°F 0.0214@392°F 0.0248@572°F 0.0280@752°F	1.0 (26)

*Numbers in parentheses are the reference numbers.

TABLE 3.3-3
MATERIAL COMPOSITION FOR KENO MODEL (TN-40 CASK)

MIXTURE	DENSITY (g/cm^3)	NUCLIDE/ ELEMENT	LIBRARY NUMBER	DENSITY (Atoms/b.cm)
UO_2 Fuel	10.39	U235	92235	9.0350E-4
3.85 wt% U235		U238	92238	2.2279E-2
		O	8016	4.6365E-2
UO_2 Fuel	10.39	U235	92235	4.4590E-4
1.9 wt% U235		U238	92238	2.2731E-2
		O	8016	4.6355E-2
Zircaloy	6.44	*	40302	4.2518E-2
Water	0.998	H	1001	6.6759E-2
		O	8016	3.3380E-2
Stainless Steel	7.92	Cr	24304	1.7430E-2
		Mn	25055	1.7364E-3
		Fe	26304	5.9359E-2
		Ni	28304	7.7182E-3
Aluminum	2.699	Al	13027	6.0242E-2
Carbon Steel	7.82	C	6012	3.9217E-3
		Fe	26000	8.3500E-2
Boral Core	2.63	B10	5010	9.4855E-3
		B11	5011	3.8518E-2
		C	6012	1.2001E-2
		Al	13027	3.4804E-2
Borated Water	1	B10	5010	1.9815E-5
		B11	5011	8.0445E-5
		H	1001	6.6732E-2
		O	8016	3.3366E-2

*40302 is a composite cross section for 97.91 wt% Zr,
1.59 Sn, 0.5%Fe

TABLE 3.3-4
TN-40 REACTIVITY DURING DRAINING

Borated Water Level below top of active fuel (ft)	$k_{eff} \pm \sigma$ *
0	0.8966 ± 0.0031
1	0.8899 ± 0.0035
2	0.9011 ± 0.0036
3	0.8936 ± 0.0031
4	0.8956 ± 0.0031
5	0.8879 ± 0.0033
6	0.8816 ± 0.0034

*Uncorrected for bias

TABLE 3.3-5
TN-40 REACTIVITY VERSUS WATER DENSITY

Fresh water density (g/cc)	$K_{eff} \pm \sigma$*
1.0	0.9306 ± 0.0036
0.8	0.8902 ± 0.0044
0.5	0.7709 ± 0.0036
0.1	0.4253 ± 0.0028
0.01	0.3137 ± 0.0032
0.0001	0.3137 ± 0.0022

*Uncorrected for bias – 1.9 w/o U235

TABLE 3.3-6
PNL BENCHMARK EXPERIMENTS

Case No.	Length x Width (Fuel rods)	Poison Plates (mm)	Fuel Cluster Separation (mm)
A*	(1) 25 x 18 (2) 20 x 18	Borated SS (2.98)	48.0
B*	(2) 20 x 18	Boral (2.92)	26.9
C*	(2) 20 x 18	SS304L (3.02)	82.8
D*	(1) 25 x 18 (2) 20 x 18	None	95.1
E*	20 x 15	None	63.9

*17.85 cm thick steel walls placed at 1.32 cm from fuel clusters.

TABLE 3.3-7
KENO-V.A BENCHMARK RESULTS

Experiment No.	Keff	Sigma
A	0.9900	0.004
B	0.9909	0.004
C	0.9865	0.005
D	0.9885	0.004
E	0.9856	0.004

Avg=0.9883

TABLE 3.3-8
MAXIMUM TRANSIENT TEMPERATURES – FIRE ACCIDENT

Location	Initial Temperature	Peak Transient Temperature
Outer Shell	233°F	1080°F @ 15 min.*
Neutron Shield	273°F	820°F @ 17 min.
Cavity Wall	302°F	354°F @ 201 min.
Basket	530°F	568°F @ 495 min.

*Time from start of fire accident

TABLE 3.4-1
DESIGN CRITERIA FOR TN-40 CASKS

Maximum gross weight on crane (with lift beams, without water)	125 tons
Maximum cask height with lid removed	16 ft. 1 in.
Minimum design life	25 years
Maximum keff. including bias and uncertainties	$\leq$ 0.95 Normal <0.98 Accident
Payload capacity fuel assemblies	40 intact PWR 14x14
Maximum external dose rate (on storage pad)	200 mrem/hr contact
Spent fuel characteristics	
a) Initial enrichment	3.85%
b) Burnup (max)	45,000 MWD/MTU
c) Cooling time (min)	10 years
d) Decay heat	27 kw (total)
Max clad temperature	340^0C
Cask cavity atmosphere	Helium gas
Maximum internal pressure	100psig
Ambient temperature (Min-Max)	-40^0 to 120^0F
Maximum solar heat load	135 BTU/hr-ft^2
Tornado wind	300mph rotational 60 mph translational
Tornado missiles	4"x12"x144" plank @300 mph; 4,000lb. auto @50 mph
Seismic design earthquake	0.12 g horizontal 0.08 g vertical
Snow and ice	50 psf load

TABLE 3.4-2
DESIGN BASIS LIMITS FOR FISSION PRODUCT BARRIERS FOR 72.48 REVIEWS

Fission Product Barrier	Design Basis Parameter	Transnuclear TN-40 Design Basis Limit
Fuel Cladding	Clad Temperature	Less than 340°
Fuel Cladding	Decay Heat Per Assembly	27 kW (total)
Fuel Cladding	Sub-criticality	K_{eff} less than 0.95
Confinement Boundary	MSB/DSC Pressure	100 psig
Confinement Boundary	MSB/DSC Vessel Stresses	Code allowable stresses for ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, Article NB-3200
Confinement Boundary	MSB/DSC Leak Rate	1 x 10^{-5} atm cm^3/sec

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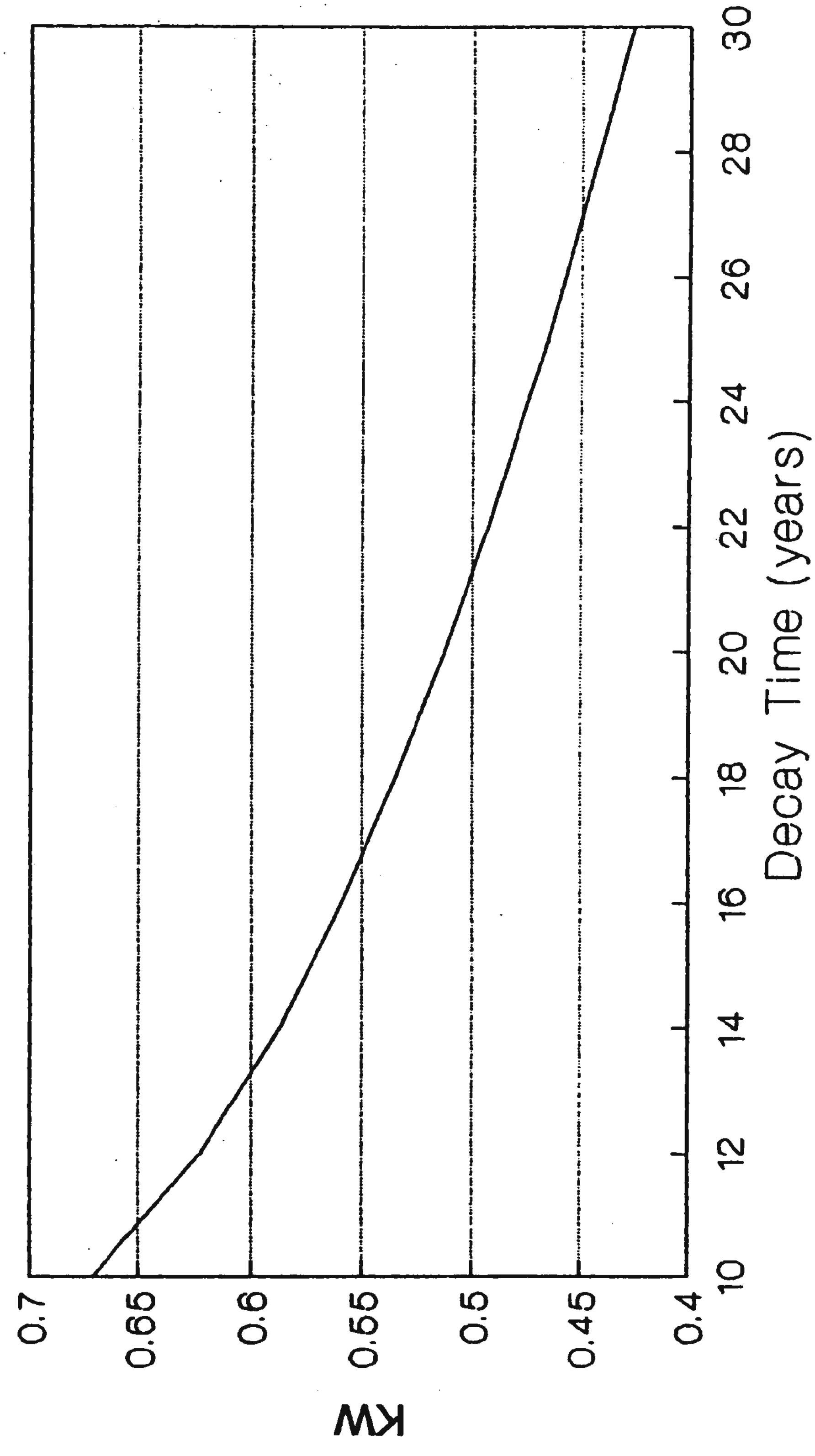

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Figure 3.1-1
DECAY HEAT
DESIGN BASIS FUEL ASSEMBLY

PRAIRIE ISLAND ISFSI
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DECAY HEAT DESIGN BASIS FUEL ASSEMBLY

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG3.1-1_REV_0
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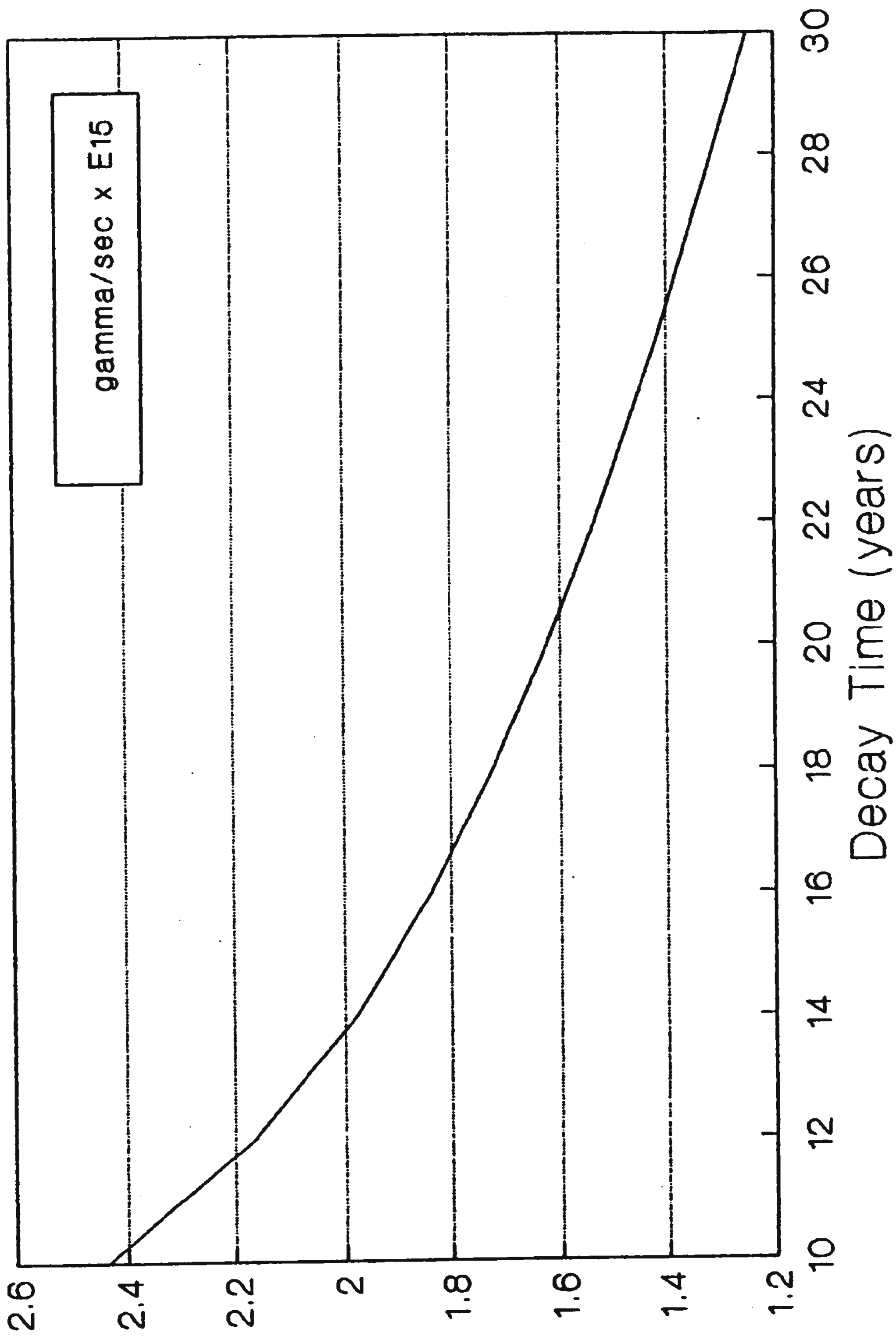

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Figure 3.1-2
GAMMA SOURCE
DESIGN BASIS FUEL ASSEMBLY

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GAMMA SOURCE DESIGN BASIS FUEL ASSEMBLY

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG3.1-2_REV_0
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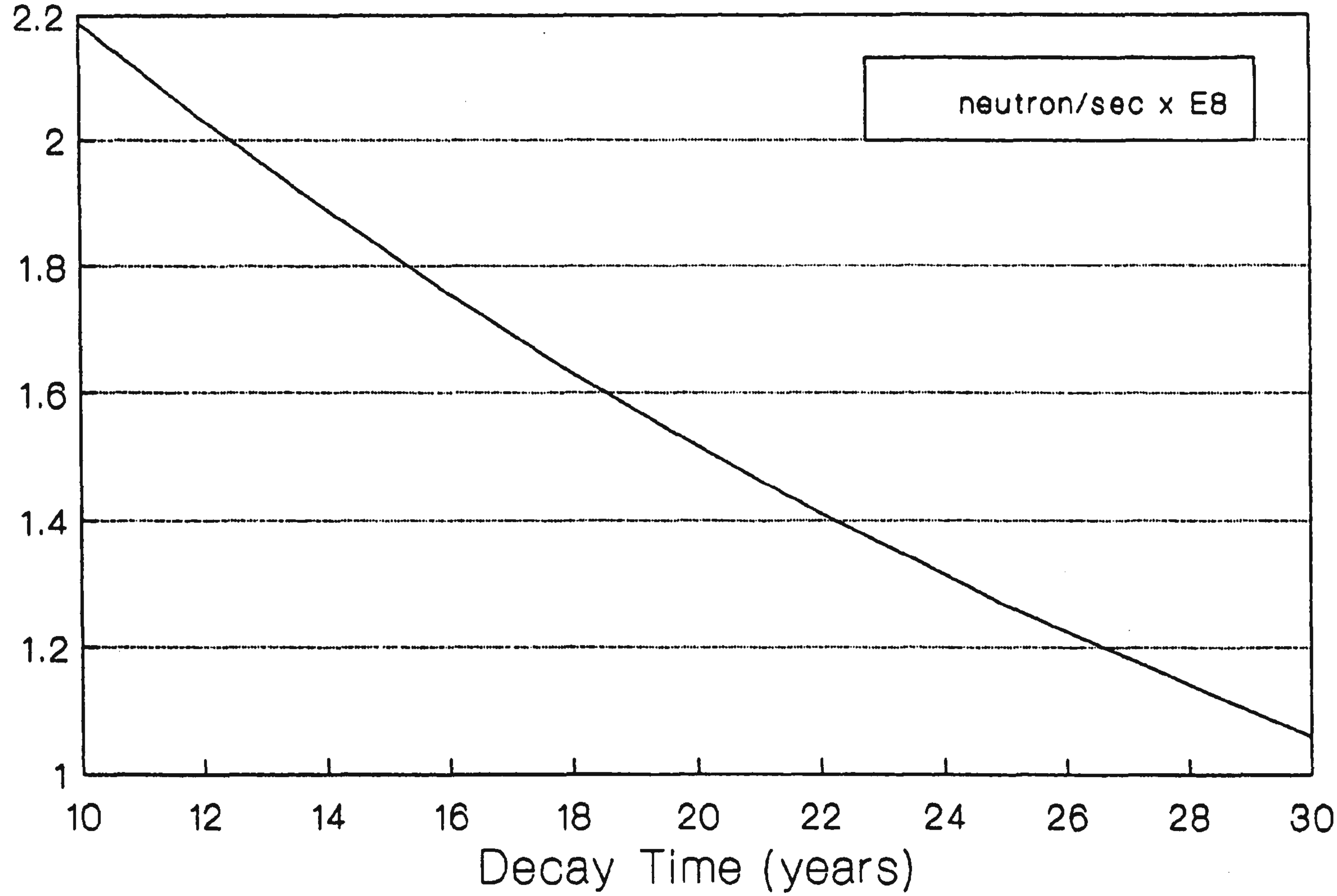

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Figure 3.1-3

NEUTRON SOURCE DESIGN BASIS FUEL ASSEMBLY

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NEUTRON SOURCE DESIGN BASIS FUEL ASSEMBLY

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG3.1-3_REV_0
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Figure 3.1-4
OFA FUEL ASSEMBLY DIMENSIONAL DATA
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SAFETY ANALYSIS REPORT

OFA FUEL ASSEMBLY DIMENSIONAL DATA			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 2	FIG3.1-4_REV_2
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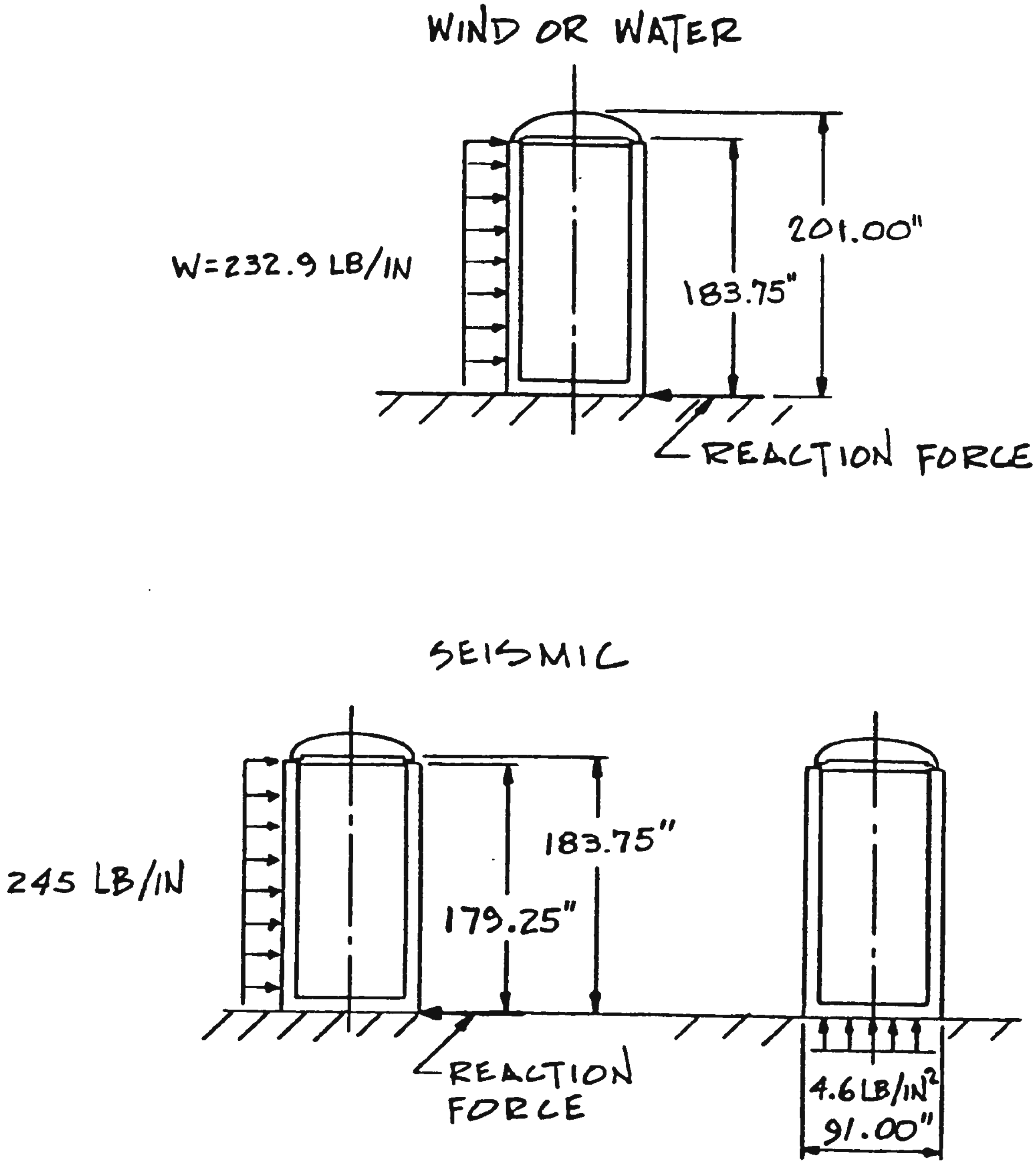

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Figure 3.2-1

EARTHQUAKE, WIND AND WATER LOADS (TN-40 CASK)

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

EARTHQUAKE, WIND AND WATER LOADS (TN-40 CASK)

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG3.2-1_REV_0
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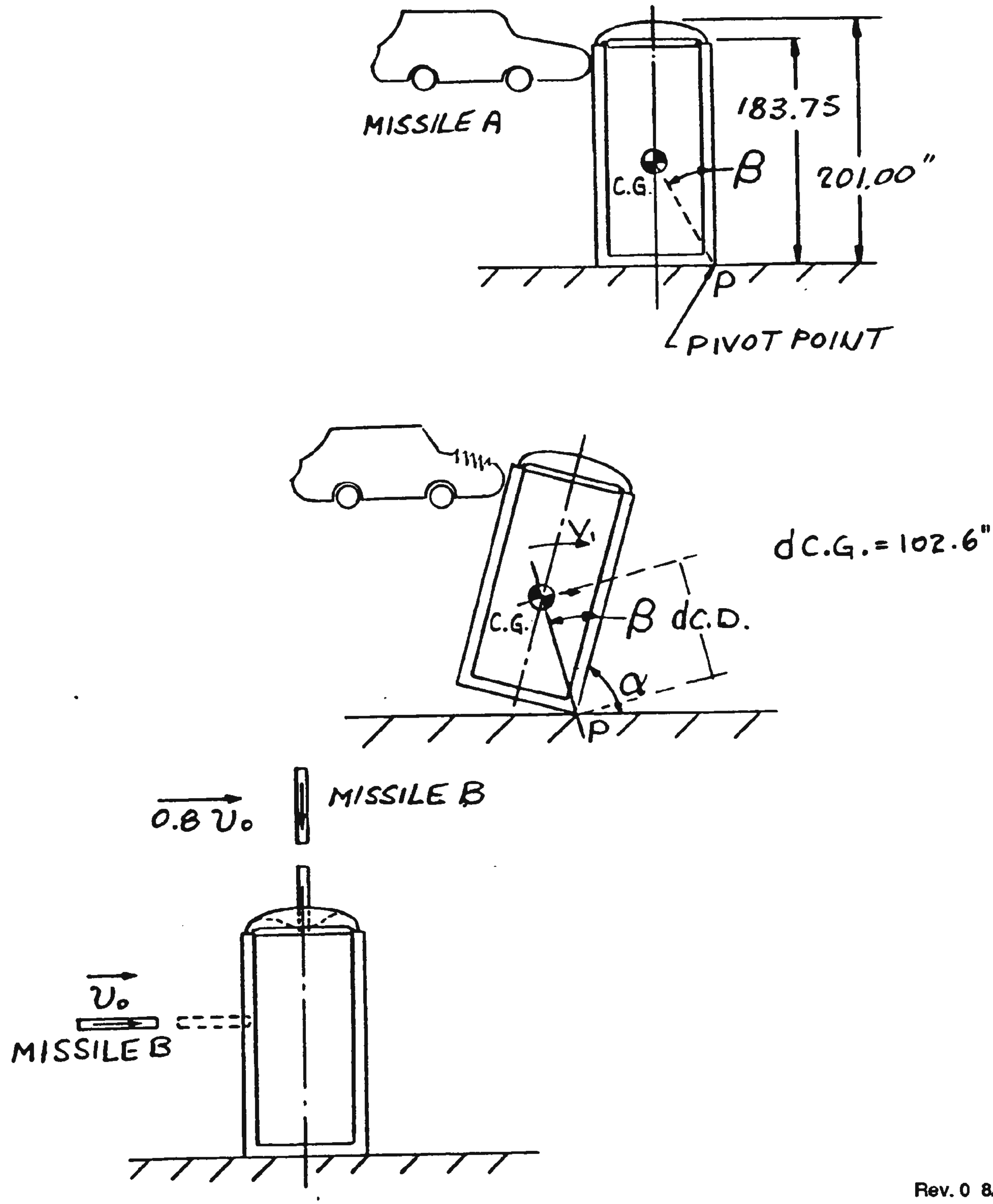

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Figure 3.2-2

TORNADO MISSILE IMPACT LOADS (TN-40 CASK)

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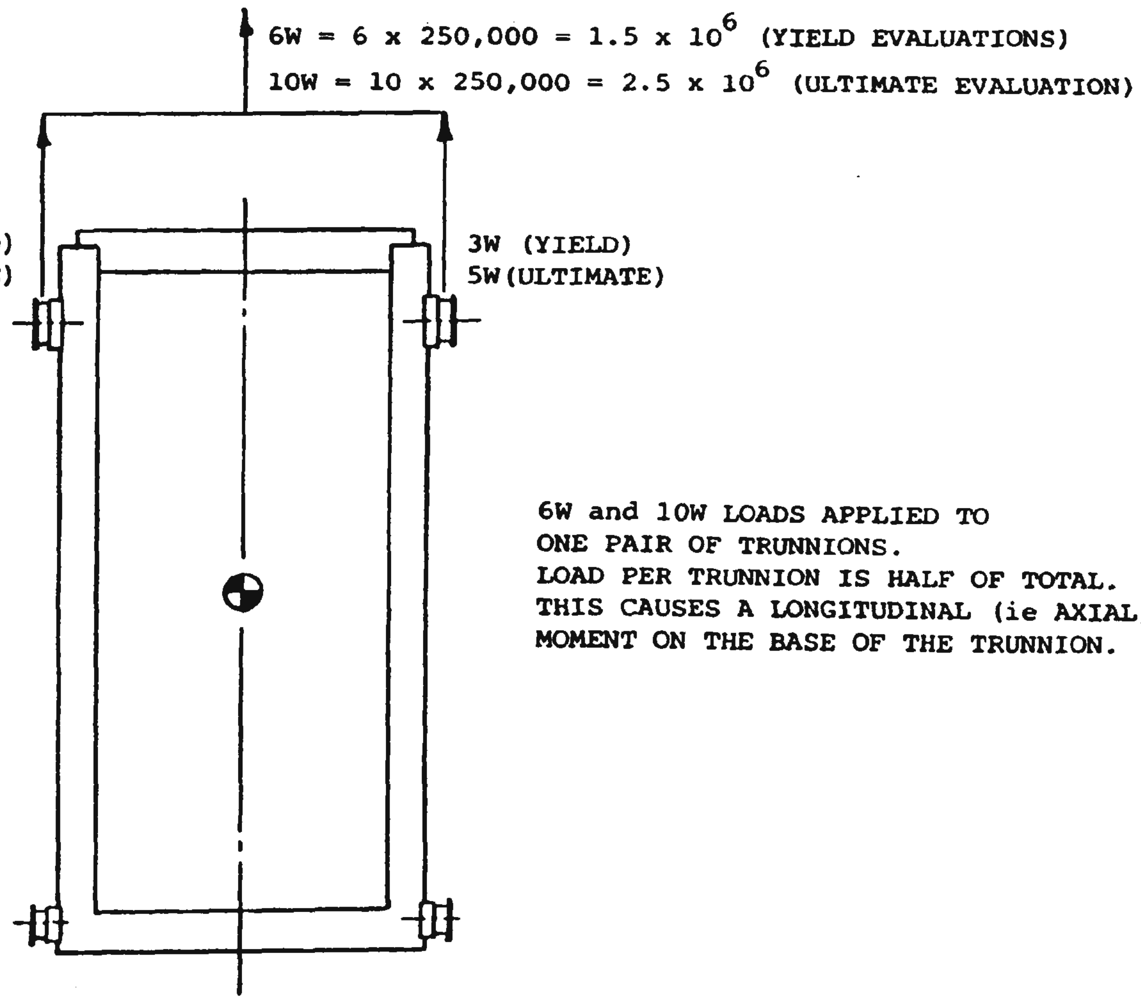

6W and 10W LOADS APPLIED TO ONE PAIR OF TRUNNIONS.
LOAD PER TRUNNION IS HALF OF TOTAL.
THIS CAUSES A LONGITUDINAL (ie AXIAL) MOMENT ON THE BASE OF THE TRUNNION.

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Figure 3.2-3

LIFTING LOADS (TN-40 CASK)

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SAFETY ANALYSIS REPORT

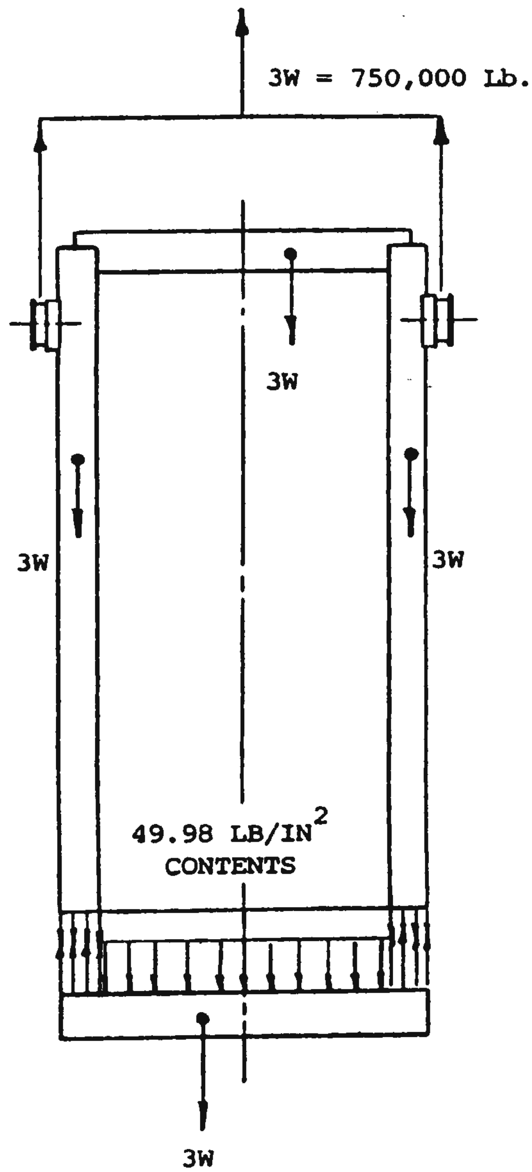

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Figure 3.2-4

DESIGN LOADS ON TN-40 CASK DUE TO LIFTING

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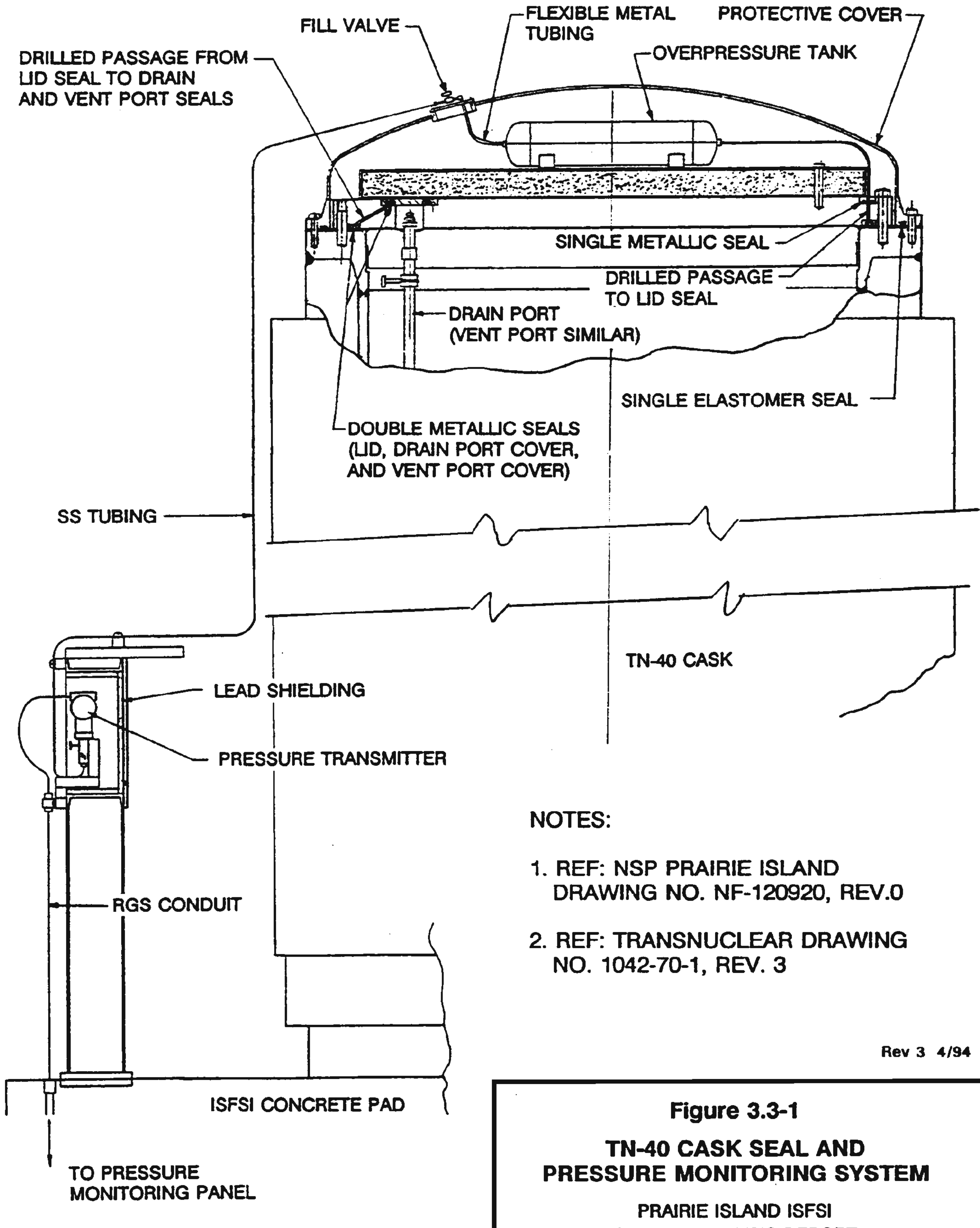
SAFETY ANALYSIS REPORT

DESIGN LOADS ON TN-40 CASK DUE TO LIFTING

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG3.2-4_REV_0
	PAGE. NO.	DATE: 04-3-06	



NOTES:

1. REF: NSP PRAIRIE ISLAND DRAWING NO. NF-120920, REV.0
2. REF: TRANSNUCLEAR DRAWING NO. 1042-70-1, REV. 3

Rev 3 4/94

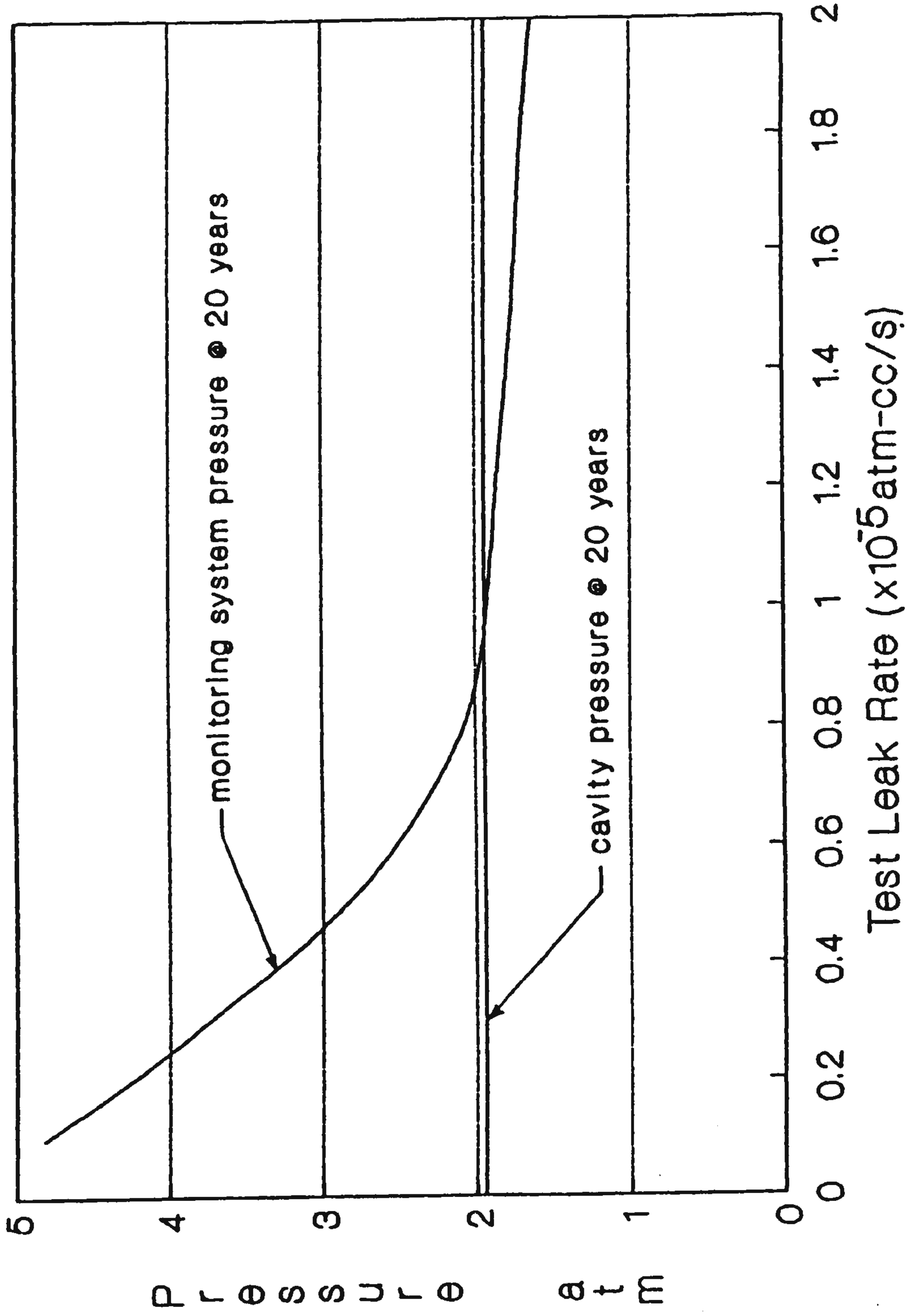
Figure 3.3-1

TN-40 CASK SEAL AND PRESSURE MONITORING SYSTEM

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

TN-40 CASK SEAL AND PRESSURE MONITORING SYSTEM

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG3.3-1_REV_3
	PAGE. NO.	DATE: 04-3-06	

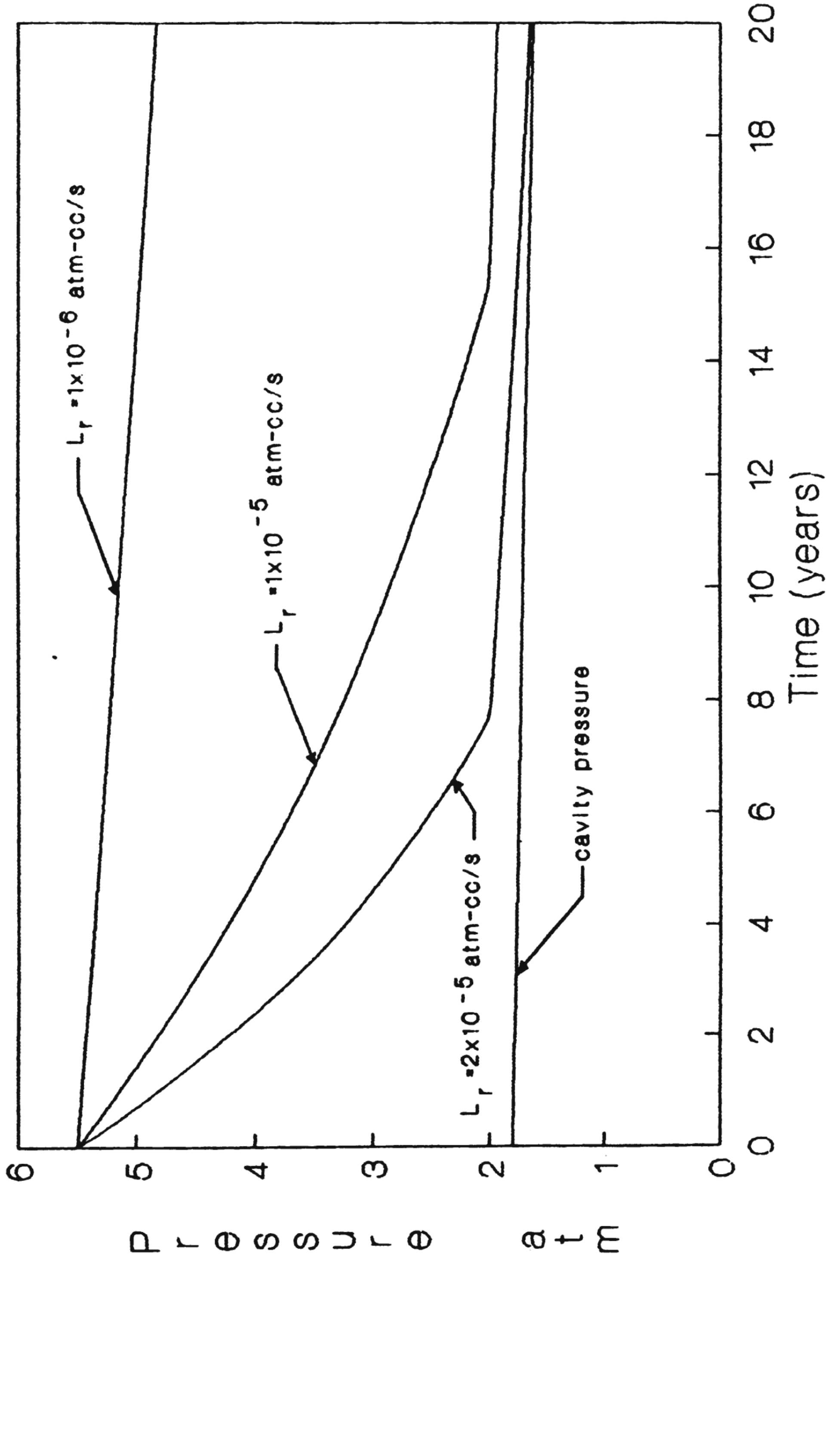


Rev. 2 9/91

Figure 3.3-2
TN-40 CASK PRESSURE MONITORING SYSTEM TEST LEAKAGE RATE
PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

TN-40 CASK PRESSURE MONITORING SYSTEM TEST LEAKAGE RATE

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: **VLS**	REVISION: 2	FIG3.3-2_REV_2
	PAGE. NO.	DATE: 04-3-06	



Rev. 0 8/90

Figure 3.3-3
TN-40 CASK PRESSURE MONITORING SYSTEM - SYSTEM PRESSURE
PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

TN-40 CASK RRESSURE MONITORING SYSTEM - SYSTEM PRESSURE

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG3.3-3_REV_0
	PAGE. NO.	DATE: 04-3-06	

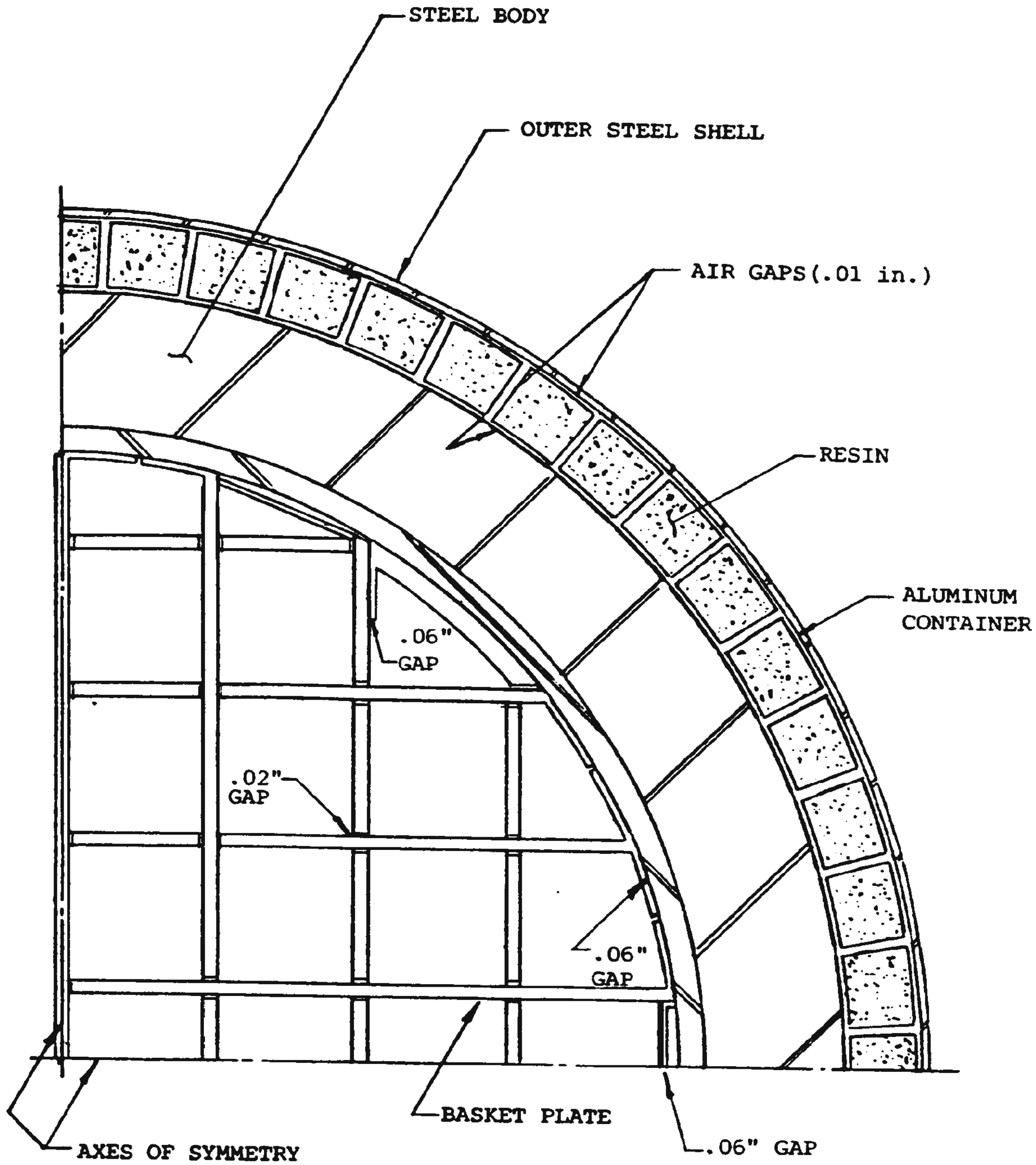


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Figure 3.3-4

TN-40 CASK THERMAL MODEL RADIAL CROSS SECTION

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

Figure Withheld Under 10 CFR 2.390

Figure 3.3-5

TN-40 CASK THERMAL MODEL AXIAL CROSS SECTION

PRAIRIE ISLAND ISFSI SAFETY ANALYSIS REPORT

TN-40 CASK THERMAL MODEL AXIAL CROSS SECTION			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG3.3-5_REV_0
	PAGE. NO.	DATE: 04-3-06	

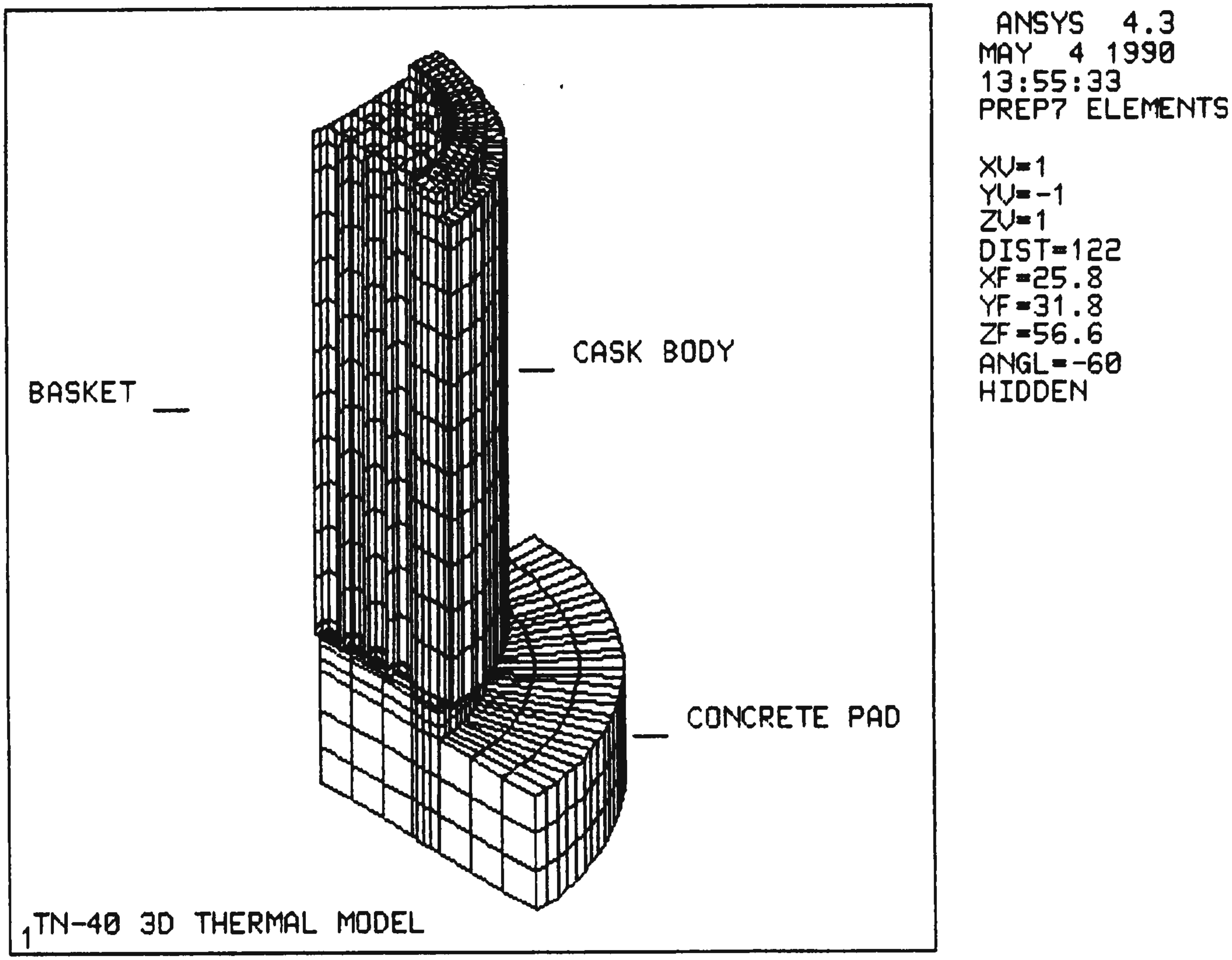

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Figure 3.3-6

TN-40 CASK FINITE ELEMENT THERMAL MODEL

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

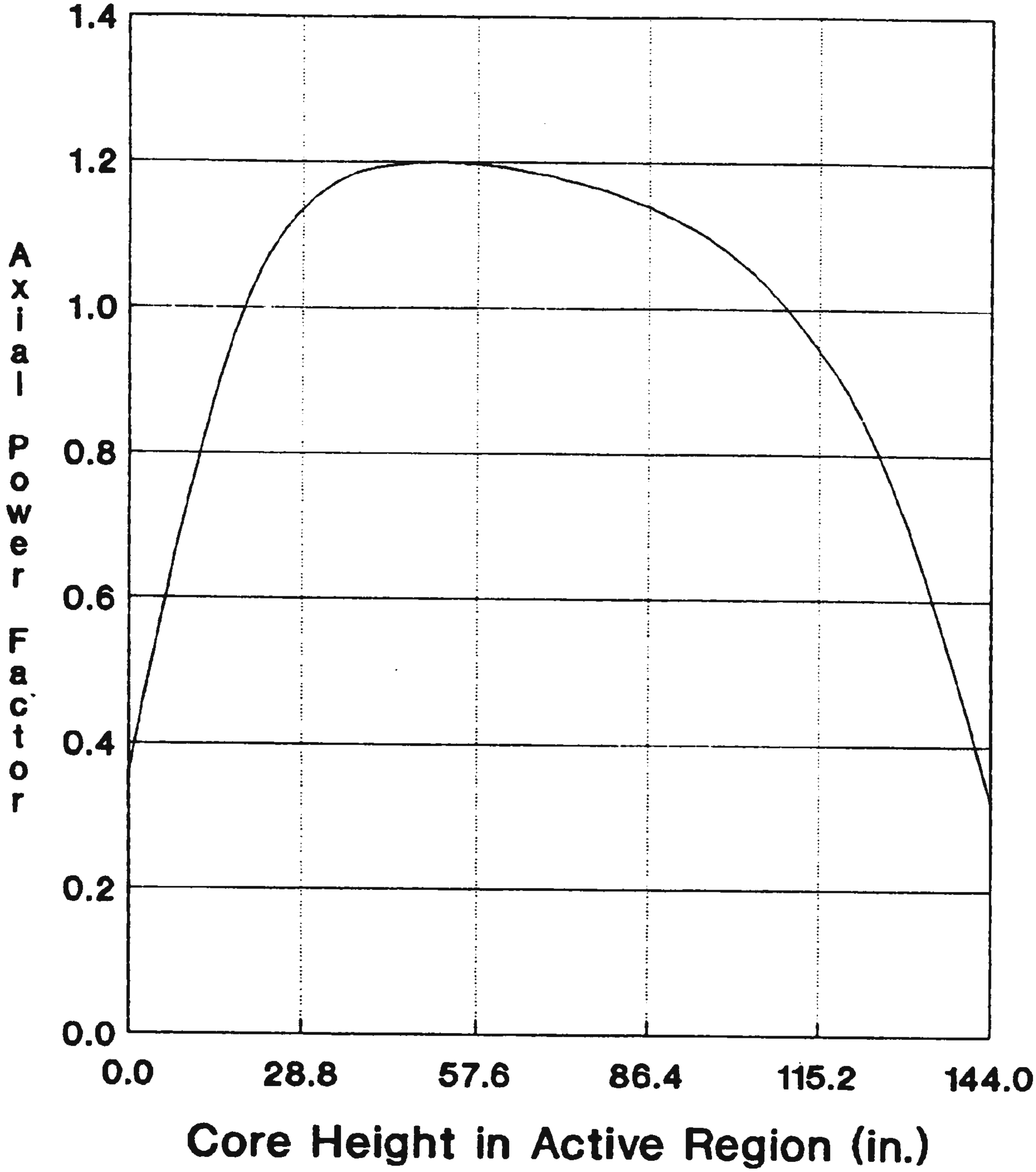

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Figure 3.3-7

AXIAL POWER DISTRIBUTION (TN-40 CASK)

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

AXIAL POWER DISTRIBUTION (TN-40 CASK)

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG3.3-7_REV_0
	PAGE. NO.	DATE: 04-3-06	

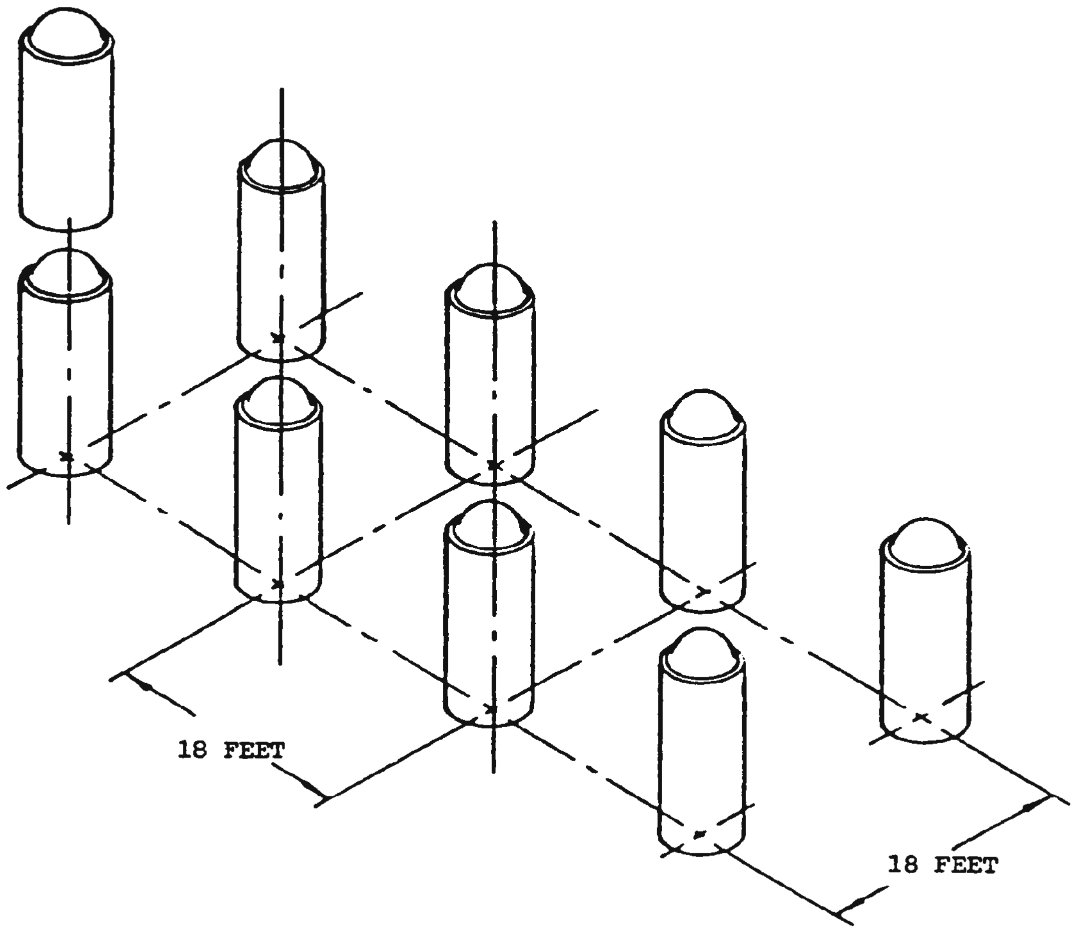

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Figure 3.3-8

TN-40 CASK STORAGE CONFIGURATION

PRAIRIE ISLAND ISFSI SAFETY ANALYSIS REPORT

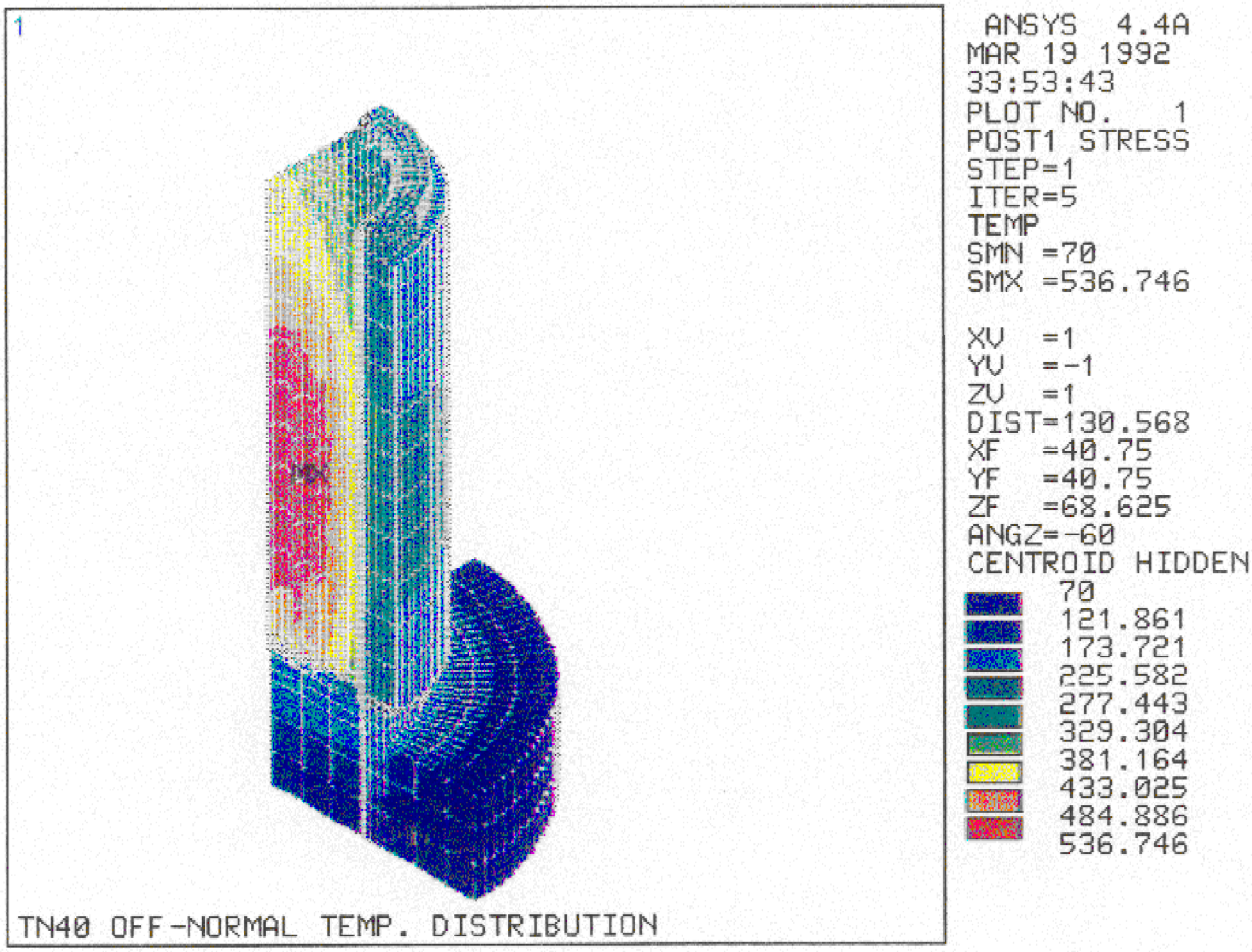

Figure 3.3-9

TN-40 CASK TEMPERATURE DISTRIBUTION OFF NORMAL CONDITIONS

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

TN-40 CASK TEMPERATURE DISTRIBUTION OFF NORMAL CONDITIONS

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG3.3-9_REV_3
	PAGE. NO.	DATE: 03-28-06	

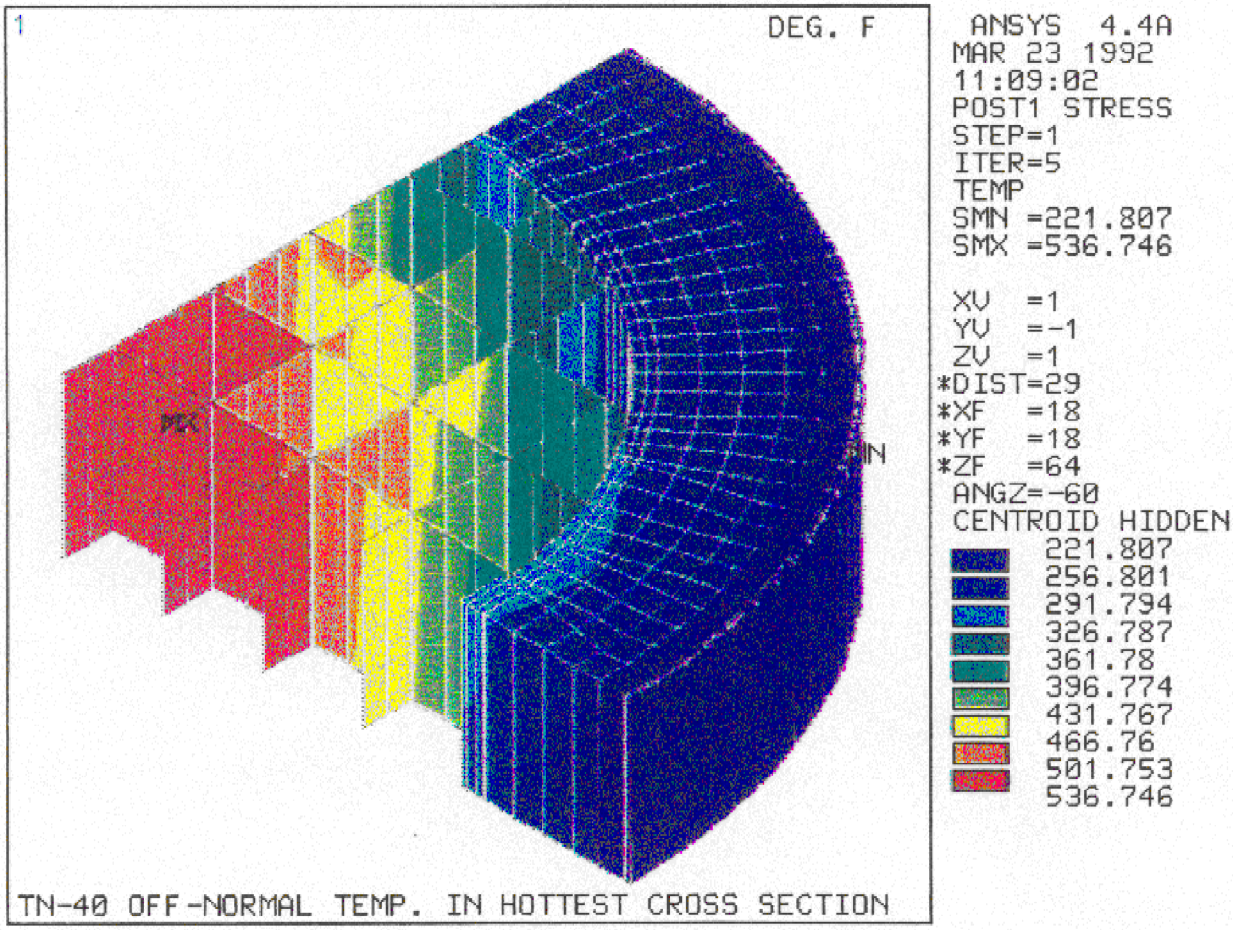

Figure 3.3-10

TN-40 CASK TEMPERATURE DISTRIBUTION - HOTTEST CROSS-SECTION

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

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TN-40 CASK TEMPERATURE DISTRIBUTION - HOTTEST CROSS - SECTION

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG3.3-10_REV_3
	PAGE. NO.	DATE: 04-3-06	

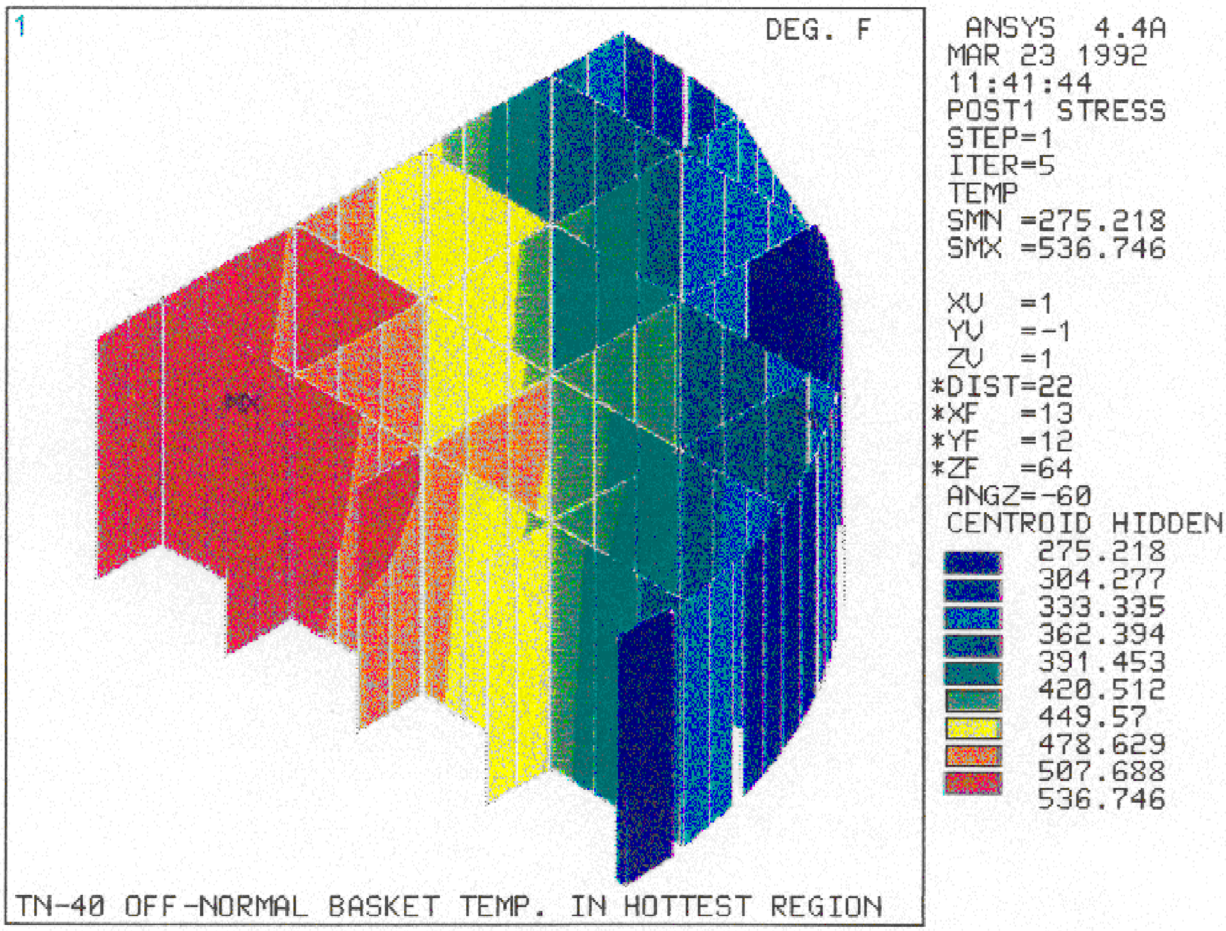

Figure 3.3-11

TN-40 BASKET SECTION TEMPERATURE DISTRIBUTION - HOTTEST REGION

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

TN-40 BASKET SECTION TEMPERATURE DISTRIBUTION - HOTTEST REGION

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG3.3-11_REV_3
	PAGE. NO.	DATE: 04-3-06	

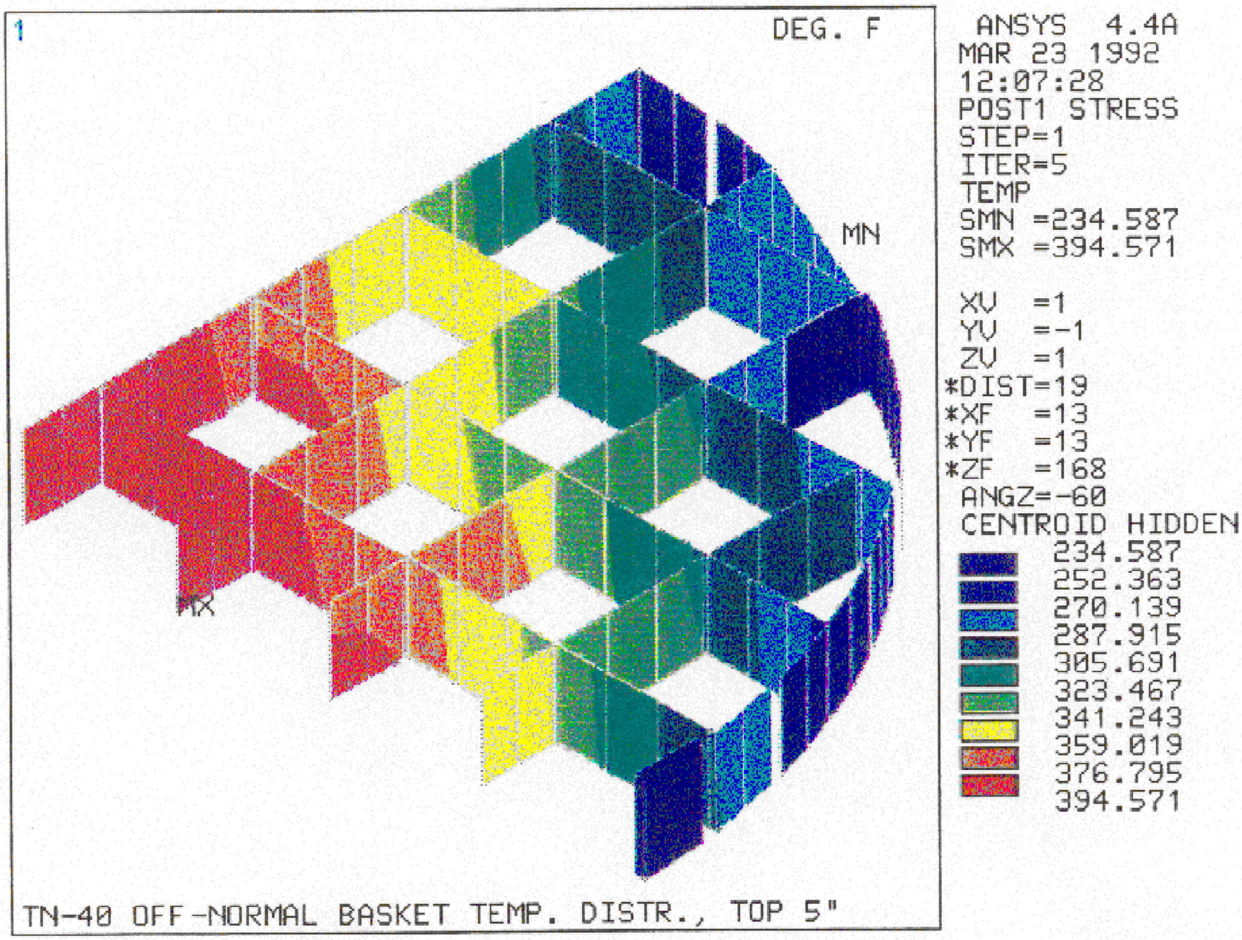

Figure 3.3-12

TN-40 BASKET TEMPERATURE DISTRIBUTION-TOP 5 INCHES

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

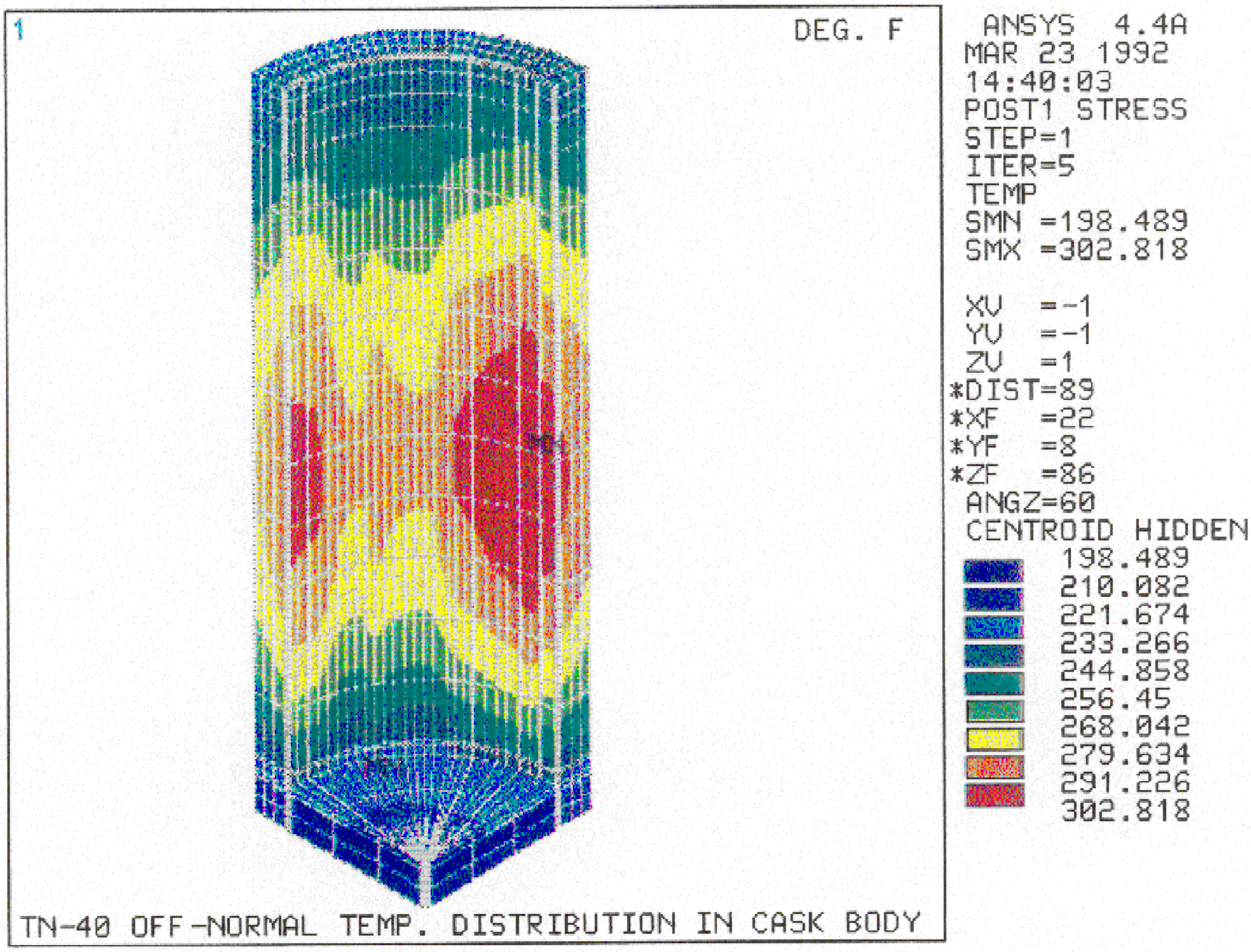

Figure 3.3-13

TN-40 CASK BODY TEMPERATURE DISTRIBUTION

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

TN-40 CASK BODY TEMPERATURE DISTRIBUTION

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG3.3-13_REV_3
	PAGE. NO.	DATE: 04-3-06	

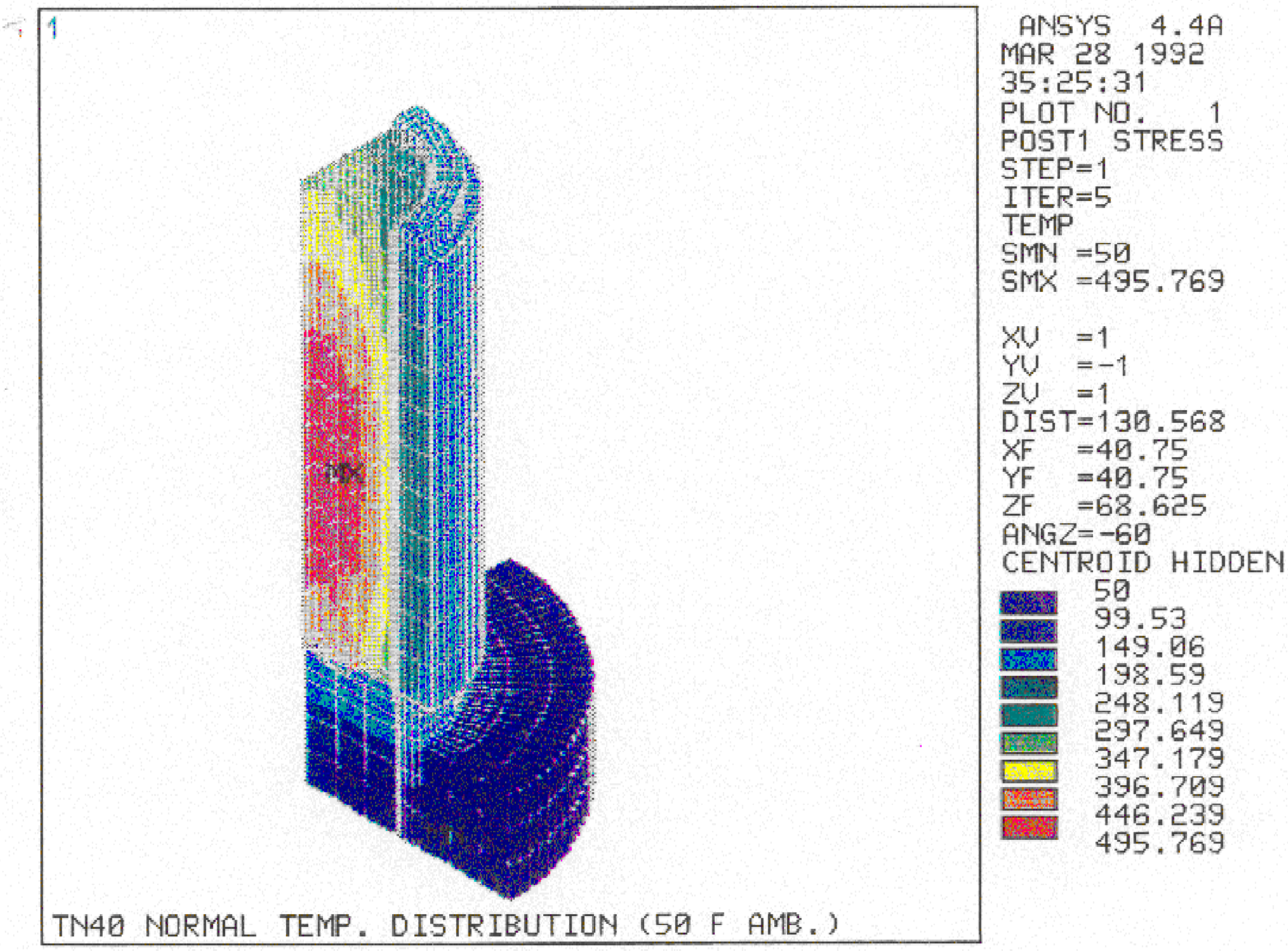

Figure 3.3-14

TN-40 CASK TEMPERATURE DISTRIBUTION NORMAL CONDITIONS

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

TN-40 CASK BODY TEMPERATURE DISTRIBUTION NORMAL CONDITIONS

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG3.3-14_REV_3
	PAGE. NO.	DATE: 04-3-06	

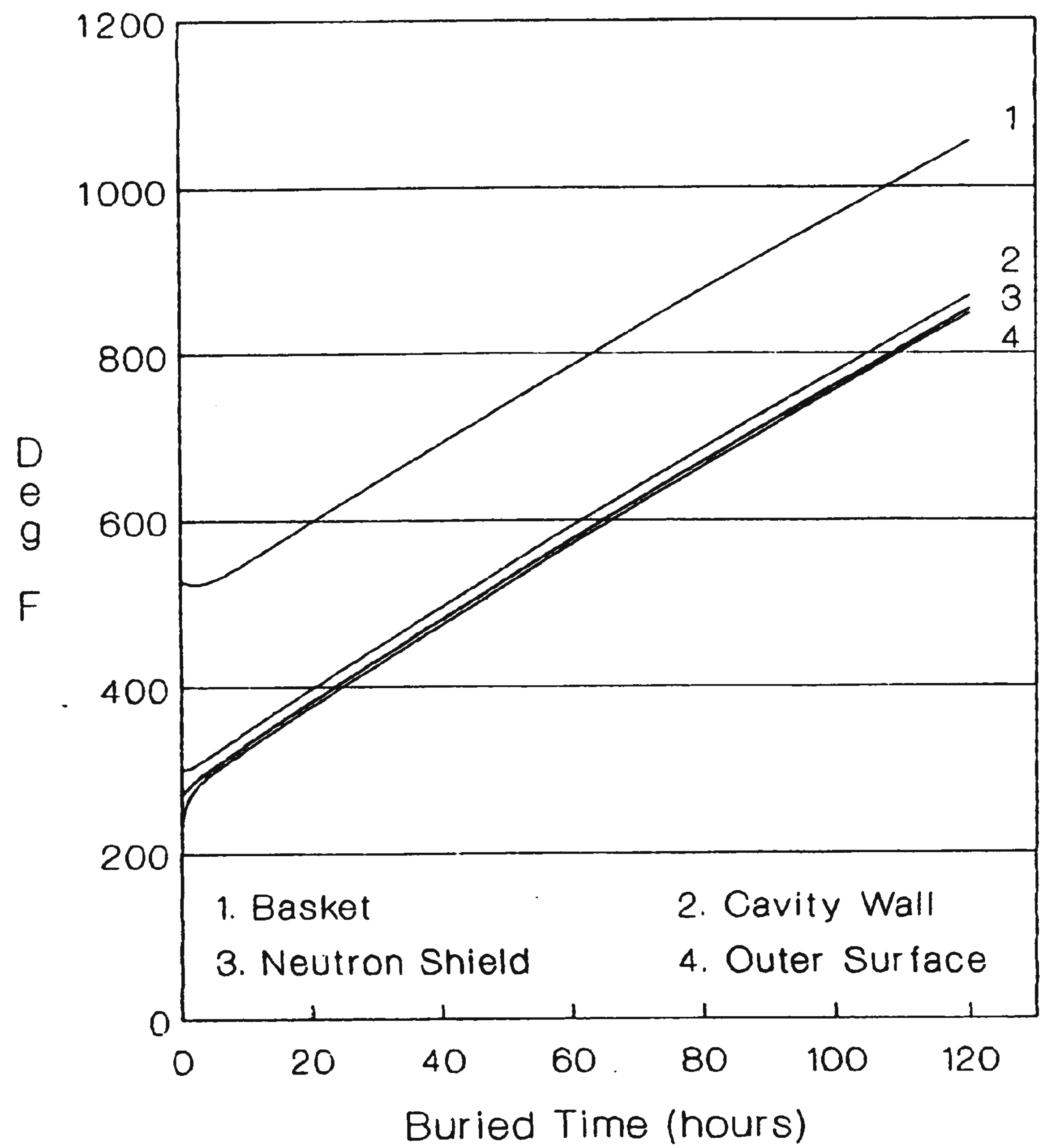

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Figure 3.3-15

BURIED CASK TEMPERATURE RESPONSE (TN-40 CASK)

PRAIRIE ISLAND ISFSI SAFETY ANALYSIS REPORT

Figure 3.3-16
KENO V.A. FUEL ASSEMBLY MODEL
PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

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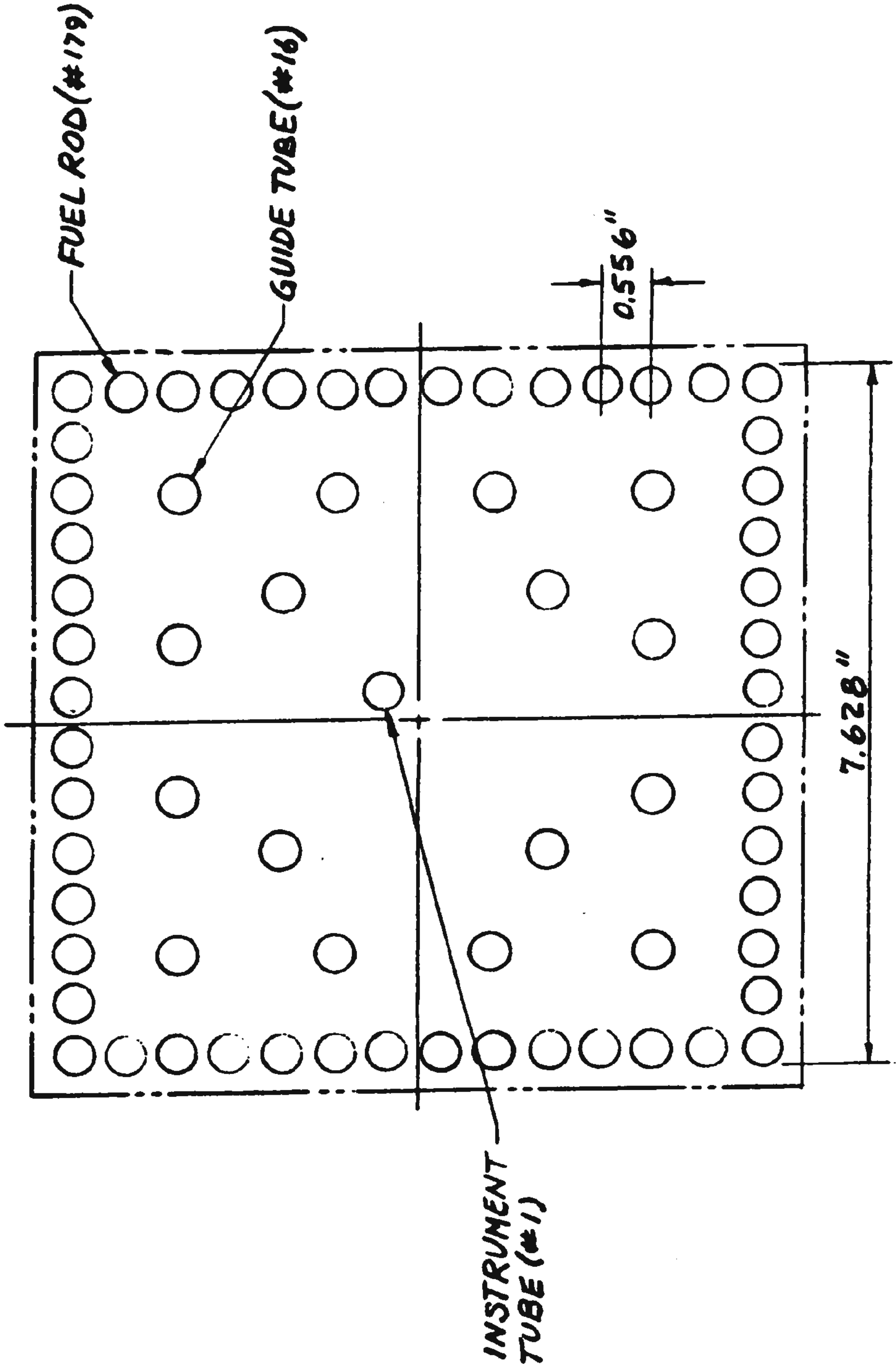

KENO V.A. FUEL ASSEMBLY MODEL

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG3.3-16_REV_0
	PAGE. NO.	DATE: 04-3-06	

CAD FILE: J / CAD / PRI / ISFSI _SAR / FIG3.3-16_REV_0

Figure Withheld Under 10 CFR 2.390

Figure 3.3-17

KENO V.A. CASK MODEL (TN-40 CASK)

PRAIRIE ISLAND ISFSI SAFETY ANALYSIS REPORT

KENO V.A. CASK MODEL (TN-40 CASK)			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG3.3-17_REV_3
	PAGE. NO.	DATE: 04-4-06	

Figure Withheld Under 10 CFR 2.390

Figure 3.3-18

BENCHMARK FUEL RODS

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

BENCHMARK FUEL RODS			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG3.3-18_REV_0
	PAGE. NO.	DATE: 04-4-06	

Figure Withheld Under 10 CFR 2.390

Figure 3.3-19

BENCHMARK EXPERIMENT

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

BENCHMARK EXPERIMENT			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG3.3-19_REV_0
	PAGE. NO.	DATE: 04-4-06	

CAD FILE: J / CAD / PRI / ISFSI _SAR / FIG3.3-19_REV_0

Figure Withheld Under 10 CFR 2.390

Figure 3.3-20

TN-40 LID SEAL THERMAL MODEL

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

TN-40 LID SEAL THERMAL MODEL			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG3.3-20_REV_3
	PAGE. NO.	DATE: 04-4-06	

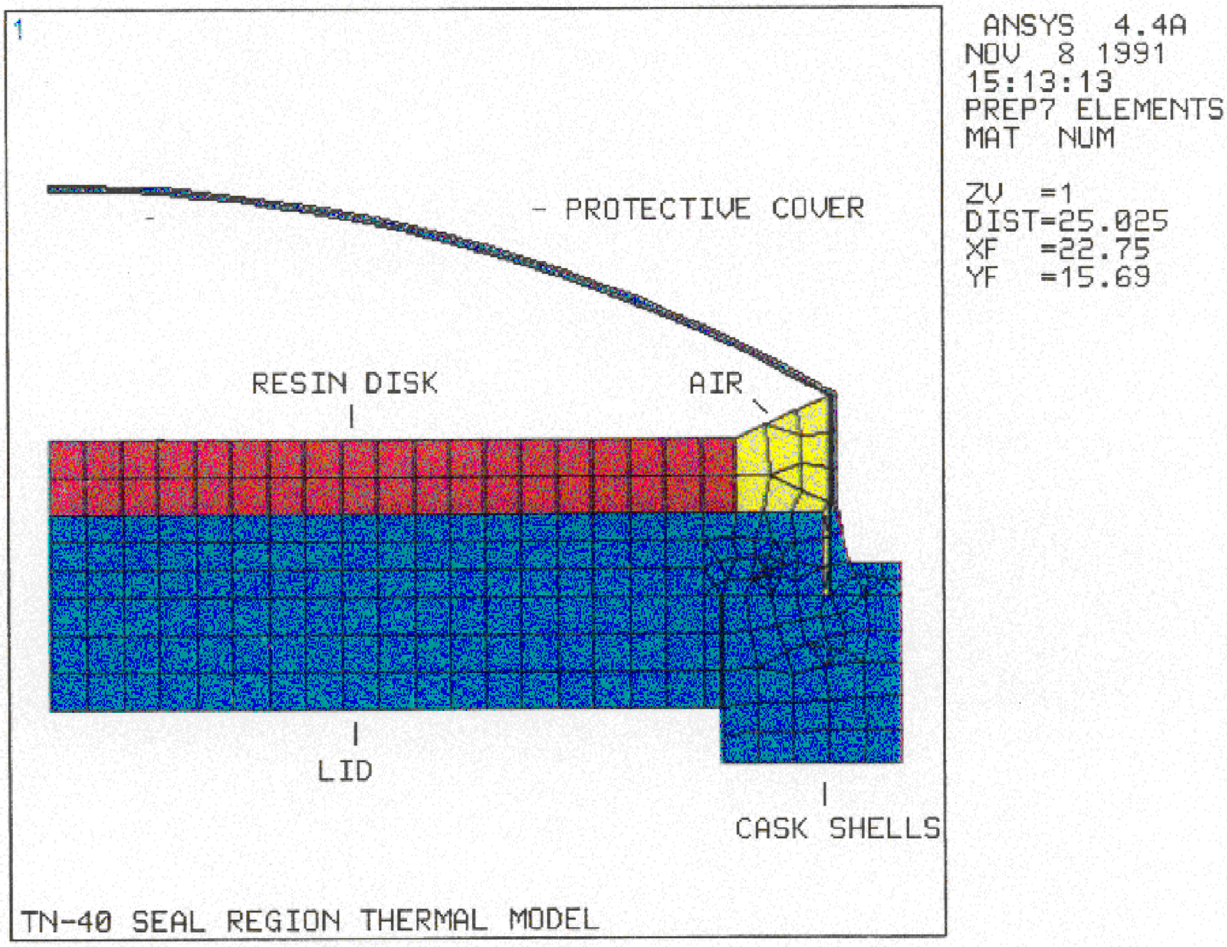

Figure 3.3-21

TN-40 LID SEAL REGION FINITE ELEMENT THERMAL MODEL

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

TN-40 LID SEAL REGION FINITE ELEMENT THERMAL MODEL

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG3.3-21_REV_3
	PAGE. NO.	DATE: 04-3-06	

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ISFSI SAR

APPENDIX 3A

TN-40 CASK
CRITICALITY EVALUATION
COMPUTER INPUT

PROPRIETARY

PROPRIETARY - TRADE SECRET INFORMATION WITHHELD PURSUANT TO 10 CFR 2.390

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PROPRIETARY - TRADE SECRET INFORMATION
WITHHELD PURSUANT TO 10 CFR 2.390

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PROPRIETARY - TRADE SECRET INFORMATION WITHHELD PURSUANT TO 10 CFR 2.390

SECTION 4

STORAGE SYSTEM

4.1 LOCATION AND LAYOUT

The location of the ISFSI site and the transport route of the casks from the Auxiliary Building to the site are shown in Figure 1.2-1. In addition, the map shows other facilities on the power plant site in the vicinity of the ISFSI site, such as roadways and railroad lines. The ISFSI and the entire transport route are within the Prairie Island Nuclear Generating Plant exclusion area.

The ISFSI, along with appurtenant facilities and equipment, is designed to interface with existing equipment and systems at the Prairie Island Nuclear Generating Plant. The Prairie Island USAR (Reference 1) provides a description of existing systems and equipment.

Roadways, buried pipes, and trenches have been designed or determined to be acceptable for the wheel loadings of the cask transport vehicle.

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4.2 STORAGE SITE

The ISFSI is designed in accordance with the General Design Criteria set forth in 10CFR72(F). Table 4.2-1 summarizes compliance with these criteria. Additional details are provided below.

4.2.1 STRUCTURES

The operational area of the ISFSI consists of two concrete pads and the surrounding compacted gravel area. Figure 4.2-1 shows the concrete storage pad plan, cross section, and details.

The primary function of the concrete pads is to provide a uniform level surface for storing the casks. The "minimum" pad elevation criterion has been set at 693 ft.-0 in. msl to preclude immersion of the cask seals during the probable maximum flood. Actual pad elevation is 694 ft.-6 in. The gravel areas around the pads are compacted to allow for movement and positioning of the transport vehicle and tow vehicle.

Cask drop and tip accidents are analyzed in Section 8.2.8. This analysis establishes that the TN-40 casks can maintain their integrity in the event of impact onto the concrete pad. The cask analysis was performed using the following nominal soil and concrete parameters generated by Transnuclear Incorporated utilizing Reference 15:

Overall Pad thickness	36 inches
Reinforcement	No. 14 bars top and bottom, two way with 1 bar each 12 in., with a 2 in. cover with $E = 30\times10^6$ psi and yield stress = 60,000 psi
Concrete Elastic Modulus	3.6×10^6 psi
Concrete Compressive Strength	4,000 psi
Concrete Cracking Strength	400 psi
Concrete Poisson's Ratio	0.17
Soil Elastic Modulus	30,000 psi
Soil Strength	300 psi

The concrete is designed for a nominal compressive strength of 3,000 psi at 28 days. Testing was performed in accordance with requirements specified in ASTM C31 (Reference 2) and ASTM C39 (Reference 3).The design of the concrete pads includes the following loads identified in Section 6.17.1 of ANSI/ANS-57.9 (Reference 5):

Dead Load *(D):*	Dead load of the structure and attachments.
Live Load *(L)*	Live loads including snow and operational loads.
Earthquake Load *(E)*	Loads generated by the ISFSI design earthquake.
Wind Load *(W)*	Loads generated by the design wind.
Flood Loads *(F)*	Loads resulting from maximum hypothetical flood including buoyancy and dynamic pressure.

The design of the concrete pads includes the following load combinations identified in Section 6.17.2.1 of ANSI/ANS-57.9:

$$U_c > 1.4D + 1.7L$$

$$U_c > 0.75\ (1.4D + 1.7L + 1.7W)$$

$$U_c > D + L + E$$

$$U_c > D + L + F$$

where U_c = available concrete strength

The factor of safety against overturning and sliding is based on Section 6.17.4.1 of ANSI/ANS-57.9 as modified below:

	Minimum Safety Factors	
Load Combinations	Overturning	Sliding
D+L+E	*1.1*	*1.1*
D+L+W	*1.1*	*1.1*
D+L+F	*1.5*	*1.5*

The concrete pad was analyzed using Images-3D (Reference 6) which is a finite element computer program. The analysis was performed to verify that the strength of the pad was adequate to prevent unacceptable cracking or differential settlement and that the casks would not tip under design loads. A 9x49 array, or a total of 441 nodes was used. The lines connecting the nodes define 384 elements of the model. Three construction joints were modeled by designating separate but coincident nodes. Node numbers 109-117 are coincident with nodes 442-450, nodes 217-225 are coincident with nodes 451-459, and nodes 325-333 are coincident with nodes 460-468 and correspond to the locations of the construction joints (See Figure 4.2-1a).

Translational springs with an arbitrary stiffness of 50,000 kips/ft. in the three orthogonal directions were used to connect these coincident nodes. This allows the model to rotate at this interface, but not translate.

Vertical translation springs were placed to model the soil stiffness. Horizontal translation springs were modeled in each direction at the corners. These horizontal springs see no forces but were included to provide numerical stability. Rotational springs were also modeled at the center of each of the four segments of the model. These springs were also included to provide numerical stability and see no forces.

4.2.1.1 STATIC ANALYSIS

Soil springs for the static model were calculated assuming a settlement of 1.5 inches over an area of 18 ft. x 18 ft. due to a 240.7 kip cask load (Reference 17). This unit area spring was then scaled on a node by node basis as a function of its tributary area.

The worst case loading pattern that produces maximum stress in the concrete pad was determined. The maximum stress occurs with the casks at positions 3-20.

In addition to cask loads, loads representing the transport vehicle were placed on nodes 46, 50, 64, and 68. At positions where casks are located, the 240.7 kip cask weight was split between the node directly beneath the cask and the four adjacent nodes. A live load factor of 1.7 was applied, giving a resulting load of 81.8 kips per node. The transport vehicle weight was assumed to total 150 kips. A live load factor of 1.7 was applied, giving a resultant load of 63.8 kips per node.

The maximum tensile or compressive stress in the mat was calculated using the above loads to be 86.1 kips/ft^2. Since all of the stress is induced by bending (no in-plane forces), these values may be used to calculate the maximum moment in the mat. Since ultimate load factors were used in the loading, these represent ultimate moments.

The ultimate moment capacity of the slab was calculated to be 276 ft.-kips. The actual moment calculated from the maximum stress was determined to be 129 ft.-kips, which therefore is within the capacity of the pad.

4.2.1.2 DYNAMIC ANALYSIS

A dynamic analysis was performed to determine stresses and to consider cask overturning due to seismic response to the ISFSI design earthquake. The model used the same geometry and connectivity as used in the static analysis. A set of soil springs calculated to model the dynamic soil-structure interaction was substituted for the static springs.

Lumped dynamic soil springs for a rigid rectangular footing were calculated in accordance with Reference 7. Values for the following were determined for both the normal condition (pad above water table) and the flood condition (pad below water table):

Vertical spring constant (K_z)

Sliding spring constant (K_x)

Rocking spring constant (K_ψ)

Torsional spring constant (K_Θ)

Results are shown on Table 4.2-2. These lumped spring constants were then apportioned to the various nodes as a function of tributary area. The soil springs for the dry condition controlled the design, since they provided the least stiffness.

Members representing the casks were modeled from the center of gravity of the cask to a point directly below on the mat. Accelerations and forces were calculated based upon the response spectra for the ISFSI design earthquake.

The analysis utilized the seismic response spectra specified in Figure 2.5-8 considering a 5% damping ratio for the soil.

Three modal spectral analyses were performed, using three, five, and eight modes, to determine the sensitivity of acceleration response to the number of modes. Analyses showed that the use of five modes would give a good approximation. Five modes were therefore used for dynamic analyses.

The maximum stresses on the pad due to dynamic loading were calculated for three cases to observe the relationship between stress and cask positioning. The results of these analyses demonstrate that the worst case is when there is a single cask. The maximum stress was calculated to be 24.1 kips/ft^2.

The total stress resulting from the addition of dynamic seismic plus maximum unfactored static (86.1 kips/ft^2 ÷ 1.7 load factor = 50.6 kips/ ft^2) stress equaled 74.7 kips/ft^2. Since the total combined stress $(D + L + E)$ was less than the ultimate static stress $(1.4D + 1.7L)$ of 86.1 kips/ft^2, it was concluded that ultimate static stresses governed the design, and dynamic seismic load combinations need not be considered further.

The springs 450 to 530 in the model represent the joints between the segments of the slab. To design the shear dowels between these segments, the maximum force in the springs for various loading conditions was determined. The maximum shear force was 88.6 kips at spring 452.

The tributary width of spring 452 is 2.604 ft., so maximum shear force at spring 452 is 34.0 kips/ft. The shear stress is resisted by reinforcing steel dowels across the joint.

The dynamic stability of the cask was determined from the results of the dynamic analysis. The cask overturning moment resulting from accelerations calculated using five modes was determined and compared to the restoring moment resulting from cask weight. The factor of safety is the ratio of the restoring moment to the overturning moment. Various cask storage configurations were analyzed. The minimum factor of safety for overturning, which resulted from the storage of a single cask on a pad, was determined to be 1.35. The minimum factor of safety for sliding was calculated to be 1.14. 01145327

4.2.2 STORAGE SITE LAYOUT

The overall layout of the ISFSI is shown on Figure 1.3-1. Engineering drawings showing sections and details of the concrete pads are presented on Figure 4.2-1.

Confinement of radioactivity is accomplished solely by the storage casks and is not dependent upon the particular layout of the installation. Therefore, other than the casks themselves, no confinement features are provided at the ISFSI.

4.2.3 STORAGE CASK DESCRIPTION

This section summarizes the structural analysis of the TN-40 storage cask. For purposes of structural analysis the cask has been divided into four components: the cask body (consisting of containment vessel and gamma shielding), the basket, the trunnions and the neutron shield outer shell. The following information is provided: a brief description of the components, the design bases and criteria, the method of analysis, a summary of stresses for the highest stressed locations, and a comparison with the allowable stress criteria.

4.2.3.1 DESIGN BASIS

4.2.3.1.1 CASK BODY

The cask body is described in detail in Section 1.3. Figures 1.3-2, 1.3-3, and 1.3-4 show the cask body. The containment shell and lid materials are SA-203 Grade D, or SA-203 Grade E and SA-350 Grade LF3, respectively. The gamma shielding is SA-105 (Table 1.3-2 for alternates). The TN-40 cask body is designed, fabricated, examined and tested in accordance with the requirements of Subsection NB of the ASME Code (Reference 8) to the maximum practical extent. The containment boundary, which consists of the inner shell and bottom plate, shell flange, lid outer plate, lid bolts, penetration cover plates and bolts, is of particular interest. The containment boundary welds are full penetration welds examined volumetrically by radiograph. These welds are also magnetic particle or liquid penetrant examined. The acceptance standards are in accordance with Article NB-5000.

The welds between the gamma shield forgings are volumetrically examined by ultrasonic or gamma x-ray techniques to verify shielding continuity. Other structural and structural attachment welds are examined by the magnetic particle or liquid penetrant method in accordance with Section V, Article 6 or 7 of the ASME Code.

Acceptance standards are in accordance with Section III, Subsection NB, Paragraph NB-5340 or NB-5350. Seal welds are examined visually and by liquid penetrant or magnetic particle methods in accordance with Section V of the ASME Code. Stainless steel overlay welds are examined by the liquid penetrant method in accordance with Section V of the ASME Code.

Electrodes, wire, and fluxes used for fabrication must comply with the applicable requirements of the ASME Code, Section II, Part C. The welding procedures, welders and weld operators must be qualified in accordance with Paragraph NB-4300 of Subsection NB.

4.2.3.1.2 BASKET

The basket structure consists of an assembly of square 304 stainless steel fuel compartment boxes or cells attached together using cylindrical plugs welded to the walls of adjacent boxes. Trapped between the adjacent boxes are two layers of 6061-T6 aluminum which surround a layer of boral. The stainless steel boxes and plugs effectively clamp and pin the aluminum thermal conductor plates and boral poison plates in place. The plugs are assembled through clearance holes in the aluminum and boral plates and are only welded to the stainless steel boxes. Additional curved aluminum plates formed to the cask cavity curvature are welded to the tips of the thermal conductor plates at the periphery of the basket. Figures 1.3-6 and 1.3-7 show details of the basket.

The basket is supported tangentially by 6061-T6 aluminum rails (shown in Figure 1.3-3) bolted to SA-203 inserts welded to the containment shell

4.2.3.1.3 TRUNNIONS

The trunnions are cylindrical SA-105 or SA-266 forgings that are groove welded to the cask body gamma shielding. The two upper trunnions are designed to lift the loaded TN-40 cask vertically. The lower trunnions provide capability to rotate the cask prior to loading of spent fuel. The trunnions are designed to meet the requirements of ANSI N14.6 (Reference 9). The trunnions are shown in Figure 1.3-3.

4.2.3.1.4 OUTER SHELL

The outer shell of the neutron shield consists of a cylindrical shell section with segmented closure plates at each end. Each segmented closure plate is welded together. The top and bottom closure plates are welded to the outer surface of the cask body gamma shielding. The outer shell provides an enclosure for the resin-filled aluminum containers and maintains the resin in the proper location with respect to the active length of the fuel assemblies in the cask cavity. The outer shell has no other structural function. The shell is carbon steel protected by a metallic coating and paint. 01136263

4.2.3.1.5 TOP NEUTRON SHIELD

The top neutron shield consists of a disc of commercial grade polypropylene surrounded by a steel enclosure. The top neutron shield is attached to and rests on the cask lid. It is protected from the environment by the protective cover.

4.2.3.2 INDIVIDUAL LOAD CASES

This section outlines the TN-40 analyses performed under the various loading conditions identified in Section 3.2. These loadings include all of the normal events that are expected to occur regularly. In addition, they include severe natural phenomena and man-induced low probability events postulated because of their potential impact on the immediate environs.

Section 3.2.5 lists all of the TN-40 loadings in Table 3.2-5. These loads are described in detail in Section 3.2.5.2. The loads selected for analysis of the cask are discussed in Section 3.2.5.3. Numerical values of these loads are listed in Tables 3.2-1 through 3.2-4.

The TN-40 components have been evaluated under these loads through numerical analysis. Finite element models of the cask body and basket have been developed, and detailed computer analyses have been performed using the ANSYS computer program (Reference 10). Other components such as the lid bolts and trunnions have been analyzed using conventional textbook methods. Table 4.2-3 lists the specific individual load cases analyzed for each major TN-40 component. The SAR sections where these analyses are described and the tables listing the stress results, where applicable, are also indicated. Note that the combined results of these analyses and their evaluation to the structural criteria of Section 4.2.3.3 below are summarized in Section 4.2.3.4.

4.2.3.3 STRUCTURAL DESIGN CRITERIA

This section describes the structural design criteria for the major components of the TN-40 storage cask. The cask consists of four major types of components:

- Containment Boundary
- Non-containment Structure
- Basket
- Trunnions

The structural design criteria for these components are described below:

4.2.3.3.1 CONTAINMENT BOUNDARY

The containment vessel consists of the cask body assembly inner shell (both cylinder and bottom) and closure flange out to the seal seating surface and the lid assembly outer plate. The lid bolts and seals are also part of the containment boundary. The containment boundary is designed to the maximum practical extent as an ASME Class I component in accordance with the rules of the ASME Code, Section III, Subsection NB. The Subsection NB rules for materials, design, fabrication and examination are applied to all of the above components to the maximum practical extent.

The stresses due to each load are categorized as to the type of stress induced, e.g. membrane, bending, etc., and the classification of stress, e.g. primary, secondary, etc., determined. Stress limits for containment vessel components, other than bolts, for Design (same as Primary Service) and Level A and D Service Loading Conditions are given in Table 4.2-4. The stress limits used for Level D conditions, determined on an elastic basis, are based on the entire structure (containment shell and gamma shielding material) resisting the accident load. Local yielding is permitted at the point of contact where the load is applied. If elastic stress limits cannot be met, the plastic system analysis approach and acceptance criteria of Appendix F of Section III may be used. The limits for the containment bolts are listed in Table 4.2-5.

The allowable stress intensity value, S_m, as defined by the Code are taken at the temperature calculated for each service load condition.

4.2.3.3.2 NON-CONTAINMENT STRUCTURES

Certain components such as the gamma shielding, the neutron shield outer shell and the trunnions are not part of the cask containment boundary but do have structural functions. These components, referred to as non-containment structures, do not have containment functions but are required to react the containment loads and in some cases share loadings with the containment structure. The design criteria for the trunnions are both unique and specific. They are specified in Section 3.2.5.4. The stress limits for the remaining non-containment structures are given in Table 4.2-6. These limits are somewhat less restrictive than those specified in Table 4.2-4 for the containment vessel.

4.2.3.3.3 BASKET

The stress limits for the basket are summarized in Table 4.2-6a.

The basket structural design criteria for a hypothetical impact accident are developed in Section 4B.5 of the Basket Analysis Appendix 4B. They are summarized here.

The basket fuel compartment wall thickness is established to meet heat transfer, nuclear criticality, and structural requirements. The basket structure must provide sufficient rigidity to maintain a subcritical configuration under the applied loads. The primary stress analysis of the basket for sustained Design and Level A Service Conditions does not take credit for the aluminum conductor plates except for through thickness compression. The aluminum is, however, considered when determining secondary stresses in the stainless steel.

The basis for the 304 stainless steel fuel compartment box stress allowables is Section III of the ASME Code. The primary membrane stress and primary membrane plus bending stress are limited to S_m (S_m is the code allowable stress intensity) and 1.5 S_m, respectively, at any location in the basket for Design and Level A load combinations. The range of primary plus secondary stress is limited to 3 S_m for Level A combinations. This allows some local yielding of the basket structure. However, the thermal stresses are self-relieving and the deformation is insignificant. In addition, the thermal stress will decrease with time as the decay heat load decreases. The average primary shear stress across a section is limited to 0.6 S_m.

The sustained Level D Service Conditions are actually elevated to Design Conditions and evaluated against Design Limits since the 3 g bounding loads are greater than any Level D loads.

See Appendix 4B for complete details of the criteria for the hypothetical drop impact accident.

The hypothetical impact accident is evaluated as a short duration Level D condition. Since elastic quasistatic analyses are performed, the primary membrane stress is limited to 2.4 S_m and the membrane plus bending stress limited to 3.6 S_m. The average primary shear stress across a section is limited to 0.42 S_u (S_u is the minimum ultimate strength).

Individual fuel compartment wall panels, when subjected to compressive loadings, are also evaluated against ASME Code rules for component supports and B96.1 (Reference 11) to ensure that buckling will not occur. The interaction between compression and bending was evaluated using the equations of paragraph NF-3322. These equations reduce to that below for members subjected to both axial compression and bending:

$$\frac{Applied\ Comprehensive\ Load}{Allowable\ Comprehensive\ Load} + \frac{Applied\ Bending\ Moment}{Allowable\ Bending\ Moment} \le 1.0$$

See Appendix 4B for the development of the stability and interaction criteria.

4.2.3.4 EVALUATION

Section 4.2.3.2, above, lists the various individual load cases analyzed to evaluate the TN-40 storage cask in Table 4.2-3. The loads are described in detail in Section 3.2 and listed in Table 3.2-5. Section 3.2 also categorizes these loads for the cask body as indicated in Tables 3.2-6 through 3.2-8 into Design, Level A and Level D Service Loadings and lists the load combinations to be evaluated. Table 4.2-7 (identical to the load combination table in Section 3.2) repeats these load combinations. Each combination is a set of loads that is assumed to occur simultaneously. Note again, that all of the combinations that would be Level D cases, except those including tornado missile loads, have been elevated to Design Conditions. Note also that the hypothetical drop accident is analyzed and evaluated in Section 8.2.8.

4.2.3.4.1 CONTAINMENT VESSEL

All of the eight combinations listed in Table 4.2-7 have been performed in Appendix 4A for each of 20 cask body locations indicated in Figure 4.2-6 (8 of these are containment locations). Tables 4.2-8 and 4.2-9 list the highest containment shell, flange, and lid stress intensities for each service condition and identify the load combination and location where those maxima occur. Also listed in the tables are the stress limits for that service condition based on the Section 4.2.3.3 structural design criteria.

Note that the highest Design stress intensity is 13,865 psi (P_l+P_b), the highest Level A value is 22,132 psi (P_l+P_b+Q), and the highest Level D value is 47425 psi (P_l+P_b). These values are well below the limits indicated. Therefore the stresses in the containment vessel are acceptable.

4.2.3.4.2 GAMMA SHIELDING

The eight load combinations for the 12 gamma shielding and weld locations indicated in Figure 4.2-6 have also been performed in Appendix 4A. Table 4.2-10 lists the highest cylinder, bottom and weld stress intensities for each service condition and identifies the load combination and location where those maxima occur.

The highest design stress intensity is 14,574 psi (P_m), the highest Level A value is 15,953 psi (P_l+P_b+Q) and the highest Level D value is 55,519 psi. These values are again below the limits indicated.

One case not included in these combinations is the stress caused by cold rain on a hot cask. A conservative analysis was performed as indicated in Appendix 4A to determine the resulting stresses. It was concluded that a stress range of 80,534 psi (Alternating stress of +28,432 psi) could occur each time the outer skin of the top of the gamma shielding is cooled. The ASME Code fatigue curves permit 22,000 cycles for this stress. Therefore this condition is acceptable. Note that the temperature and stress level in the containment vessel does not cycle.

4.2.3.4.3 LID BOLTS

The stress intensities in the lid bolts as calculated in Appendix 4A are summarized in Table 4.2-11. The highest Design value is 25,000 psi tension, and the combined Level A and Level D stress is 80,534 psi. These values are well below the allowables.

4.2.3.4.4 BASKET

Table 4.2-12 summarizes the stresses in the basket. The values listed are for the 304 stainless steel boxes and plug welds. The aluminum conductor plates and boral poison plates are assumed to have no load carrying capability, except through thickness compression between boxes, to react long duration primary loads.

It should be noted that Design and Level D stresses are identical since the basket is arbitrarily analyzed for a conservatively high 3 g bounding lateral load. No Level A or Level D TN-40 load produces such a high lateral acceleration. The highest primary stress is 6,056 psi (P_l+P_b) for that load and the lowest limit for Design Conditions is 25,700 psi. This limit is for the basket location at the highest temperature (530°F). The highest stress actually occurs at a lower temperature where the allowable is higher. See Section 4B.6.

The aluminum conductor plates are assumed to have strength to apply differential expansion induced (thermal) secondary stresses to the stainless steel plates and plug welds. The highest Level A stress in the stainless boxes is 49,036 psi (P_l+P_b+Q) and the indicated $3S_m$ limit is 51,000 psi. The highest weld primary plus secondary stress intensity is 50,902 psi, which meets the $3S_m$ indicated limit. Note that the ASME Code permits $3S_m$ to be exceeded in this case since thermal ratcheting and fatigue cannot occur. See Section 4B.6.

4.2.3.4.5 TRUNNIONS

The trunnion stresses are summarized in Tables 4.2-13 and 4.2-14. As required by ANSI N14.6, the trunnions are analyzed under a cask loading of 6 g and the stresses are shown to be below the trunnion yield strength of the material. Under a 10 g load the stresses are less than the ultimate strength

4.2.3.4.6 OUTER SHELL

The neutron shield outer shell stresses are summarized in Table 4.2-15. The shell stresses are highest when the cask is vertical and subjected to internal and handling loads. The shell is not analyzed under tornado missile loading, but it would undoubtedly be damaged by either Missile A or Missile B, as defined in Section 3.2.1.2. Radiological effects have been shown to be acceptable, as shown in Table 7A-4. 01136263

4.2.4 INSTRUMENTATION SYSTEM DESCRIPTION

No safety related instrumentation is required due to the passive nature of the ISFSI design.

4.3 TRANSPORT SYSTEM

4.3.1 FUNCTION

The function of the transport system is to move the loaded storage casks from the Auxiliary Building rail bay to the concrete pads in the ISFSI. The transport vehicle is capable of being towed over several different types of ground surfaces, including compacted gravel, concrete paving, and asphalt paving. The performance objectives of the transport system are to move the loaded cask in a vertical position in a manner which will preclude damage to the cask body and its internals and to any other safety or security related system or component which could be affected by cask transport. The transport vehicle shall be designed for a minimum of 100 fully-loaded one-way trips over approximately a 25 year period.

4.3.2 COMPONENTS

The transport vehicle is designed and fabricated by Ederer, Inc., of Seattle, Washington, based on the general design criteria discussed below. Figures 4.3-1 and 4.3-2 show side and plan views of the transport vehicle.

The Transport vehicle structural frame is fabricated of welded steel plates and shapes sized and connected as required by design stress analysis.

All transport vehicle powered functions are hydraulic. An electro-hydraulic power unit is located on the frame of the transport vehicle. When in operation, the power unit is connected with an outside electrical source at the Auxiliary Building or ISFSI. The power unit consists of a 480V, 3 phase motor coupled to a heavy duty pump. The pump supplies hydraulic fluid to operate the vehicle and cask hoist hydraulic functions. In the event of an electrical power failure, a means is provided at the power unit for lowering the cask. Between power supply locations, the hydraulic operating functions of the hoist are not required and the system is valved closed to prevent hoist movement.

The hoisting mechanism consists of a 'U' shaped steel lift beam with pivot pins at one end connected to the steel structural frame. The other end is raised and lowered by a 12 inch hydraulic cylinder. Lifting links for engagement of the upper cask trunnions are located near the mid-point of each side of the lift beam. The lift links move inward and outward on pins by means of hydraulic cylinders, and can be positioned independently to accommodate off-center positioning of the transport vehicle relative to the cask.

The hydraulic cylinder used for cask lifting is a heavy duty design made of steel. Flow control valves are located on the outlet of the cylinders in order to restrict the flow of oil from the cylinders and control the rate of descent. The operating pressure of the cylinders is designed not to exceed 3,000 psi. The cylinders are equipped with lift cylinder locking valves which assure that the cask will not be accidentally lowered.

The transport vehicle and its cask load are carried on eight wheels. The wheels are arranged in four pairs. The two front pairs of wheels which are on the normal towing end of the vehicle are steerable and have a 30 ft. minimum turning radius. The front wheel pairs are steered by a tow bar and steering linkage from a separate tow vehicle.

A detachable dolly with two wheels can be coupled to the transport vehicle to allow towing in the reverse direction. The vehicle is towed into the Auxiliary Building in the reverse direction to avoid having to back up the vehicle. When in place, the dolly lifts the rear transport vehicle wheels off the ground to enable the dolly to steer the vehicle. The dolly cannot be used when the transport vehicle is loaded with a cask. A locking pin is used to lock the front wheel steering when towing the vehicle in reverse.

The rear wheels of the transport vehicle are designed to move inward and outward. The wheels must be moved inward (traveling position) to pass through the Auxiliary Building access door and moved outward (load/unload position) to provide clearance to load and unload the casks. Hydraulic jacking pads extend to the ground to raise the rear wheels and carry the vehicle weight when the wheels are moved to the required position. Overall transporter width is 13'-0" in the travel position and 19'-4" in the load/unload position.

The hydraulic cylinders for the jacking pads are fitted with restraint links which are used to engage the lower trunnions of the cask. The cask restraints ensure the cask remains secure while en route to the storage area.

Each of the four pairs of wheels have a parking and emergency brake disc assembly. The parking brakes are designed with springs which set the brakes when hydraulic pressure is released. The brakes must be pressurized by an operator to release and allow the vehicle to move. The parking brakes are also interlocked and must be set to enable lifting and lowering operations. The emergency brakes are dynamic brakes for use in the event the tow vehicle brakes fail. The emergency brakes are designed to gradually engage to minimize shock loads, and are manually engaged by emergency stop levers located on either side of the transport vehicle. The brake system is equipped with an accumulator which supplies hydraulic fluid pressure to operate the brakes when the vehicle is not connected to the outside power source. A manual hydraulic pump is provided to repressurize the accumulator if the pressure has dropped too low to release the parking brakes.

A pendant control station is suspended from a swing arm at the rear of the transport vehicle allowing it to be positioned to either side of the vehicle. Controls include pump on-off, cask raise-lower, rear wheel jacking raise-lower, rear wheel travel-load/unload, cask lift links in-out, and front steering lock-unlock. The control station functions can only be operated when outside power is supplied to the vehicle.

The transport vehicle tires are designed to distribute the load and provide adequate mobility. Pneumatic rubber earthmover type tires are used. Tire pressure is maintained at a maximum of 100 psi to assure smooth travel on haul roads. The tires are filled with polyurethane foam to prevent tire deflation.

The transport vehicle travels over a cask transport route designed with a maximum ground slope of 2%. A maximum ground slope of 5% is considered in the design of the transporter structure, the cask restraints, and the disc brakes. New transport roads are designed for a CBR (California Bearing Ratio) of 80. Existing roads are paved with asphalt and have been tested for transport vehicle load acceptability.

The transport vehicle, loaded with cask, is designed for a maximum towing speed of 3 miles per hour and to remain dynamically stable at this speed.

The transport vehicle is equipped with eight perimeter marker lights and four overhead adjustable headlights for safety and/or night travel. The lights are powered from the tow vehicle electrical system.

4.3.3 DESIGN BASIS AND SAFETY ASSURANCE

The transport vehicle is designed to limit cask lift height to a maximum of 10 in. Hydraulic lock valves and rigid attachment of lugs to the trunnions maintain the lifting system integrity during transport of casks. Positive restraints prevent excessive cask motion during transport.

The transport vehicle design considers static, dynamic, and fatigue loads associated with the transport process. Load cases include:

Normal Loads: Loads on the vehicle with a maximum weight payload through all ranges of motion for each mechanism while the vehicle is stopped or braking on a 5% road grade, 2% road crown side slope, resultant tire deflection slope and 3.5" maximum cask shift from vehicle centerline.

Cyclic Loads: Loads produced by normal operation or by towing the vehicle over the roadway ridge where either an inside or outside front wheel and the diagonally opposite rear wheel are on the ridge (Figure 4.3-3 and 4.3-4). These loads are the bases for the fatigue load analysis.

Worst Case Loads: Loads produced by road conditions beyond those allowed by specification, but which must not produce vehicle failure. The worst case loading is assumed to occur when the vehicle is towed diagonally over the road ridge, as in the cyclic load case, where the ridge is elevated an additional 4" due to a bump (i.e. rock, wood, etc.) (Figure 4.3-5 and 4.3-6). The allowable stresses for the worst case loading is 90% of yield stress.

All hydraulic lifting and restraining cylinders are designed such that they will not fail when supporting a load equal to five times the cask weight. Structural design is in accordance with allowable stresses as set forth in AISC Manual of Steel Construction (Reference 12) or one-third of ultimate, whichever is more stringent.

4.3.4 DETERMINATION OF NATURAL FORCES ON A LOADED TRANSPORT VEHICLE

The time requirement necessary to move the casks from the plant to the storage facility is less than one day. This time space is so short that the probability a natural force caused by a tornado, flood or seismic event could occur while transferring a cask is extremely unlikely. However, the following sections are provided to quantify the effect of natural phenomena on the transporter loaded with a cask full of spent fuel assemblies, to show that a loaded transport vehicle tip over is not a credible event.

4.3.4.1 STABILITY OF A LOADED TRANSPORT VEHICLE UNDER TORNADO LOADING

Tornado loadings specified in Table 12.2-9 of the Prairie Island USAR (Reference 1) that were used are as follows:

- A lateral force caused by a funnel of wind having a peripheral tangential velocity of 300 mph and a forward progression of 60 mph.

- The design tornado driven missile assumed equivalent to an airborne 4" x 12" x 12'0" plank traveling end-on 300 mph, or a 4000 lb automobile flying through the air at 50 mph and not more than 25 ft above ground level.

4.3.4.1.1 WIND LOADING

The maximum wind loads are assumed to be the same as tornado loads. The maximum tornado wind is assumed to be 360 mph based on a summation of the peripheral tangential velocity and forward progression of a design basis tornado.

Tornado winds may contact the transport vehicle from any direction. Since the vehicle is equipped with parking brakes which must be released by the operator to begin movement, it is assumed the vehicle will not roll due to the wind pressure. The maximum surface area is a loaded transport vehicle's side profile. In the case of tipping, wind pressure normal to the side profile also occurs in the same plane as the transport vehicle's minimum wheel base. Therefore, the side profile is used to analyze a tipping accident.

The transport vehicle is designed with a clearance of 3.5" between either side of the cask and the vehicle side members. In the tipping analysis, the cask is assumed to be located 3.5" off the transport vehicle centerline against the vehicle's side member. This "cask shift" is unlikely but is considered in order to be conservative.

The roadway used for transport is designed with only a 2% side slope (road crown) and the drainage embankments along the road maintain a slight 3 to 1 slope. Assuming the transport vehicle is traveling entirely to the side of the roadway centerline, additional loads due to the 2% side slope plus transport vehicle tilt caused by tire deflection from the cask shift and tornado wind loads are also considered. The transport vehicle loading conditions are shown on Figure 4.3-7.

A tornado wind velocity of 408 mph is required to cause the loaded transport vehicle to begin to tip. This velocity is calculated as follows:

The tipping moment about the bottom outside tire of the transport vehicle (pivot point P) is:

$$M_p = \left[\frac{b}{2} - x\right] W \cos\Theta - \frac{h}{2} Aq - h_{cg} W \sin\Theta$$

where:

b = outside wheel base width = 130"

x = CG shift + road side slope offset + tire deflection offset

W = loaded transport vehicle weight

= 240,000 (cask) + 150,000 (transport vehicle) = 390,000 lbs.

Θ = vertical angle vehicle tilts due to tire deflection

h = overall vehicle height = 238"

A = projected area of loaded transport vehicle conservatively calculated as approximately 50,000 in^2

q = wind pressure, psi

h_{cg} = height of loaded vehicle center of gravity = 104"

As the wind pressure increases, the transport vehicle will tend to tilt due to tire deflection. The vehicle has four front and rear sets of tires as shown on Figure 4.3-7. If the point of vehicle tip occurs when the windward side tire load becomes zero, indicating the tire is about to raise, then the relationship between the deflection of the four tire sets, which is linear, is expressed as:

$$d_2 = \frac{94.5}{130} d_1 \qquad d_3 = \frac{35.5}{130} d_1 \qquad d_4 = 0$$

where d is tire deflection from the pivot point tire d_1, to the tire without a load, d_4.

The deflection spring constant for a single tire is 14,000 lbs/in (Reference 16) such that:

tire load (R) = 14,000d x 2 (front + rear tires)

Since $R_1 + R_2 + R_3 + R_4 = W$ = 390,000 lbs and $R_4 = 0$ then deflections for the second and third set of tires can be substituted into the above equations to yield:

$$28{,}000d_1 + 28{,}000\frac{94.5}{130}d_1 + 28{,}000\frac{35.5}{130}d_1 = 390{,}000$$

or

$$d_1 = 7''$$

Considering the 3.5" cask shift, the mean center of gravity between the cask and the transport vehicle is shifted horizontally toward the pivot point (P) a distance:

$$h_{cg} = \frac{240{,}000\,(65 - 3.5) + 150{,}000\,(65)}{390{,}000} = 62.8''$$

Solving for the total offset (x) at the center of gravity:

$$x = (65 - 62.8) + .02(104) + 7(104/130) = 9.88''$$

The vehicle tilt angle is then, $\Theta = \text{Arctan } 9.88/104 = 5.43°$

Therefore, solving for wind pressure (q):

$$\left[\frac{130}{2} - 9.88\right]390{,}000\cos 5.43° = \frac{238}{2}50{,}000q + (104)390{,}000\sin 5.43°$$

$$q = 2.95psi = 425psf$$

And the corresponding wind speed (v) required to tip the loaded transport vehicle is:

$$v = \sqrt{\frac{425}{0.002558}} = 408 \text{ mph}$$

Since 408 > 360, the loaded transport vehicle will not tip as a result of the design tornado wind loads.

4.3.4.1.2 TORNADO MISSILE IMPACT

The 4,000 lb automobile traveling at 50 mph is more likely to tip the loaded transport vehicle than the airborne plank due to its higher initial momentum. The automobile impact with the transport vehicle is shown in Figure 4.3-7 and is placed at the top of the transport vehicle for maximum tipping potential.

The impact is assumed to be totally inelastic such that all kinetic energy from the airborne missile is transferred to the loaded transport vehicle into potential energy as the transport vehicle tips and the center of gravity lifts. It is also assumed that the transport vehicle components will retain structural integrity during missile impact. In the event a component, such as the hoist beam, fails, the cask will simply drop 10" to the ground. The cask is determined to be structurally sound in Section 8.2.8.2.2 for drops up to 18 inches.

Using the conservation of momentum, the transport vehicle angular velocity about point P (ω_p) is:

$$\omega_p = \frac{m d_{cg} V_0}{m(d_{cg})^2} + I_p$$

where:

m = mass of missile = 4000/386 $lbs \cdot sec^2/in$

d_{cg} = distance between center of gravity and pivot point P (bottom outside tire centerline)

V_o = initial velocity of missile = 73 fps = 876 in/sec

I_p = moment of inertia of loaded transport vehicle about pivot point P

The distance d_{cg} from the loaded transporter center of gravity and pivot point P, is calculated from the CG height of 104" and horizontal mean distance of 62.8" as 121.5".

The moment of inertia of the cask about pivot point P is:

$$I_{p\,cask} = m/12(3r^2 + l^2) + m{d_{cg}}^2{}_{cask}$$

where:

m = mass of cask = 240,000/386 $lb{\cdot}sec^2/in$

r = radius of cask = 45.5"

l = length of cylinder = 183.75"

$d_{cg\ cask}$ = distance from cask center of gravity to pivot point P

$$I_{p\ cask} = \frac{240{,}000}{386x12}\left[3(45.5)^2 + (183.75)^2\right] + \frac{240{,}000}{386}\left[(92+10)^2 + (65-3.5)^2\right]$$

$= 10.89x10^6\ in{\cdot}lb{\cdot}sec^2$

The moment of inertia of the transport vehicle about pivot point P is:

$$I_{p\ transport\ vehicle} = m/12\ (h^2 + w^2) + md_{cg}{}^2{}_{transport\ vehicle}$$

where:

m = mass of transport vehicle = 150,000/386 $lb{\cdot}sec^2/in$

h = height of transport vehicle (calculated as a rectangular parallelepiped) is taken as twice the center of gravity height of 107" = 214"

w = overall width of transport vehicle = 156"

$d_{cg\ transport\ vehicle}$ = distance from transport vehicle center of gravity to pivot point P

$$I_{p\ transport\ vehicle} = \frac{150{,}000}{386x12}\left[(214)^2 + (156)^2\right] + \frac{150{,}000}{386}\left[(107)^2 + (65)^2\right]$$

$= 8.36x10^6\ in{\cdot}lb{\cdot}sec^2$

Total $I_p = 10.89x10^6 + 8.36x10^6 = 19.25x10^6\ in{\cdot}lb{\cdot}sec^2$

Therefore, the angular velocity (ω_p) about P is:

$$\omega_p = \frac{\frac{4{,}000}{386}(121.5)(876)}{\frac{4{,}000}{386}(121.5)^2 + 19.25\times10^6} = 0.06\ rad/\sec$$

As the loaded transport vehicle tips about point P at impact, the kinetic energy is transferred to potential energy as the center of gravity rises a distance y:

$$E_{tipping} = \text{Kinetic Energy} = \text{Increase in Potential Energy}$$

$$= \tfrac{1}{2}\, I_p\, \omega_p^2 = Wy$$

$$= \tfrac{1}{2}\,(19.25 \times 10^6)(.06)^2 = 390{,}000y$$

$$y = 0.09''$$

Clearly, the effect of the airborne automobile impact on the loaded transport vehicle is negligible and will not tip over the transport vehicle.

4.3.4.2 FLOOD LEVEL

It is reasonable to assume that cask transportation will not be allowed during a potential flood occurrence. However, the transport vehicle will withstand loads from forces developed by the probable maximum flood including hydrostatic effects and dynamic phenomena. As determined in Section 3.2.2, the hydrostatic forces developed by a maximum flood on the cask were 20% less than the wind forces. It is therefore concluded that the hydrostatic forces against the transport vehicle are only a fraction of the wind forces and that the probable maximum flood will not cause the loaded transport vehicle to tip.

4.3.4.3 EARTHQUAKE

The transport vehicle is not designated a safety related component and therefore is not subject to specific seismic design requirements. However, this section provides the necessary evaluation based on the ISFSI design earthquake response spectra at the Prairie Island Nuclear Generating Plant ensuring that the loaded transport vehicle will not tip due to seismic loading.

The loaded transport vehicle is generally a flexible system with low frequencies which would probably not be excited due to the short duration of a seismic event. The rubber foam filled vehicle tires will tend to dampen any ground motions and minimize seismic loads such that the normal dynamic loads, due to vehicle travel, are assumed to be greater than potential seismic loads. The seismic loads would not be sufficient to result in a tip over of the transport vehicle containing a loaded cask. In the event a seismic load could cause vehicle failure, the cask would drop or lower to the ground as vehicle members fail or yield. The cask is determined to be structurally sound for drops up to 18 inches in Section 8.2.8.2.2.

The loaded transport vehicle with its eight tires can be represented as a large mass supported by eight springs. Using the simple harmonic equation, the frequency of the loaded transport vehicle is:

$$f = \frac{\pi}{2}\sqrt{\frac{k}{m}}$$

where:

f = frequency of the loaded transport vehicle

k = spring constant of the tires = 14,000 lbs/inch/tire (Reference 16)

m = mass of loaded transport vehicle = 390,000/386 lb·sec²/in

so:

$$f = \frac{\pi}{2}\sqrt{\frac{14{,}000(8)396}{390{,}000}} = 1.7 \text{ cycles per second}$$

From the ground response spectra for the ISFSI design earthquake in Figure 2.5-8, the seismic acceleration can be determined. As mentioned above, the damping effects of the transport vehicle tires should be relatively high; however, 5% damping, which is the highest damping plotted on the spectra, is used to be conservative. For a frequency of 1.7 cycles per second at 5% damping, the response spectra yields a seismic acceleration of 0.14g.

Since the transport vehicle is rectangular, consider an earthquake in the worst case direction which is perpendicular to the minimum wheel base dimension. Calculating the tipping or overturning moment about pivot point P with the upward vertical acceleration as 2/3 horizontal, then:

$$M_p = 0 = gWv_{cg} + 2/3gWh_{cg} - Wh_{cg}$$

where:

g = acceleration value necessary to tip the transport vehicle

W = weight of loaded transport vehicle = 390,000 lbs

v_{cg} *= vertical distance to center of gravity = 104"*

h_{cg} *= horizontal distance to center of gravity from pivot point P (from Section 4.3.4.1.1) = 62.8"*

The overturning moment must also consider the 2% road side slope and the tire deflection caused by that slope which is 0.9% (Reference 16).

$$2\% + 0.9\% = 0.029 \qquad\qquad Arctan\ 0.029 = 1.66°$$

Therefore, the g value necessary to tip the loaded transport vehicle is:

$$g = \frac{62.8\cos 1.66° - 104\sin 1.66°}{104 + 2/3(62.8)} = 0.41$$

Since the acceleration necessary to tip the transport vehicle, 0.41g, is greater than the actual seismic acceleration of 0.14g, the loaded transport vehicle will not tip over.

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4.4 OPERATING SYSTEMS

4.4.1 LOADING AND UNLOADING SYSTEMS

4.4.1.1 FUNCTION

The functions of the loading and unloading systems are to transfer the spent fuel from the spent fuel pool to the storage cask and move the storage cask from the spent fuel pool to the transport vehicle in the Auxiliary Building rail bay.

The performance objectives are to load the fuel into the casks in such a manner as to preclude damage to the fuel or criticality and to move the loaded cask in a manner which will preclude damage to the cask body and its internals and to any other safety related system or component in the load path.

4.4.1.2 MAJOR COMPONENTS AND OPERATING CHARACTERISTICS

The fuel storage area at Prairie Island Nuclear Generating Plant is located between the two reactor buildings, and consists of a new fuel pit, two pools for storing spent fuel, and a canal for transfer of fuel elements between the reactors and the pool. The two spent fuel storage pools are designated as Pool No. 1 and Pool No. 2. Pool No. 1, the smaller of the two pools, has inside plan dimensions of 18 ft.-11 in. x 18 ft.-3 in. Pool No. 2 has inside plan dimensions of 18 ft.-11 in. x 43 ft.-5 in. Normal water depth for both pools is about 40 ft. The southeast corner of Pool No. 1 is designated as the cask loading and unloading area. Five racks in Pool No. 1 could be used for fuel storage during loading of the spent fuel cask.

The spent fuel pool area is surrounded by a reinforced concrete enclosure. Access into the enclosure for the spent fuel cask is provided by a door and a narrow slot in the ceiling for attaching the spent fuel cask to the overhead Auxiliary Building crane which physically restricts cask movement to the north-south path over the cask set down area of Pool No. 1. Figure 4.4-1 is a layout drawing showing the cask load path between the spent fuel pool enclosure and the Auxiliary Building rail bay. Figure 4.4-2 and 4.4-3 are section views showing the cask load path.

4.4.1.3 SAFETY CONSIDERATION AND CONTROLS

Fuel handling activities in the spent fuel pool are subject to limiting conditions for operation as set forth in Section 3.7 of the Prairie Island Technical Specifications. A single-failure-proof crane was installed for cask handling in the spent fuel pool enclosure and Auxiliary Building rail bay. This crane and the fuel loading and cask handling is subject to conditions as set forth in the Prairie Island Technical Specifications and Operating License.

4.4.2 DECONTAMINATION SYSTEM

Standard decontamination methods will be used to remove surface contamination from the casks resulting from their submersion in the spent fuel pool during fuel loading. Decontamination of the casks will be performed in the cask decontamination area, located in the rail bay of the Auxiliary Building. Decontamination will be done manually, using water detergents and wiping cloths.

4.4.3 STORAGE CASK REPAIR AND MAINTENANCE

Maintenance on the casks can be performed as described in Section 5.1.3.3.

4.4.4 UTILITY SUPPLIES AND SYSTEMS

The storage casks are passive devices. No utility services are needed for operation of the casks.

4.4.5 OTHER SYSTEMS

4.4.5.1 ELECTRICAL SYSTEMS

Non-safety related electrical power is provided to the ISFSI for lighting, general utility, and pressure monitoring instrumentation purposes. Electrical power is provided from a new pole mounted transformer and an existing overhead distribution line.

4.4.5.2 ALARM SYSTEM

Cask interseal pressure will be monitored. The pressure monitoring devices will provide an analog input signal to actuate alarm indication at a monitoring panel outside the ISFSI gate.

4.4.5.3 FIRE PROTECTION SYSTEM

No fires other than small electrical or tow vehicle fuel fires are considered credible at the ISFSI. Accordingly, only portable fire extinguishers are provided. Smoke detectors are installed in the ISFSI buildings to alert operators if a fire is started. The fire fighting equipment at the Prairie Island Nuclear Generating Plant is available, if needed.

4.4.5.4 VACUUM SYSTEMS

A Vacuum Drying System will be used to remove residual water left in the cask after it has been removed from the fuel pool and drained. The system applies vacuum at the vent port vaporizing any water present and sweeping the water out of the cask. The drying time is approximately 12 to 16 hours.

A Vacuum Backfill System will be used to replace air in the cask with helium. The system applies vacuum at the vent port and evacuates the cask cavity to 10 millibar. Once evacuated, the system backfills the cavity with dry helium gas. 01106798

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4.5 CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

The structures, components and systems of the TN-40 Dry Storage Cask and storage facility are classified as Safety Related, Augmented Quality, or Non QA Related. A tabulation of the structures, components, and systems by their classification is shown in Table 4.5-1. The criteria for selecting the classification for particular structures, components, or systems, are based on the definitions given in QF-0202 (Reference 18) 01213464

Safety Related Item

Any structure, system or component that prevents or mitigates the consequences of postulated nuclear accidents that could cause undue risk to the health and safety of the public. Any component whose failure would produce radiation levels at the site boundary in excess of 10CFR100 limits is classified as safety related.

Augmented Quality

Augmented quality is a procurement classification for items or services which do not perform a safety related function, but are subject to special utility requirements or NRC imposed regulatory requirements. This includes items and services classified as non-safety QA related, fire protection related, 10CFR71 related, security related and other applicable site specific items and services. The difference between safety related and augmented quality items and services is that 10CFR50 Appendix B and 10CFR21 requirements do not apply to items and services purchased as augmented quality.

Commercial Material (CM) Standard Quality 01213464

This is a procurement quality classification for items or services which do not have a safety related function and are not subject to special utility requirements or NRC imposed regulatory requirements.

When purchasing safety related or augmented quality items and services, the procurement documents must contain the technical requirements, as well as the source of those requirements. These requirements must be verified by means such as vendor supplied documentation, receipt inspection, or testing. This process is similar for both safety and augmented quality items.

All items related to the storage cask, regardless of their classification, are designed in accordance with the requirements of the TN-40 Design Criteria which ensure that the General Design Criteria of 10CFR72(F), are satisfied. Those items related to the storage cask which are classified as safety related are designed, fabricated, inspected and tested, to the maximum practicable extent, in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB. The lifting trunnions are designed to ANSI N14.6 as described in Appendix 4A. Those items related to the concrete storage pads which are classified as safety related are designed, installed, inspected and tested in accordance with the specific requirements of the American Concrete Institute (ACI) and American Society for Testing and Materials (ASTM).

The quality assurance requirements of 10CFR50 Appendix B to are applied.

Those items which are classified as augmented quality or commercial material
(standard quality) are designed in accordance with design rules which are indicated in 01213464
the structural analysis of those items in Section 4.2.

4.5.1 CONTAINMENT VESSEL

The containment vessel and trunnions are classified as safety related since they serve as the primary confinement structure for the fuel assemblies and are designed to remain intact under all accident conditions analyzed in Chapter 8. The basket is classified as safety related because it provides criticality control, as well as serving as the structural support for the fuel and is designed to remain intact during all of the accidents described in Chapter 8.

4.5.2 PENETRATION GASKETS

Cask interseal pressure will be monitored. The pressure monitoring devices will provide an analog input signal to actuate alarm indication at a monitoring panel outside the ISFSI gate.

4.5.3 SHIELDING

The neutron shield body shield and lid shield are classified as augmented quality items. The basis for this classification is that they perform no function required by the accident analysis in Chapter 8 but they do provide radiation protection for personnel and are related to ISFSI Technical Specification requirements.

4.5.4 PROTECTIVE COVER AND OVERPRESSURE SYSTEM

The protective cover and overpressure system serve no safety function and are
therefore classified as commercial material (standard quality) items. 01213464

4.5.5 CONCRETE STORAGE PADS

The concrete and reinforcing steel in the concrete storage pads are classified as safety related and are Seismic Category 1. The concrete storage pads provide structural support for the storage casks and are designed to prevent the failure of the casks due to an accident described in Chapter 8 while being moved or stored at the ISFSI.

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4.6 DECOMMISSIONING PLAN

The dry cask design concept to be utilized at the ISFSI features inherent ease and simplicity of decommissioning. At the end of its service lifetime, cask decommissioning could be accomplished by one of several options described below.

The casks, including the spent fuel stored inside, could be shipped to a suitable fuel repository for permanent storage. Depending on licensing requirements existing at the time of shipment off site, placement of the entire cask inside a supplemental shipping container or overpack would be considered.

The spent fuel could be removed from the ISFSI cask and shipped in a licensed shipping container to a suitable fuel repository. If desirable, cask decontamination could be accomplished through the use of conventional high pressure water sprays to further reduce contamination on the cask interior. The sources of contamination on the interior of the cask would be crud from the outside of the fuel pins and the crud left by the spent fuel pool water. The expected low levels of contamination from these sources could be easily removed with a high pressure water spray. After decontamination, the ISFSI cask could either be cut up for scrap or partially scrapped and any remaining contaminated portions shipped as low level radioactive waste to a disposal facility.

For surface decontamination of the ISFSI cask, chemical etching using hydrochloric acid or nitric acid can be applied to remove the contaminated surface of the cask. Alternatively, electropolishing can also be used to achieve the same result.

Cask activation analyses have been performed to quantify specific activity levels of cask materials after years of storage. The following assumptions were made:

1. The cask contains 40 reference PWR assemblies.

2. The neutron flux is assumed constant for 20 years.

3. The neutron spectrum is the same as in a PWR reactor.

The activation calculation is performed using the computer codes ORIGEN2 (Reference 14) with the total neutron fluxes taken from the radial shielding calculation performed with the XSDRN-PM code (see Section 7). The fluxes at the cask centerline, the cavity wall, the neutron shield, and the outer shell are used to irradiate the basket, the body and lid, the neutron shield, and the outer shell and protective cover, respectively. The fluxes, material compositions, and masses or irradiated material are listed in Table 4.6-1. The ORIGEN2 cross section library for PWR's at a burnup of 33,000 MWD/MTU is used. The results listed in Table 4.6-2 indicate that after 20 years irradiation and 30 days decay (to eliminate very short lived radionuclides), the total activity is less than 0.13 Ci.

To evaluate the TN-40 cask and basket for disposal, the specific activity of the isotopes listed in Tables 1 and 2 of 10CFR 61.55 is determined and compared with the limits for Class A wastes in those tables. The actual material volumes of the cask components are used to evaluate their specific activity, rather than diluting the activity over the envelope of the entire cask.

The results of the calculation, shown in Table 4.6-3, show that the TN-40 cask will be far below the specific activity limits for both long and short lived nuclides for Class A waste. Consequently, it is expected that after application of the surface decontamination process as described above, the radiation level due to activation products will be negligible and the cask could be scrapped. A detailed evaluation will be performed at the time of decommissioning to determine the appropriate mode of disposal.

Due to the leak tight design of the storage casks, no residual contamination is expected to be left behind on the concrete base pad. The base pad, fence, and peripheral utility structures will require no decontamination or special handling after the last cask is removed.

The spent fuel pool at Prairie Island Nuclear Generating Plant will remain functional until the ISFSI is decommissioned. This will allow the pool to be utilized to transfer fuel from storage casks to licensed shipping containers for shipment off site if this decommissioning option is chosen.

The volume of waste material produced incidental to ISFSI decommissioning will be limited to that necessary to accomplish surface decontamination of the casks once the spent fuel elements are removed. Furthermore, it is estimated that the cask materials will be only very slightly activated as a result of their long-term exposure to the relatively small neutron flux emanating from the spent fuel, and that the resultant activation level will be well below allowable limits for general release of the casks as noncontrolled material. Hence, it is anticipated that the casks may be decommissioned from nuclear service by surface decontamination alone, which could be performed in the cask decontamination area in the Auxiliary Building.

The costs of decommissioning the ISFSI are expected to represent a small and negligible fraction of the cost of decommissioning the Prairie Island Nuclear Generating Plant.

4.7 REFERENCES

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6. Celestial Software, Images 3-D, Version 1.6, 1984.

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9. American National Standards Institute, ANSI N14.6, American National Standard for Special Lifting Devices for Shipping Containers Weighing 11,000 Pounds or More for Nuclear Materials, 1986.

10. De Salvo, G. J. and Swanson, J.A., ANSYS Engineering Analysis System, Users Manual for ANSYS, Revision 4.3, Swanson Analysis Systems, Inc., Houston, PA, June 1987.

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15. Electric Power Research Institute, Report No. NP-4830, The Effects of Target Hardness on the Structural Design of Concrete Storage Pads for Spent-Fuel Casks, October 1986.

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17. Safety Evaluation 72-448, TN-40 Cask Weight/Storage Slab Design, May 14, 1996.

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TABLE 4.2-1
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COMPLIANCE WITH GENERAL DESIGN CRITERIA

10CFR72.122(a)	Quality Standards	The design criteria require that the structures, systems and components which are safety related be designed, fabricated and delivered to the site, according to recognized commercial codes and standards and in accordance with the NSPM QA program for equipment.
10CFR72.122(b)	Protection against environmental conditions and natural phenomena	Design basis environmental conditions and natural phenomena are defined in Chapter 2. The design criteria for the storage casks provide for prevention of criticality, maintenance of cask integrity and limitation of damage to fuel assemblies under these design bases conditions.
10CFR72.122(c)	Protection against fires and explosions	No large fire within the ISFSI is considered credible. The design criteria require that storage casks be designed to withstand extreme ambient temperatures and peak overpressure resulting from postulated nearby explosions.
10CFR72.122(d)	Sharing of structures, systems and components	The ISFSI activities will be done without jeopardizing the safe shutdown capability of the Prairie Island Nuclear Generating Plant, Units 1 and 2.
10CFR72.122(e)	Proximity of sites	The design and operation of the ISFSI result in minimal additions of risk to the health and safety of the public.
10CFR72.122(f)	Testing and maintenance of systems and components	The storage casks require minimum maintenance. The design criteria require that the storage casks be capable of being inspected and monitored.
10CFR72.122(g)	Emergency capability	Scenarios requiring emergency actions are neither considered credible, nor postulated to occur. Nevertheless, all emergency facilities at the Prairie Island Nuclear Generating Plant would be available if needed.

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TABLE 4.2-1
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COMPLIANCE WITH GENERAL DESIGN CRITERIA

10CFR72.122(h)	Confinement barriers and systems.	The design of the storage casks will ensure that the stored fuel is maintained in a safe condition. No paths for radioactive releases are considered credible. Therefore, no ventilation or off gas systems are needed.
10CFR72.122(i)	Instrumentation and control systems	No instrumentation or control systems are needed for the storage casks to perform their safety functions. Nevertheless, some monitors and alarms will be provided.
10CFR72.122(j)	Control room and control area	The ISFSI is a passive installation, with no need for operator actions. Thus no control room is needed.
10CFR72.122(k)	Utility or other services	The storage casks and the concrete storage pads are the only safety-related components at the ISFSI. There are no utility or emergency systems required to perform any safety functions at the ISFSI.
10CFR72.122(l)	Retrievability	The design of the storage casks will enable subsequent removal of the stored fuel in the Prairie Island Nuclear Generating Plant spent fuel pool and repackaging of the fuel into transportation casks. The spent fuel pool will remain operational for the life of the ISFSI.
10CFR72.124(a)	Design for criticality safety	The design criteria require that the storage casks be designed to maintain subcriticality at all times, assuming a single active or credible passive failure.
10CFR72.124(b)	Methods of criticality control	Criticality control will be provided by use of favorable geometry combined with control neutron absorbing materials. Boration of spent fuel pool water is taken into account.

TABLE 4.2-1
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COMPLIANCE WITH GENERAL DESIGN CRITERIA

10CFR72.124(c)	Criticality monitoring	Not required.
10CFR72.126(a)	Exposure control	Operations at the ISFSI will be done according to ALARA procedures. Minimal maintenance operations are needed following storage cask emplacement at the ISFSI. Cask loading, sealing, decontamination, and preparation are done at the Auxiliary Building according to health physics procedures in effect for the Prairie Island Nuclear Generating Plant.
10CFR72.126(b)	Radiological alarm systems	No radioactive releases are considered credible at the ISFSI. No safety-related systems are therefore considered appropriate.
10CFR72.126(c)	Effluent and direct radiation monitoring	Operation of the ISFSI does not result in radioactive contamination of any effluents. No safety-related monitors are needed. Direct radiation monitors will be installed around the ISFSI.
10CFR72.126(d)	Effluent control	No radioactive releases are considered credible at the ISFSI.
10CFR72.128(a)	Spent fuel and high-level radioactive waste storage and handling systems	The design criteria require the storage casks to have adequate provisions to monitor the cask performance, provide sufficient shielding to lower surface doses to below prescribed levels, maintain leak tightness under all operating and credible conditions, provide for heat removal based upon inherent design without use of active components, and maintain fuel in a safe condition. Only minimal amounts of radioactive waste are generated in the decontamination of the casks.

TABLE 4.2-1
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COMPLIANCE WITH GENERAL DESIGN CRITERIA

10CFR72.128(b)	Waste treatment	Radioactive wastes generated in the decontamination of the storage casks are processed by the Prairie Island Nuclear Generating Plant waste processing systems.
10CFR72.130	Decommissioning	Operation of the ISFSI does not result in contamination of the outside surface of the storage casks or any other ISFSI components. Therefore, there is no need for provisions to facilitate decommissioning.

TABLE 4.2-2

DYNAMIC SPRING CONSTANTS

Spring Constant	Above Water Table	Below Water Table
K_z	3.98 x 10^5 kips/ft.	5.07 x 10^5 kips/ft.
$K_x = K_y$	3.20 x 10^5 kips /ft.	3.43 x 10^5 kips/ft.
K_ψ (x axis)	1.77 x 10^8 ft.-kips/rad	2.66 x 10^8 ft.-kips/rad
K_ψ (y axis)	2.61 x 10^9 ft.-kips/rad	3.32 x 10^9 ft.-kips/rad
K_Θ	2.06 x 10^9 ft.-kips/rad	2.06 x 10^9 ft.-kips/rad

K_z = Vertical Spring Constant

$K_x = K_y$ = Sliding Spring Constant

K_ψ = Rocking Spring Constant

K_Θ = Torsional Spring Constant

TABLE 4.2-3

INDIVIDUAL LOAD CASES ANALYZED

Component/ Analysis	Loading	SAR Section	Individual Stress Results Table
CASK BODY			
Bolt Preload	Preload	4A.3	4A.3.3-1
Internal Pressure (1)	100 psig	4A.3	4A.3.3-2
External Pressure	25 psig	4A.3	4A.3.3-3
Gravity	1g	4A.3	4a.3.3-4
Lifting	3g	4A.3	4A.3.3-5
Thermal Stress	Off-Normal Temps	4A.3	4A.3.3-6
Bounding Loads(1)	2g down, 1g lateral	4A.3	4A.3.3-7
Trunnion Attachment	Attachment	4A.3.4.1	4A.3.4.1-3
Tornado Missile	Impact	4A.3.4.2	- - -
Hypothetical Accident	Impact	8.2.8.2.2	Figure 8.2-1A
LID BOLTS			
Preload	Preload Tension	4A.4.1	- - -
Thermal Effects	Differential Expansion	4A.4.2	- - -
Torquing	Preload Torsion	4A.4.3	- - -
Bending	Bending	4A.4.4	- - -
Hypothetical Accident	Impact	8.2.8.2.3	- - -
BASKET			
Bounding Side Load (2)	3g lateral	4B.6	4B.6-3 & 4
Bounding Down Load (2)	3g down	4B.7	4B.7-1
Thermal Stress	Off-Normal Temps	4B.6	4B.6-6
Hypothetical Accident	Impact	8.2.8.2.4	
TRUNNIONS			
Lifting	6g & 10g	4A.6	4A.6-2

NOTES

1. The above pressures and bounding loads conservatively envelop all possible pressure effects as well as tornado wind load, flood water load and seismic load.

2. The bounding loads selected for basket evaluation are extremely conservative. These loads are more severe than any loads that will actually be applied to the basket.

TABLE 4.2-4

CONTAINMENT VESSEL STRESS LIMITS (1)

Classification	Stress Intensity Limit (4)
DESIGN	
P_m	S_m
P_l	$1.5\ S_m$
$(P_m \text{ or } P_l) + P_b$	$1.5\ S_m$
Shear Stress	$0.6\ S_m$
LEVEL A	
Classification	**Stress Intensity Limit (4)**
$(P_m \text{ or } P_l) + P_b + Q$	$3\ S_m$
$(P_m \text{ or } P_l) + P_b + Q + F$	S_a
LEVEL D (2)	
Classification	**Stress Intensity Limit (3) (4)**
P_m	$2.4\ S_m$ or $0.7\ S_u$
P_l	$3.6\ S_m$ or S_u
$(P_m \text{ or } P_l) + P_b$	$3.6\ S_m$ or S_u
Shear Stress	$0.42\ S_u$

NOTES

1. Quantities are as defined in ASME Code, Section III, Subsection NB.
2. Limits are in accordance with ASME Code, Section III, Appendix F.
3. Whichever is lower.
4. When using materials data from Section III Appendix Tables other than I-1.0, S values may be substituted for S_m values in these expressions

TABLE 4.2-5

CONTAINMENT BOLT STRESS LIMITS (1)

DESIGN CONDITIONS

Classification	Stress Intensity Limit	ASME Section (1)
P_m (Tensile)	S_m	NB-3231

LEVEL A

Classification	Stress Intensity Limit	ASME Section (1)
P_m (Tensile)	$2S_m$	NB-3232.1
$P_l + P_b$ (Tensile and Bending)	$3S_m$	NB-3232.2
Combined	$3S_m$	NB-3232.2

LEVEL D (2)

Classification	Stress Intensity Limit
P_m (Tensile)	Smaller of S_y or 0.7 S_u
$P_l + P_b$ (Tensile and Bending)	S_u
Shear	Smaller of 0.42 S_u or 0.6 S_y
Combined Shear & Tension	$\frac{(ft)^2}{(Ftb)^2} + \frac{(fv)^2}{(Fvb)^2} \leq 1$

NOTES

1. Terms are as defined in ASME Code, Section III, Subsection NB, Paragraph 3230.

2. Limits are in accordance with ASME Code, Section III, Appendix F.

TABLE 4.2-6

NON CONTAINMENT STRUCTURE STRESS LIMITS (1)

DESIGN

Classification	Stress Intensity Limit (2) (4)
P_m	S_m or 0.67 S_y
P_l	1.5 S_m or S_y
(P_m or P_l) +P_b	1.5 S_m or S_y (3)
Shear Stress	0.65 S_m or 0.5 S_y

LEVEL A

Classification	Stress Intensity Limit (2, 3 & 4)
(Pm or P_l) + P_b +Q	3 S_m or 2 S_y

LEVEL D

Classification	Stress Intensity Limit
P_m	0.7 S_u
P_l	S_u
(P_m or P_l) +P_b	S_u
Shear Stress	0.6 S_u

NOTES

1. Quantities are as defined in ASME Code, Section III, Subsection NB.
2. Whichever is higher.
3. These limits may be exceeded for non containment structure if the resulting deflection can be accommodated.
4. When using materials data from Section III Appendix Table other than I-1.0, S values may be substituted for S_m values in these expressions.

TABLE 4.2-6a

BASKET STRESS LIMITS (1)

DESIGN (4)	
Classification	**Stress Intensity Limit** (4)
P_m	S_m
P_l	$1.5\ S_m$
$(P_m + P_l) + P_b$	$1.5\ S_m$
Shear Stress	$0.6\ S_m$
LEVEL A	
Classification	**Stress Intensity Limit** (4)
$(P_m \text{ or } P_l) + P_b + Q$	$3\ S_m$
$(P_m \text{ or } P_l) + Pb + Q + F$	S_a
LEVEL D (2) (4) (5)	
Classification	**Stress Intensity Limit** (3) (4)
P_m	$2.4\ S_m$ or $0.7\ S_u$
P_l	$3.6\ S_m$ or S_u
$(P_m \text{ or } P_l) + P_b$	$3.6\ S_m$ or S_u
Shear Stress	$0.42\ S_u$

NOTES

1. Quantities are as defined in ASME Code, Section III, Subsection NB.

2. Limits are in accordance with ASME Code, Section III, Appendix F.

3. Whichever is lower.

4. Under sustained primary loads the strength of the 6061-T6 basket plates shall not be considered.

5. For short duration impact loading the strength of the 6061-T6 basket plates may be considered. For these conditions (Level D Impact) the value of $2/3\ S_y$ may be substituted for S_m.

TABLE 4.2-7

LOAD COMBINATIONS FOR CASK BODY

	Bolt Preload	1 G Down	Internal Pressure	Ext. Pressure	Temp	3 G On Trunnions	Trunnion Local	Seismic Tornado of Flood	Tornado Missiles
Design Des (1)	X	X	X					X	
Des (2)	X		X			X	X		
Des (3)	X	X		X				X	
Level A A (1)	X	X	X		X			X	
A (2)	X		X		X	X	X		
A (3)	X	X		X	X			X	
Level D D (1)	X	X	X					X	X
D (2)	X	X		X				X	X

TABLE 4.2-8

COMPARISON OF ACTUAL WITH ALLOWABLE STRESS INTENSITY CONTAINMENT VESSEL

Service Condition	Component	Location (Figure 4.2-6)	Load Combination (Table 4.2-6)	Stress Category	Maximum* Stress Intensity (psi)	Allowable Stress Intensity (psi)
DESIGN	Shell	-	-	P_m	-	16,200 (S_m)
		3	Design (1)	$P_l + P_b$	13,865	24,300 (1.5 S_m)
	Flange	-	-	P_m	-	17,500 (S_m)
		10	Design (2)	$P_l + P_b$	6,049	26,250 (1.5 S_m)
	Lid	-	-	P_m	-	17,500 (S_m)
		12	Design (2)	$P_l + P_b$	6,535	26,250 (1.5 S_m)
LEVEL A	Shell	3	A (1)	$P_l + P_b + Q$	22,132	48,600 (3 S_m)
	Flange	10	A (2)	$P_l + P_b + Q$	8,184	52,500 (3 S_m)
	Lid	12	A (2)	$P_l + P_b + Q$	8,060	52,500 (3 S_m)
LEVEL D	Shell	-	-	P_m	Same as	38,880 (2.4 S_m)
		-	-	$P_l + P_b$	Level A	58,320 (3.6 S_m)
	Flange	-	-	P_m	Same as	42,000 (2.4 S_m)
		-	-	$P_l + P_b$	Level A	63,000 (3.6 S_m)
	Lid	12	D(1)	$P_l + P_b + Q$	6,140	42,000 (2.4 S_m)
		12	D(1)	$P_l + P_b + Q$	47,425	63,000 (3.6 S_m)

*Maximum values from Tables 4A.3.5-1 through 8

TABLE 4.2-9

SUMMARY OF MAXIMUM STRESS INTENSITIES AND ALLOWABLE STRESS LIMITS FOR THE CONTAINMENT VESSEL

Service Condition	Location	Maximum Stress Intensity (psi)	Allowable (psi)	Margin of Safety
Design	Containment Shell Near Bottom	$P_l + P_b = 13{,}865$	$1.5S_m=24{,}300$	0.75
Level A	Containment Shell Near Bottom	$P_l + P_b + Q = 22{,}132$	$3S_m=48{,}600$	1.20
Level D	In the Lid Near the Center	$P_l + P_b = 47{,}425$	$3.6S_m=63{,}000$	0.33

TABLE 4.2-10

COMPARISON OF ACTUAL WITH ALLOWABLE STRESS INTENSITY GAMMA SHIELDING.

SERVICE CONDITION	COMPONENT	LOCATION (Figure 4.2-6)	LOAD COMBINATION (Table 4.2-6)	STRESS CATEGORY	MAXIMUM* STRESS INTENSITY (psi)	ALLOWABLE STRESS INTENSITY (psi)
Design	Cylinder	--	-	P_m	-	21,866 (2/3 S_y)
		15	Design 2	$P_l + P_b$	830	32,800 (S_y)
	Bottom	-	-	P_m	-	21,866 (2/3 S_y)
		14	Design 2	$P_l + P_b$	3,670	32,800 (S_y)
	Welds	20	Design 1	P_m	14,574	21,866 (2/3 S_y)
Level A	Cylinder	16	A (2)	$P_l + P_b + Q$	1,854	65,600 (2 S_y)
	Bottom	13	A (2)	$P_l + P_b + Q$	5,779	65,600 (2 S_y)
	Welds	18	A (1)	$P_l + P_b + Q$	15,953	65,600 (2 S_y)
Level D	Cylinder	-	-	P_m	-	49,000 (0.7 S_u)
		15	D (1)	$P_l + P_b$	12,638	70,000 (S_u)
	Bottom	-	-	P_m	Same as	49,000 (0.7 S_u)
		-	-	$P_l + P_b$	Level A	70,000 (S_u)
	Welds	20	D (1)	P_m	6,140	49,000 (0.7 S_u)
		20	D (1)	$P_l + P_b$	55,519	70,000 (S_u)

*Maximum values from Tables 4A3.5-1 through 8.

TABLE 4.2-11

SUMMARY OF MAXIMUM INTENSITY AND ALLOWABLE STRESS LIMITS FOR LID BOLTS

	STRESS CATEGORY				**ALLOWABLE STRESS INTENSITY (PSI)**				
Service Condition	Tensile	Tensile + Bending	Shear	Combined	Tensile	Tensile + Bending	Stress Shear	Combined	Margin of Safety
Design (2)	25,000				S_m =31,900				0.28
Level A	51,000				2 S_m =63,800				0.25
		75,457				$3S_m$=95,700			0.27
			14,071	80,534(1)			3 S_m =95,700		0.19
Level D	---------	---------------	---------	Same as Level A	-----------	-----------------	---------------	---------------	----------------

(1) Combined Stress = $\sqrt{\sigma^2 + 4\tau^2}$

(2) "DESIGN" condition remains unchanged.
Applied preload need not be included.

TABLE 4.2-12

COMPARISON OF ACTUAL WITH ALLOWABLE STRESS INTENSITY IN BASKET

SERVICE CONDITION	COMPONENT /STRESS CATEGORY	STRESS INTENSITY (PSI)	ALLOWABLE STRESS INTENSITY (PSI) (3)
DESIGN, LEVEL D (Bounding 3 g Lateral Load)	304 SS Fuel Boxes		17,000 (S_m)
	P_m	1,571 (1)	25,700 (1.5 S_m)
	$P_l + P_b$	6,056(1)	
	Plug Weld Stress		
	$P_m + Q\ (2\tau)$	172(1)	17,000(S_m)
LEVEL A (3 g Lateral Load plus Thermal)	304 SS Fuel Boxes		51,000 (3 S_m)
	$P_l + P_b + Q$	49,036(2)	
	Plug Weld Stress		
	$P_m + Q(2\tau)$	50,902(2)	51,000 (3 S_m)

NOTE

1. These primary stresses have been calculated on an elastic basis neglecting the structural contribution from the aluminum.

2. These primary plus secondary stresses have been determined as permitted by the ASME Code where the structural action is calculated on a plastic basis. The $3S_m$ limit has been met. ASME permits $3S_m$ to be exceeded in this case since thermal ratcheting and fatigue cannot occur. See Section 4B.6.

3. These allowable stresses are based on the maximum metal temperature of 530°F. Some of the maximum stresses occur in cooler portions of the basket where allowables are higher.

TABLE 4.2-13

COMPARISON OF MAXIMUM LIFTING TRUNNION STRESS INTENSITIES WITH ALLOWABLES

LOAD	MAXIMUM STRESS INTENSITY(PSI)		ALLOWABLE STRESS (psi)
	LIFTING SHOULDER	ATTACHMENT WELD	
LIFTING INERTIA LOAD 6 g	20,320	28,630	S_y=31,900
LIFTING INERTIA LOAD 10 g	33,866	47,718	S_u=70,000

TABLE 4.2-14

COMPARISON OF MAXIMUM TURNING TRUNNION STRESS INTENSITIES WITH ALLOWABLES

LOAD	MAXIMUM STRESS INTENSITY(PSI)		ALLOWABLE STRESS (psi)
	LIFTING SHOULDER	ATTACHMENT WELD	
LIFTING INERTIA LOAD 6 g on 4 Trunnions	18,174	27,953	31,900
LIFTING INERTIA LOAD 10 g on 4 Trunnions	30,294	46,589	70,000

TABLE 4.2-15

COMPARISON OF MAXIMUM STRESS INTENSITY WITH ALLOWABLES IN OUTER SHELL

LOAD	MAXIMUM STRESS INTENSITY (PSI)	ALLOWABLE STRESS (psi)
	TOP CLOSURE PLATE	
25 psi + 3 g HANDLING VERTICAL INTERTIA LOAD	21,117	S_y=26,600

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TABLE 4.5-1

CLASSIFICATION OF STRUCTURES, COMPONENTS AND SYSTEMS

SAFETY RELATED	AUGMENTED QUALITY	COMMERCIAL MATERIAL (STANDARD QUALITY)
Containment Vessel	Lid Gaskets	Protective Cover
Cask Body Shell		
Cask Body Bottom	Lid Penetration covers, Bolts Gaskets	Overpressure System
Lid		Transport Vehicle
Lid Bolts		
Basket Assembly	Port Covers, Bolts, Gaskets	ISFSI Buildings
Trunnions	Neutron Shield	Electrical Power
		Lighting
		Receptacles
Concrete Pads	Body Shielding	
Concrete		
Reinforcing Steel	Lid Shielding	
	Security System	

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TABLE 4.6-1

DATA FOR TN-40 ACTIVATION ANALYSIS

ZONE	FLUX N/CM^3/S	MATERIAL	MASS kg (approx)	COMPOSITION WT %
Basket	7.55E5	Stainless Steel (SA-240 TP304)	2,770	Mn 2.0 Cr 19.0 Ni 9.5 Si 0.75 Fe 68.75
		Aluminum (6061-T6)	2,906	Si 0.6 Mg 1.0 Cr 0.2 Cu 0.3 Al 97.9
Body, Lid, Basket Supports	3.87E5	Carbon Steel (SA-105)	56,070	Fe 98.5 Mn 1.0 C 0.2 Si 0.3
		Aluminum	1,135	See Above
		"Containment" Steel (SA-203)	12,260	Si 0.35 Mn 0.9 Ni 3.75 C 0.2 Fe 94.8
		Stainless Steel	681	See Above
Neutron Shield	1.07E4	Resin	4,858	H 5.05 B 1.05 C 35.13 O 41.73 Al 14.93 Zn 2.11
		Aluminum	908	See Above
Shell, cover	1.11E3	Carbon Steel (SA-516 Gr 55)	4,131	Mn 0.7 Fe 99.3

TABLE 4.6-2

RESULTS OF ORIGEN2 ACTIVATION CALCULATION

NUCLIDE	ACTIVITY (Ci)				
	Basket	Body	N-Shield	Shell	TOTAL
H3	-----	-----	2.12E-10	-----	2.12E-10
C14	-----	2.02E-10	5.12E-10	3.49E-14	5.12E-10
Cr51	3.81E-3	6.35E-4	-----	-----	4.44E-3
Mn54	4.61E-4	8.36E-3	-----	-----	8.82E-3
Fe55	5.06E-3	9.08E-2	-----	1.57E-5	9.59E-2
Fe59	9.36E-5	1.69E-3	-----	-----	1.78E-3
Co58	5.88E-4	6.00E-4	-----	-----	1.19E-3
Co60	8.32E-6	8.31E-6	-----	-----	1.66E-5
Ni63	3.64E-4	3.72E-4	-----	-----	7.36E-4
Ni59	3.17E-6	3.24E-6	-----	-----	6.41E-6
Zn65	-----	-----	1.12E-05	-----	1.12E-5
TOTAL					1.13E-1

NOTE

1. Only nuclides with activity greater than 10^{-5} curie and those nuclides listed in 10CFR61.55 are reported here.

TABLE 4.6-3

COMPARISON OF TN-40 ACTIVITY WITH CLASS A WASTE LIMITS

Long lived Isotopes, 10CFR61.55, Table 1

Zone	Volume m^3	Nuclide	Specific Activity Ci/m^3	Limit Ci/m^3
Basket	1.43	C14	-----	80
		Ni59	3.13E-06	220
Body	9.27	C14	2.17E-11	80
		Ni59	3.49E-07	
N-Shield	3.37	C14	1.52E-10	80
Shell	0.53	C14	6.59E-14	80

Short Lived Isotopes, 10CFR61.55, Table 2, Column 1

Zone	Nuclide	Specific Activity Ci/m^3	Limit Ci/m^3
Basket	Co60	5.82E-06	700
	Ni63	2.55E-04	35
	T5*	7.00E-03	700
Body	Co60	8.96E-07	700
	Ni63	4.01E-05	35
	T5*	1.10E-02	700
N-Shield	H3	6.28E-11	40
	T5*	8.56E-05	700
Shell	T5*	2.96E-05	700

*Nuclides with half lives less than 5 years:
Cr52, Mn54, Fe55, Fe59, Co58, Zn65.

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FIGURE 4.1-1, REV. 2

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Figure 4.2-1

CONCRETE PAD

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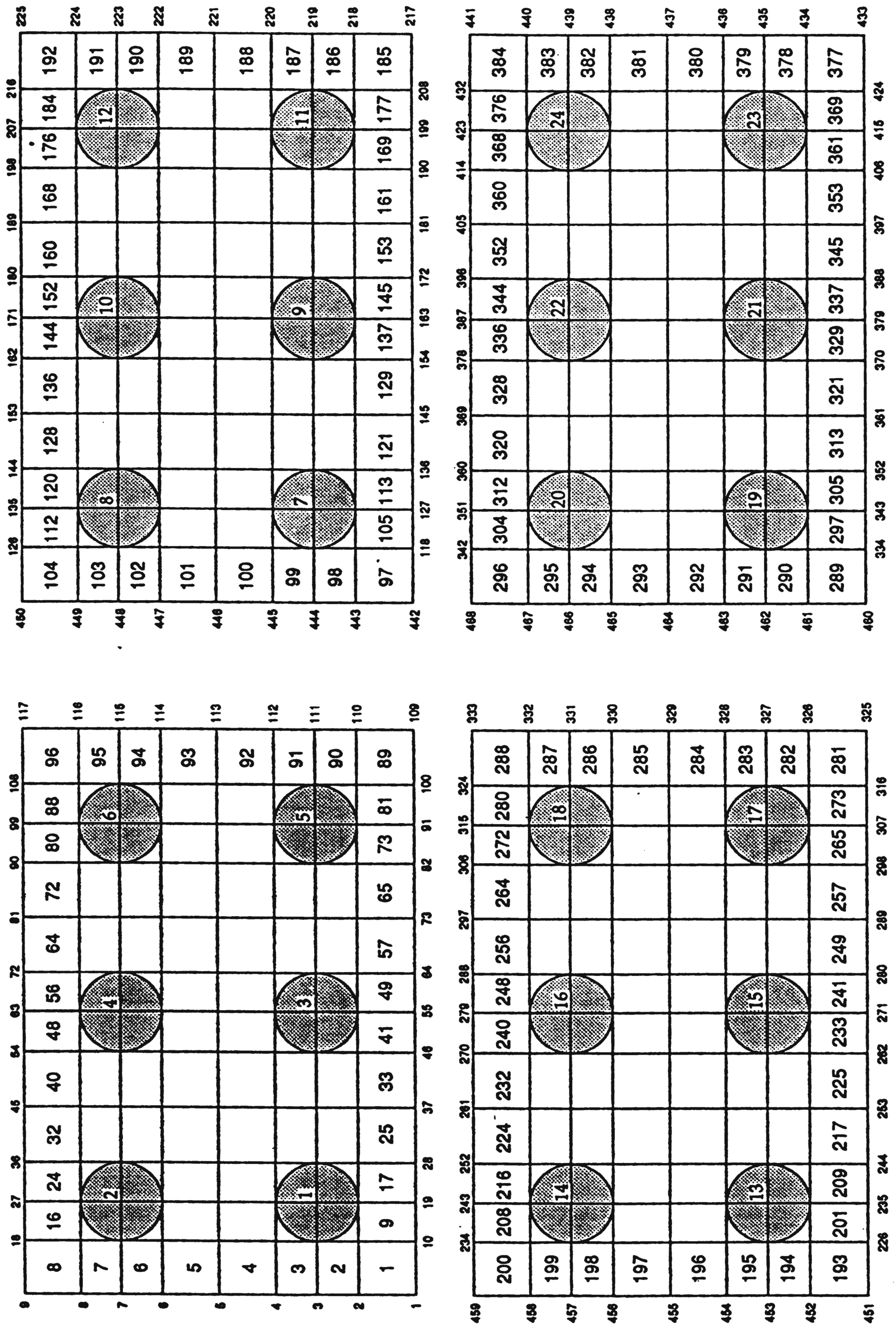
Figure Withheld Under 10 CFR 2.390

CONCRETE PAD			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG4.2-1_REV_3
	PAGE. NO.	DATE: 04-05-06	



Note: Nodes 442 to 450 are coincident with nodes 109 to 117, nodes 451 to 459 are coincident with nodes 217 to 225 and nodes 460 to 468 are coincident with nodes 325 to 333.

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Figure 4.2-1a

IMAGES - 3D COMPUTER MODEL

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IMAGES-3D COMPUTER MODEL			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG4.2-1A_REV_3
	PAGE. NO.	DATE: 04-5-06	

FIGURE 4.2-2, REV. 1

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FIGURE 4.2-3, REV. 1

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FIGURE 4.2-4, REV. 1

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FIGURE 4.2-5, REV. 1

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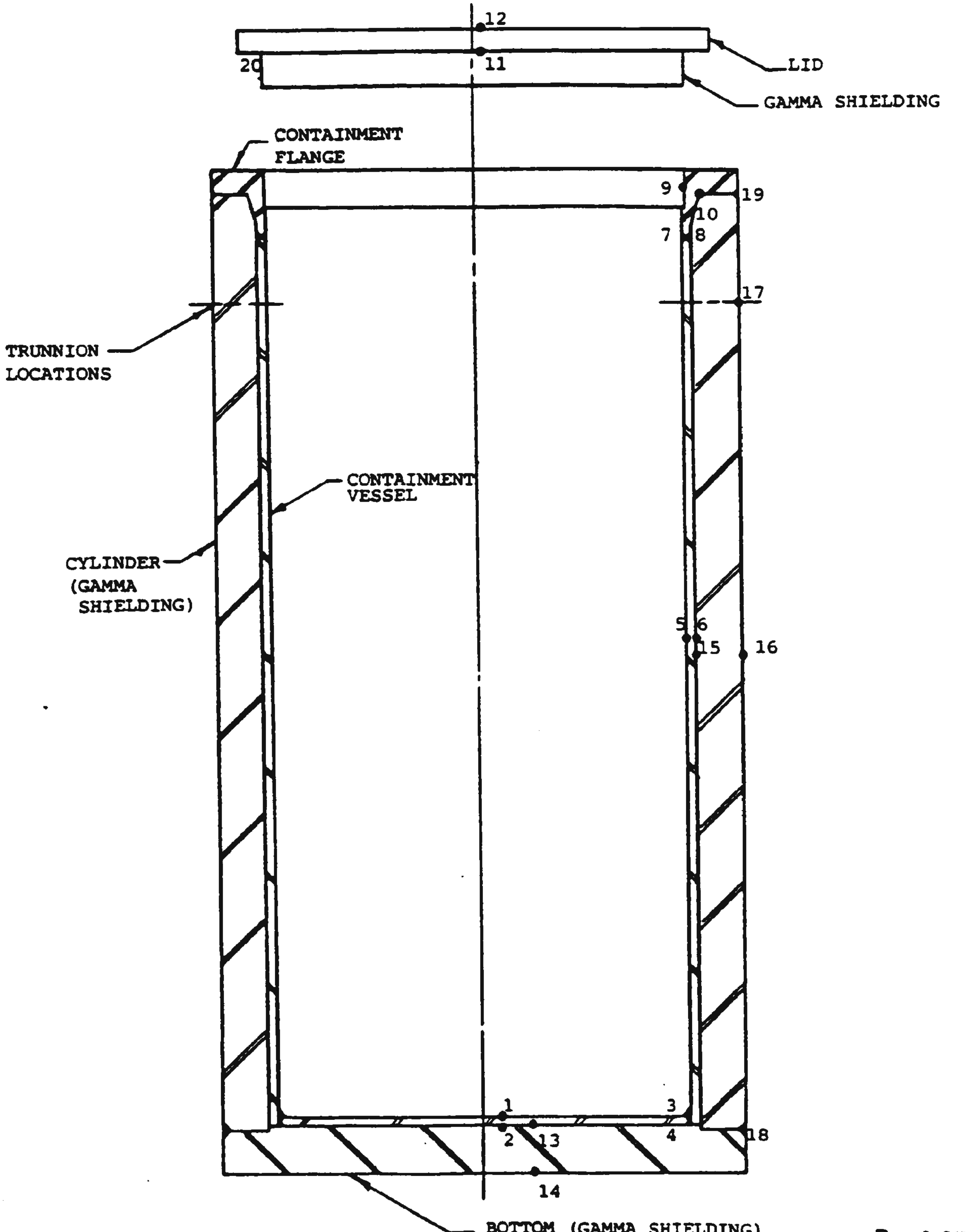

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Figure 4.2-6

STANDARD REPORTING LOCATIONS FOR CASK BODY (TN-40 CASK)

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STANDARD REPORTING LOCATIONS FOR CASK BODY (TN-40 CASK)

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG4.2-6_REV_0
	PAGE. NO.	DATE: 04-5-06	

Figure Withheld Under 10 CFR 2.390

Figure 4.3-1

TRANSPORT VEHICLE - SIDE VIEW

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TRANSPORT VEHICLE - SIDE VIEW			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG4.3-1_REV_3
	PAGE. NO.	DATE: 04-5-06	

Figure Withheld Under 10 CFR 2.390

Figure 4.4-1

LOAD PATH FOR SPENT FUEL CASK - PLAN VIEW

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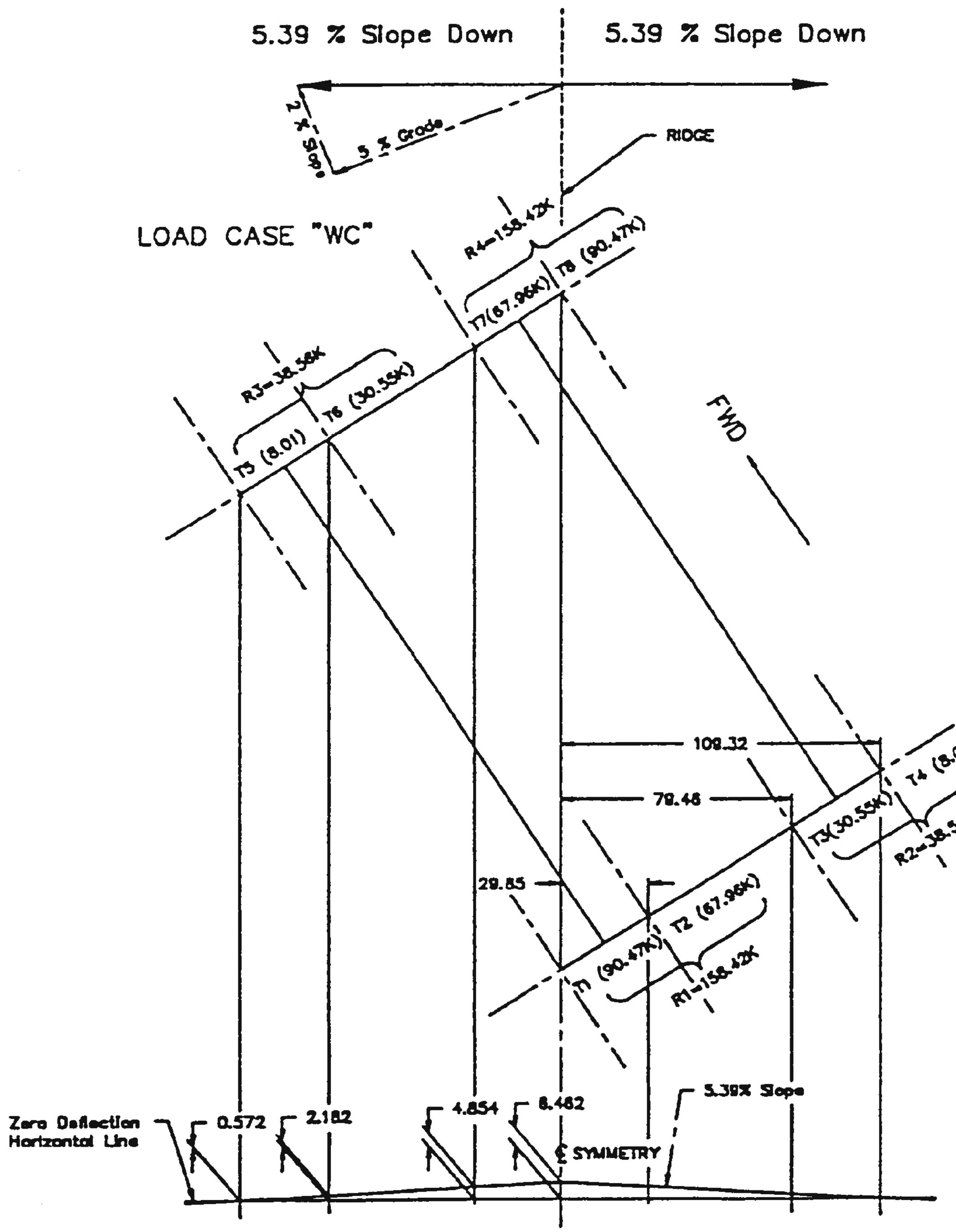
LOAD PATH FOR SPENT FUEL CASK - PLAN VIEW			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG4.4-1_REV_0
	PAGE. NO.	DATE: 04-19-06	



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NOTE:

1. REF: EDERER CALCULATION NO. D-15372, REV. B, p. 1-24

Figure 4.3-3

CYCLIC LOAD, TIRE VERTICAL REACTIONS (RIDGE UNDER T1 AND T8)

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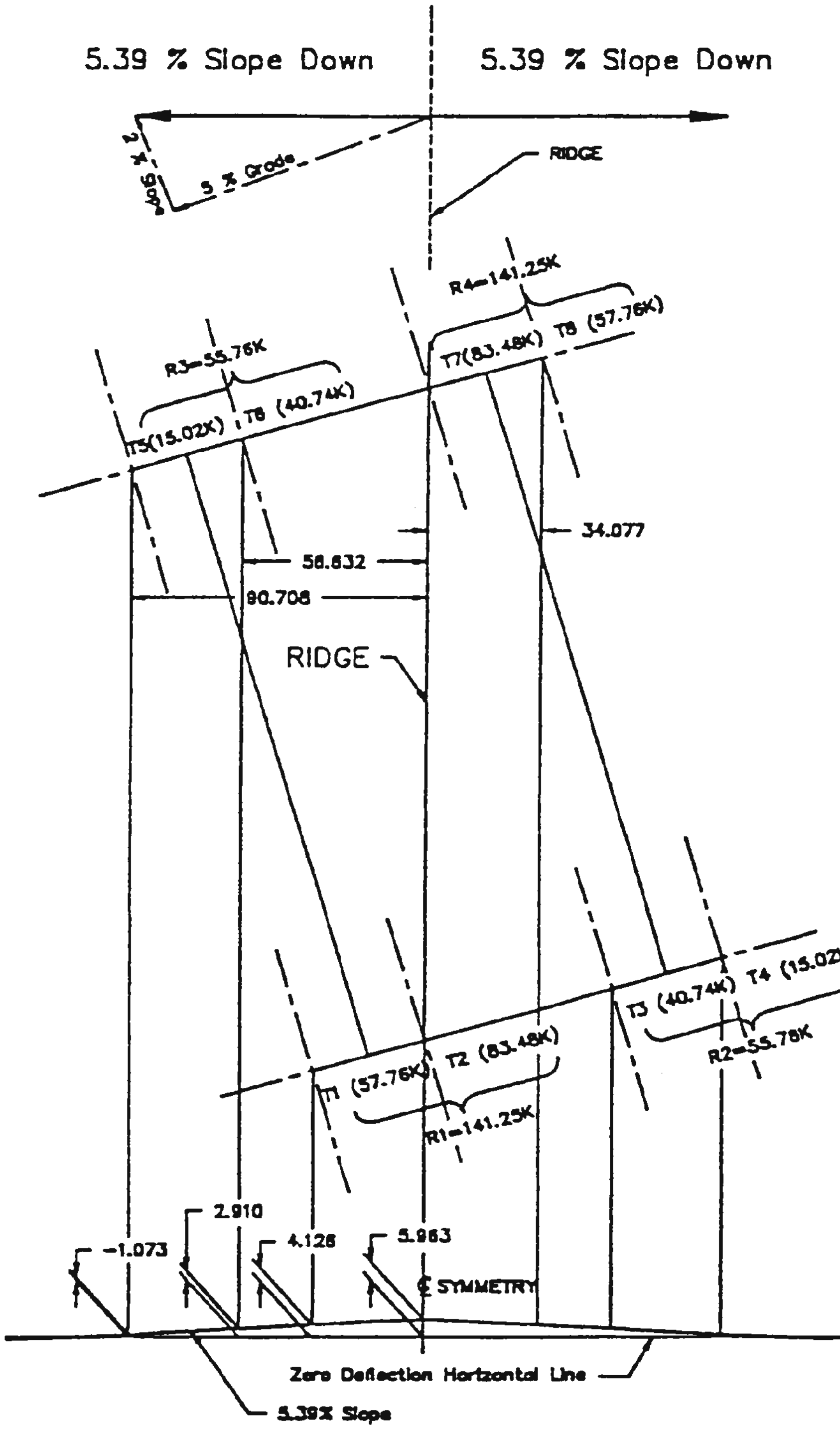
CYCLIC LOAD, TIRE VERTICAL REACTIONS (RIDGE UNDER T1 AND T8)

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG4.3-3_REV_3
	PAGE. NO.	DATE: 04-18-06	



Rev 3 4/94

NOTE:

1. REF: EDERER CALCULATION NO. D-15372, REV. B, p. 1-27

Figure 4.3-4

CYCLIC LOAD, TIRE VERTICAL REACTIONS (RIDGE UNDER T2 AND T7)

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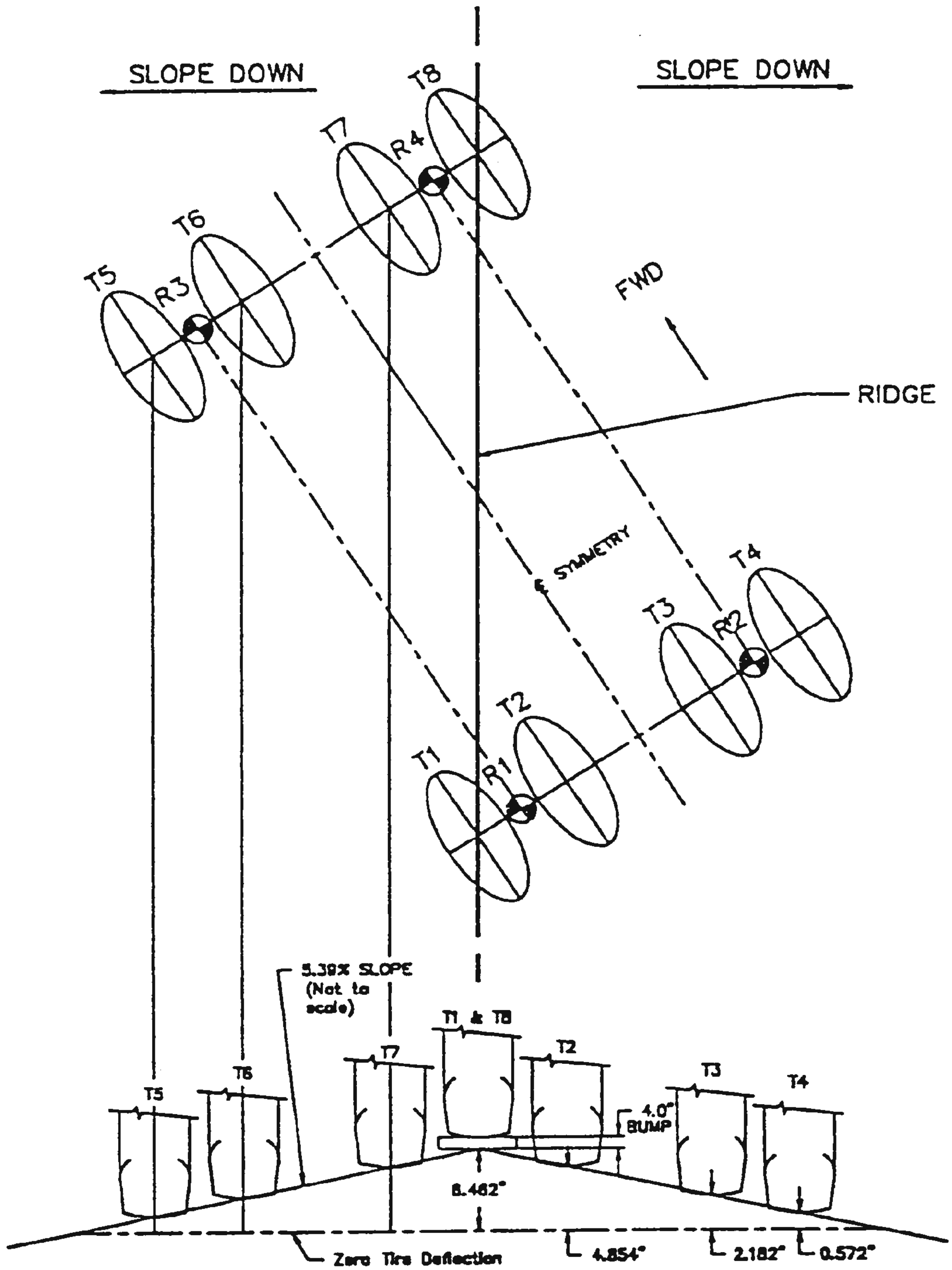

Rev 3 4/94

NOTE:

1. REF: EDERER CALCULATION NO. D-15372, REV. B, p.1-29

Figure 4.3-5

WORST CASE LOAD, TIRE VERTICAL REACTIONS (4" BUMP UNDER T1 AND T8)

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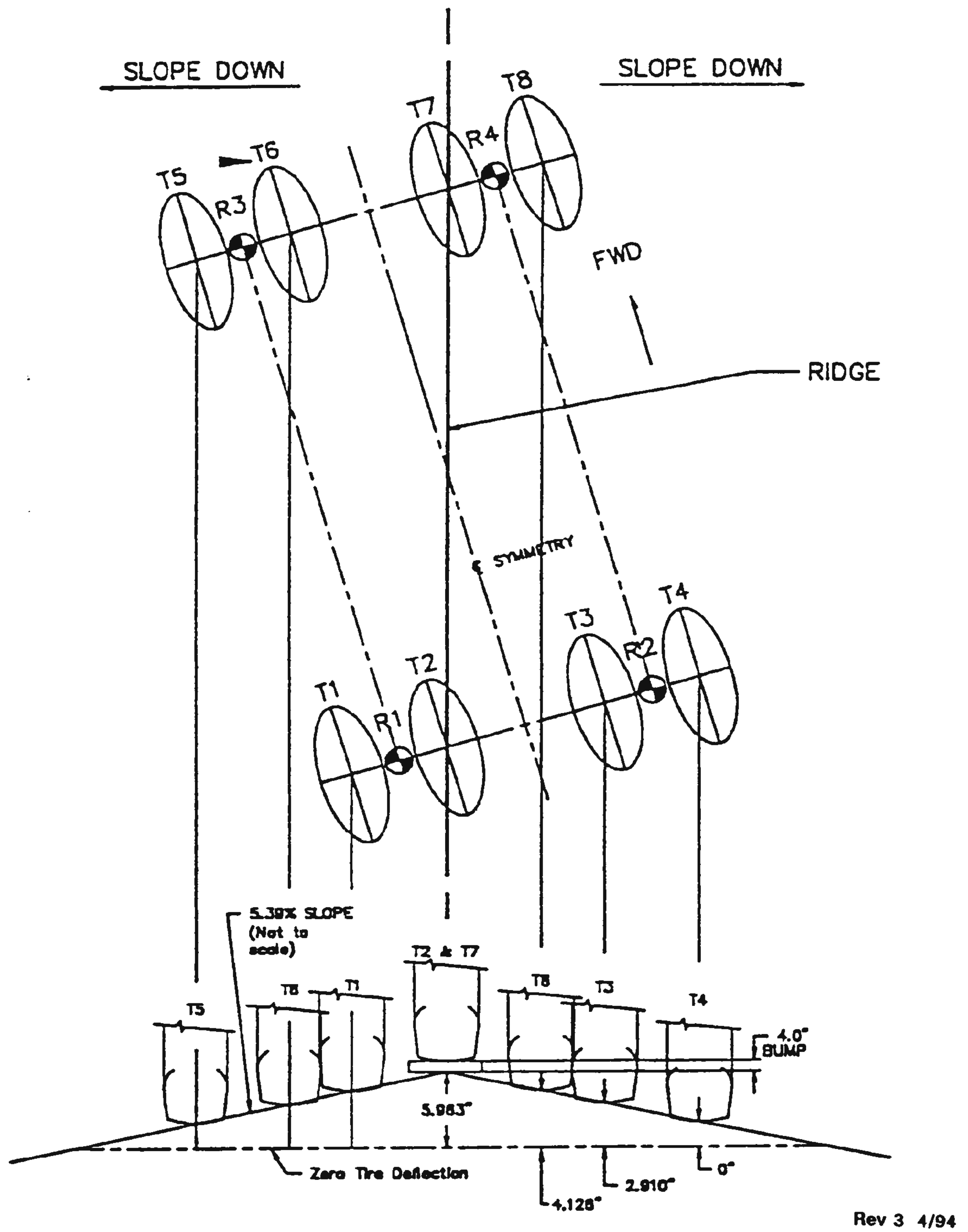
WORST CASE LOAD, TIRE VERTICAL REACTION (4" BUMP UNDER T1 AND T8)

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG4.3-5_REV_3
	PAGE. NO.	DATE: 04-19-06	



NOTE:

1. REF: EDERER CALCULATION NO. D-15372, REV. B, p.1-33

Figure 4.3-6

WORST CASE LOAD, TIRE VERTICAL REACTIONS (4" BUMP UNDER T2 AND T7)

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WORST CASE LOAD, TIRE VERTICAL REACTION (4" BUMP UNDER T2 AND T7)

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG4.3-6_REV_3
	PAGE. NO.	DATE: 04-19-06	

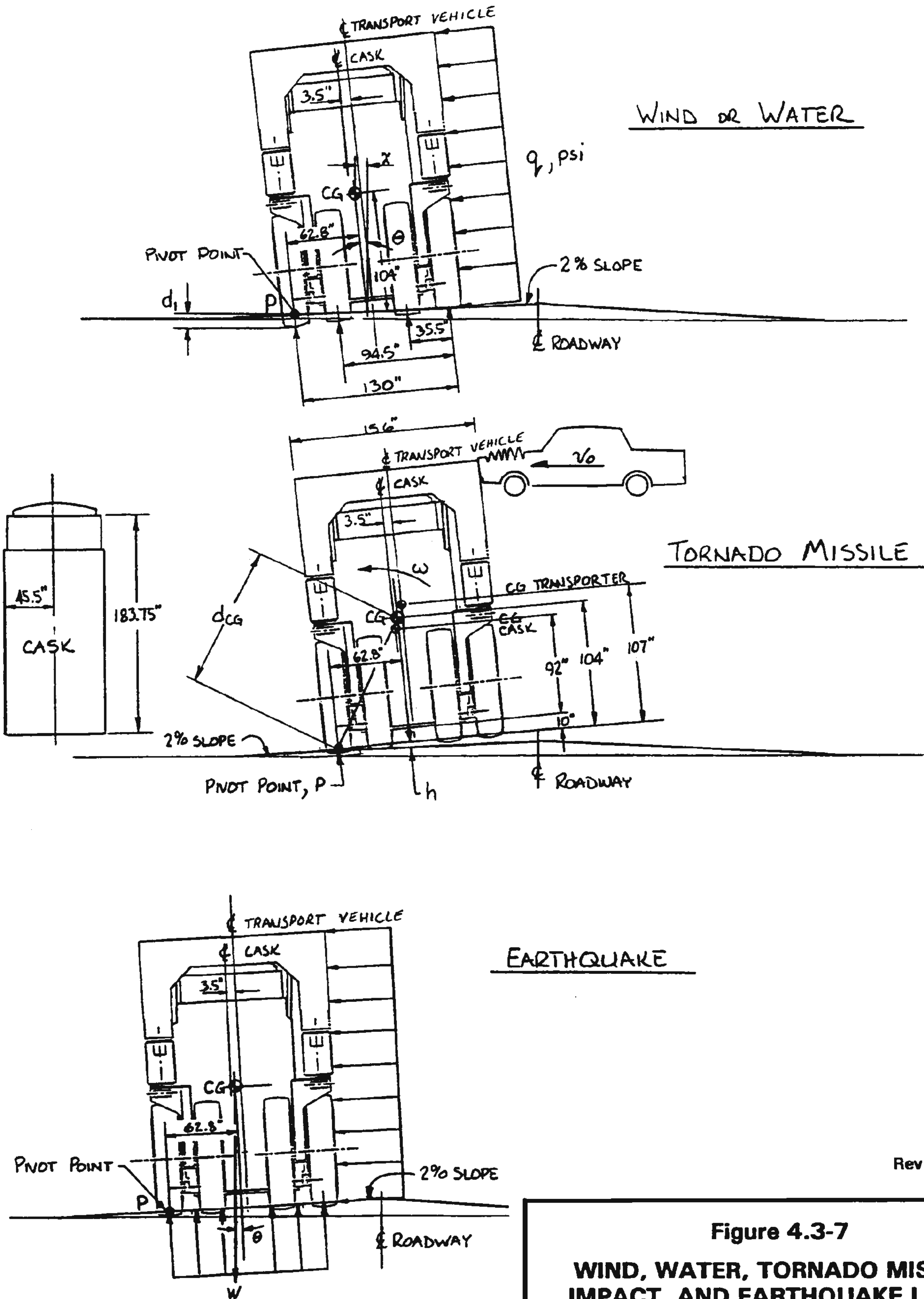

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Figure 4.3-7

WIND, WATER, TORNADO MISSILE IMPACT, AND EARTHQUAKE LOADS

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WIND, WATER, TORNADO MISSILE IMPACT, AND EARTHQUAKE LOADS

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG4.3-7_REV_3
	PAGE. NO.	DATE: 04-19-06	

Figure Withheld Under 10 CFR 2.390

Figure 4.4-2

LOAD PATH FOR SPENT FUEL CASK-ELEVATION A (LOOKING EAST)

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LOAD PATH FOR SPENT FUEL CASK - ELEVATION A (LOOKING EAST)			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG4.4-2_REV_0
	PAGE. NO.	DATE: 04-19-06	

Figure Withheld Under 10 CFR 2.390

Figure 4.4-3

LOAD PATH FOR SPENT FUEL CASK - ELEVATION B (LOOKING SOUTH)

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SAFETY ANALYSIS REPORT

LOAD PATH FOR SPENT FUEL CASK - ELEVATION B (LOOKING SOUTH)			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS PAGE. NO.	REVISION: 0 DATE: 04-19-06	FIG4.4-3_REV_0

Figure Withheld Under 10 CFR 2.390

Figure 4.4-3

LOAD PATH FOR SPENT FUEL CASK - ELEVATION B (LOOKING SOUTH)

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

LOAD PATH FOR SPENT FUEL CASK - ELEVATION B (LOOKING SOUTH)			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS PAGE. NO.	REVISION: 0 DATE: 04-19-06	FIG4.4-3_REV_0

APPENDIX 4A

STRUCTURAL ANALYSIS OF THE TN-40 CASK BODY

4A.1 INTRODUCTION

This appendix presents the structural analysis of the TN-40 storage cask body which consists of the cask body, the trunnions and the outer shell. Analyses are performed to evaluate the various cask components under the loadings described in **Section 3.2.** Additional analyses are provided in Chapter 8 to evaluate the cask body under hypothetical accident loadings.

The detailed calculations for the cask body are presented in Section 4A.3 and the lid bolt analysis is reported in a separate Section 4A.4. The calculations for the trunnions and outer shell are reported in Sections 4A.6 and 4A.7, respectively.

The design criteria used in the analyses of the cask components are in accordance with the ASME Code, Section III, Subsection NB, (Reference 1). The material properties used are those obtained from the Code appendices (Reference 2). Key dimensions of the storage cask are shown in Figure 4A.1-1.

4A.2 MATERIALS PROPERTIES DATA

This section provides the mechanical properties of materials used in the structural evaluation of the TN-40 storage cask. Table 4A.2-1 lists the materials selected, the applicable components, and the minimum yield, ultimate, and design stress values specified by the ASME Code. All values reported in Table 4A.2-1 are for metal temperature up to 100°F. For higher temperatures, the temperature dependency of the material properties is reported in Table 4A.2-2.

Table 4A.2-3 is provided to summarize thermal analysis results from Chapter 3 which support the selection of cask body component design temperatures for structural analysis purposes.

4A.3 CASK BODY STRUCTURAL ANALYSIS

4A.3.1 DESCRIPTION

The cask body as shown in Figure 4A.1-1 consists of:

1. A 1 1/2-in. thick inner vessel with a welded flat bottom, a flange at the top, and a lid bolted to the flange by 48, 1 1/2" diameter high strength bolts and sealed with two metallic o-rings. This is the containment vessel, i.e. the primary containment boundary of the cask.

2. A thick cylindrical vessel with a welded flat bottom surrounding the containment. This vessel and a lid disc welded to the lid inner surface provide the gamma shielding.

The lid and the flange are carbon steel forgings as are the gamma shielding components. The cask body is designed as a Class 1 component in accordance with the rules of the ASME Code. A static, linear elastic analysis is performed on the cask body so that combinations of loads can be obtained by superposition of individual loads. The stresses and deformations due to the applied loads are generally determined using the ANSYS computer program (Reference 4). A 2D ANSYS Model was specifically developed for this purpose. Exceptions include the analyses of the local effects at the trunnions and of the lid bolts.

4A.3.2 ANSYS CASK MODEL

A two-dimensional ANSYS model is used to evaluate the stresses in the cask body due to the individual load cases. The finite elements used in the model are the axisymmetric shell element, STIF 61, and the axisymmetric harmonic element, STIF 25. Both of these elements consider axisymmetric and non-axisymmetric loadings.

The cylindrical containment shell and bottom are modeled using STIF 61 elements. The remainder of the cask body is modeled with STIF 25 elements except the lid bolts for which the two dimensional elastic beam, STIF 3 is used. The finite element model of the cask body is shown in Figure 4A.3-1.

Figure 4A.3-2 shows an enlarged view of the bottom corner with the weld joining the gamma shielding flat bottom to cylinder simulated by coupling nodes 67-203 and 63-202.

The weld connecting the gamma shielding cylinder to the containment flange is simulated by coupling nodes 348-178 and 349-182 as shown in Figure 4A.3-3. Also shown in this figure are the lid bolts connecting the lid to the containment flange. The connection is simulated by coupling nodes 800, 801 and 802 of the bolts to the corresponding nodes 159, 160, and 161 of the flange; and nodes 804, 805, and 806 of the bolts to the corresponding nodes 571, 567 and 575 of the lid. In this manner the threaded portion of the bolt is fixed to the flange while the bolt head is fixed to the top surface of the lid. In order to prevent the lid from moving into the flange, nodes 157 and 564 are also coupled in the axial or Y direction. The enlarged view in Figure 4A.3-4 shows the coupling of nodes 458-552 and 463-556 which simulates the weld connecting the containment lid to the gamma shielding disc.

The pairs of nodes listed above, with the exception of nodes 157-564, are coupled in the X, Y and Z directions. The coupling of nodes 157-564 is in the Y direction only and is accomplished using a constraint equation. The reaction at the nodes is monitored during the analysis to insure that tensile forces between the cylinder and the lid are not developed.

Appropriate boundary conditions are applied to prevent rigid body motion and to show that the system of forces applied to the cask in each of the individual load cases is in equilibrium. Generally a node at the center of the vessel bottom is held in all directions and one at the center of the lid is held in the X and Z directions.

4A.3.3 INDIVIDUAL LOAD CASES

Individual load cases are evaluated to determine the stress contribution due to specific individual loads. Stress results are reported in this Appendix for each individual load. Since the individual load cases are linearly elastic, their results can be ratioed and/or superimposed as required in order to obtain the load combinations characteristic of the particular loading condition.

The following individual loads are analyzed using the ANSYS model described in the previous section:

1. Bolt preload and seal seating pressure.

2. Internal Pressure loading.

3. External Pressure loading.

4. 1 g down with cask standing in a vertical position on the concrete storage pad.

5. Lifting (Cask Vertical)

6. Worst thermal condition.

7. 1 g lateral and 1 g down bounding loads on the cask standing in a vertical position on the concrete pad.

Loadings for Cases 1 through 6 are axisymmetric. In Case 7 Fourier series representation of the nonaxisymmetric loads are required. Each discrete load acting on the cask body is expanded into a Fourier series and is input into ANSYS as a series of load steps. Each load step contains all of the terms from the applied loads having the same mode number. The number of terms in the Fourier series required to adequately represent a load varies with the type of load (concentrated or distributed) and the degree of accuracy required. In this case, the load applied by the internals to the inside wall of the containment is assumed to be a distributed load varying sinusoidally in the arc 90° to 270° and acting on the total length of the cavity. Figure 4A.3-5 shows that only a few terms of the series are required to get a satisfactory representation of the load.

Since Case 7 is asymmetric, the resulting stresses are also asymmetric. Therefore in order to properly characterize the stress condition in the cask body, results are obtained at the two worst diametrically opposite locations and reported for the location where they are maximum.

The individual loads are described in the following paragraphs:

1. Bolt Preload and Seal Seating Pressure

   A lid bolt preload corresponding to 25,000 psi direct stress in the bolt shank is simulated by specifying an initial strain in the elements representing the bolts. A portion of this strain becomes elastic preload strain in the bolts, and a portion becomes strain in the clamped parts. The required initial strain value of 0.00134 in/in (in the bolts) was determined by trial and error.

   The selected bolt preload is sufficient to insure a full seating of the metallic seals under a maximum design internal pressure of 100 psig. The metallic seal seating load is 2198 lb./in./seal (Reference 5) or 4396 lb./in. for 2 seals. This load is simulated by applying a pressure of 1946.48 psi on an annular ring on both the containment lid and flange surfaces as shown in Figure 4A.3-6.

2. Internal Pressure Loading

   A conservative design pressure of 100 psig is used as the maximum pressure acting in the containment vessel cavity as shown in Figure 4A.3-7

3. External Pressure Loading

   A pressure of 25 psig is used as the maximum external pressure acting on the outer surface of the cask body as shown in Figure 4A.3-8.

4. I g Down

   The cask is stored vertically on the concrete storage pad as shown in Figure 4A.3-9, with the following loads acting on it:

   a. A distributed vertical down inertia force of 1 g acting at each finite element in the model. For practical purposes, the resultant of all these forces is shown acting at the C.G. of the cask. Note that the resin, the outer shell and the trunnions are not included in the model. They are accounted for by increasing the density of the gamma shielding.

   b. Since the internals are not included in the model, their loading effects are simulated by a distributed pressure acting on the inside bottom surface of the cask cavity.

   c. A vertical up reaction from the concrete pad is simulated as a uniformly distributed pressure acting on the outside bottom surface of the cask body. All of these forces acting on the cask form a system of forces in equilibrium

5. Lifting Cask Vertical

The cask is oriented vertically in space held by the 2 top trunnions and subjected to a vertical down load of 3 g, as shown in Figure 4A.3-10.

The intertia force acting on the cask elements and the pressure from the internals on the containment bottom inner surface are as described in Case 4 multiplied by a factor of 3. The pressure p at the bottom outer surface is eliminated and replaced by forces applied to the top 2 trunnions so that the system of forces acting on the cask is again in equilibrium. The cask weight of 250,000 lb. is used in calculations. The two trunnion forces F_{TR} = 1.5W are replaced by a uniform line force

$$q_y = \frac{3W}{2\pi R} - \frac{3\times 250{,}000}{2\times 3.14\times 45.5} = 2{,}623.43\ lb./in.$$

acting in the Y direction on the outer surface of the gamma shielding at the trunnion location. Superimposed on this solution are the local trunnion effects at two locations around the circumference which are determined by using the Bijlaard method. 1 g Down

6. Worst Temperature Distribution in the Cask Body (Off-Normal Condition)

A thermal analysis of the cask body using a 3D ANSYS thermal model is described in Chapter 3. The thermal model is used to obtain the steady state metal temperatures in the cask body for the off-normal condition which includes 100°F ambient air temperature, maximum decay heat and maximum solar heat loading. These temperatures are then used as ANSYS input for the thermal stress analysis.

7. 1 g Lateral and 1 g down Bounding Loads - Cask Standing in a Vertical Orientation on the Pad

The sinθ and cosθ terms of the Fourier series are used to represent the 1 g lateral load acting at the CG of each finite element of the model. The load applied by the internals to the inside surface of the containment is assumed to vary sinusoidally on a 180° arc as shown in Figure 4A.3-5, and the same Fourier representation applies. The 1 g down load is applied simultaneously (as described in 4, above) with the 1 g lateral load. The cask is held at the bottom and no tilting or sliding is allowed (See Figure 4A.3-11). This load combination is an upper bound loading for tornado wind, flood water, seismic loads, etc. (See Table 3.2-4).

Stress results for these individual loads are reported in Tables 4A.3.3-1 through 4A.3.3-07. Figure 4A.3-12 shows the locations on the cask body, where stress results are reported. These locations are divided into two groups, containment and non containment. Stress components and stress intensities at nodal locations on the inner and outer surfaces of each cask body component are reported in these tables.

These results are provided in this report to indicate the relative significance of the individual loads. These point-wise results are combined below in Section 4A.3.5 with each other and with the results of several hand computations to provide results for the various load combinations which are compared to the design criteria in Section 4.

4A.3.4 ADDITIONAL CASK BODY ANALYSES

Two additional analyses of the cask body were performed using classical methods rather than the ANSYS finite element method. These analyses determined the maximum stresses at local points on the body: (a) due to the trunnion reactions (while lifting the cask) and (b) in the locations where tornado missile impact might occur. The stress intensities from these loadings are combined with those from the other FEA loadings in Section 4A.3.5, below.

4A.3.4.1 TRUNNION LOCAL STRESSES

This section discusses the analysis performed to calculate the local stresses in the cask body outer gamma shielding at the trunnion locations due to the loadings applied through the trunnions. These local effects are not included in the ANSYS stress result tables reported above in Section 4A.3.3. The local stresses must be superimposed on the above stress results for the cases where the inertial lifting loads are reacted at the trunnions. The local stresses are calculated in accordance with the methodology of WRC Bulletin 107 (Reference 6) which is based on the Bijlaard analysis for local stresses in cylindrical shells due to external loadings.

Loading

The Bijlaard analysis was performed to support various structural evaluation cases. A summary of the trunnion loads is provided in Table 4A.3.4.1-1.

Section 4A.6 provides the analyses of the trunnions themselves under the limiting 6/10g lifting (cask vertical) loading. Those analyses were performed to demonstrate that the trunnions satisfy the ANSI N14.6 (Reference 11) design requirements for special lifting devices. The Bijlaard analysis described in this section was performed to verify that the trunnion induced stresses in the cask body are also acceptable.

Method of Analysis

The local stresses induced in the cask body by the trunnions are calculated using Bijlaard's method. The neutron shield and thin outer shell are not considered to strengthen either the trunnions or the gamma shielding cylinder. The trunnion is approximated by an equivalent attachment so that the curves of Reference 6 can be used to obtain the necessary coefficients. These resulting coefficients are inserted into blanks in the column entitled "Read Curves For " in a standard computation form, a sample of which is attached as Table 4A.3.4.1-2.

The stresses are calculated by performing the indicated multiplication in the column entitled "Compute Absolute Values of Stress and Enter Result." The resulting stress is inserted into the stress table at the eight stress locations, i.e., AU, AL, BU, BL, etc. Note that the sign convention for this table is defined on the figure for the load directions as shown. The membrane plus bending stresses are calculated by completing Table 4A.3.4.1-2.

Model, Boundary Conditions and Assumptions

The cylindrical body is assumed to be a hollow cylinder of infinite length. This is conservative since end restraints reduce the local cylinder bending effects.

Input Data

The only required input data for this analysis, are the dimensions of the trunnion and the cylinder. These are obtained from Section 1.3 drawings. The dimensions and Bijlaard parameters are listed as follows:

LIST OF BIJLAARD PARAMETERS

Parameter	Parameter Description	Parameter Value
R_m	Mean radius of shell	*41.1475 in.*
T	Wall thickness of shell	*7.295 in.*
$\gamma = \frac{R_m}{T}$	Shell Parameter	*5.64*
$r_\circ$	Outside radius of attachment	*6.0 in.*
$B = \frac{.875 r_\circ}{R_m}$	Attachment parameter	*0.1275 in.*

Results

Table 4A.3.4.1-3 summarizes the resulting local membrane stresses and bending (surface) stresses for the above loading conditions. These local stresses are combined with the finite element results at the same locations from Section 4A.3.3 above and compared with allowables in Section 4A.3.5 below.

4A.3.4.2 TORNADO MISSILE IMPACT

It is assumed that Missile A or Missile B, studied in Section 3.2.1.2.2 for cask stability, may impact against the cask during a tornado. That section concludes that Missile A, the automobile, could produce an impact force of 222,682 lb. over a 3 ft. x 6 ft. area of the cask producing an average contact pressure of 85.9 psi. Missile B, the wood plank, could deliver 442,080 lb. over 48 in.^{2} producing a 9210 psi local contact pressure. Section 3.2.1.2.2 shows that Missile A has a greater effect on cask stability than does Missile B because of its greater momentum (mv_o is greater for the 4000 lb. automobile at 50 mph than for the 200 lb. plank at 300 mph). However, in this section a quasi-static analysis is performed for the conditions of peak impact where Missile B delivers greater force and applies it to a smaller area than does Missile A.

Missile B is assumed to have the highest crush strength of any wood in Table 2, page 6-147 of Marks Handbook (Reference 7). That table lists a crush strength of 9210 psi for hickory. Therefore the impact force could reach the crush strength x end area for end impact of the plank (9210 psi x 4 in. x 12 in. = 442080 lb.). The minimum yield stress of the SA-105 gamma shielding forging material is 36,000 psi at room temperature and 31,900 psi at 300°F. Therefore the wood plank will crush before it penetrates the forged steel cask. Local damage of the neutron shielding might occur since the outer shell is relatively thin, but neither the massive gamma shielding forging nor the containment vessel (inside of the gamma shielding) will be punctured.

If Missile B were to strike the gamma shielding, the shear stress around the plug of material loaded by the blank would be:

$$\tau_{shielding} = \frac{Force}{shield\ thickness \times plank\ perimeter} = \frac{442{,}080 lb.}{8 in. \times 32 in.}$$
$$= 1{,}727\ psi$$

If it is assumed that the Missile B impact on the side of the cask is reacted by a 36 in. high cylindrical section of the gamma shielding as shown in Figure 4A.3-13, the bending stresses in the cylinder can be determined. If we consider the formula for Case 18 from Table VIII of Roark (Reference 8), the circumferential bending moment in the shielding ring is:

$$M_{\max} = \frac{3}{2} WR^2 \quad \textit{(maximum at section where F is applied)}$$

where $2\pi RW$ = Impact Force, F

R = mean radius of gamma shield, 41.5 in.

F = Impact Force, 442,080 lb.

then, $W = \frac{F}{2\pi R} = \frac{442{,}080}{2\pi \times 41.5} = 1{,}695\ lb./in.$

$$M_{max} = \frac{3}{2} \times 1{,}695 \times 41.5^2 = 4.38 \times 10^6\ in.lb.$$

The bending stress in the ring is:

$$P_{b\,shielding} = \frac{M_{max}\,C}{I\ of\ 36\ in.\ ring} = \frac{4.38 \times 10^6 \times 8/2}{1/12 \times 36 \times 8^3} = 11{,}405\,psi$$

The membrane stress is quite small.

If the plug shear stress of 1,727 psi is combined with this bending stress, the resulting stress intensity is:

$$SI_{shielding} = (11405^2 + 4 \times 1727^2)^{1/2} = 11{,}916\ psi$$

This stress intensity is far below the Level D allowable for SA-105 shielding material. This stress intensity will be combined with those for other loads and evaluated in Section 4A.3.5 below. The shield cylinder surrounds and protects the containment vessel so that containment stresses are negligible if the missile strikes the side of the cask.

If missile B were to strike the top of the cask, it could puncture the weather cover and neutron shield. The shear stress in the lid would be:

$$\tau_{lid} = \frac{Force}{lid\ thickness \times plank\ perimeter} = \frac{442{,}080lb.}{4.5in. \times 32in.} = 3{,}070\ psi$$

If we idealize the lid loaded by the impact force as indicated in Figure 4A.3-14, we can determine the lid bending stresses. Roark, in Case 2 from Table X, indicates that the bending stress in the center of the lid in both radial and tangential directions is:

$$P_{b_{lid}} = \frac{3F}{2\pi m t^2}\left[m + (m+1)\ln\frac{a}{r_\circ} - (m-1)\frac{r_\circ^2}{4a^2}\right]$$

r_o is the radius of the area where the load is applied

$\pi r_o^2 = 4 \times 12 = 48\ in.^2$ *(end area of plank)*

$r_o = (48/\pi)^{1/2} = 3.91\ in.$

m = 1/Poisson's ratio = 3.33

t = lid thickness = 4.5 in.

a = lid radius @ flange inside = 36 in.

$$P_{b_{lid}} = \frac{3\times 442{,}080}{2\pi\times 3.33\times 4.5^2}\left[3.33+4.33\,ln\frac{36}{3.91}-(2.33)\frac{3.91^2}{4\times 36^2}\right]=40{,}483\,psi$$

Combining the bending stress with the plug shear stress:

$SI_{lid} = (40483^2 + 4 \times 3070^2)^{1/2} = 40{,}945$ *psi*

This stress intensity is below the Level D allowable for the SA 350 LF3 lid material. This value will be combined with other cases and evaluated in Section 4A.3.5 below.

4A.3.5 EVALUATION (LOAD COMBINATIONS VS. ALLOWABLES)

The TN-40 cask loading conditions are listed in Section 3.2, Table 3.2-5. The individual loads acting on the various cask components due to these loading conditions have been applied to the cask and the resulting stresses are reported above in Tables 4A.3.3-1 through Table 4A.3.3-7.

The loading conditions listed in Table 3.2-5 are categorized according to the rules of the ASME Code, Section III, Subsection NB for Class 1 nuclear components. These categories include Design (same as Primary Service), Level A and Level D loading conditions. See Tables 3.2-6 through 3.2-8 for these categories. Next, the load combinations are determined based on those loads that can occur simultaneously. The individual loads making up each combination are indicated in Table 3.2-9.

The stress intensities for the combined load cases are evaluated at the locations indicated in Figure 4A.3-12 and compared to the stress limits associated with each service loading. To simplify the analysis only the containment shell stress limits were used in Tables 4A.3.5-1 through 4A.3.5-8. SA-203 Grade D containment shell stress limits are lower than the lid on the non containment stress limits (See Section 4.2.3.4, Tables 4.2-8 and -9 for containment and Table 4.2-10 for non containment).

The following conservative approach is used to arrive at the load combination stress intensities. At each location, instead of algebraically adding the corresponding stress components for the various load cases and determining the resulting stress intensity, the stress intensities for the various individual load cases are simply added together. The net stress intensity thus obtained is an upper bound value since it represents the absolute sum of the stresses rather than the algebraic sum (stress intensities have no signs). Also the membrane and bending stresses are not separated so the combined stress intensity is compared to the lower membrane allowable. In nearly all of the locations selected the stress intensities thus calculated are less than the membrane allowable. At those two locations where this simple conservative approach does not show margin, the membrane and bending stresses are separated.

The stress intensities at the selected locations for the Design, Level A and Level D combinations listed in Table 3.2-9 are reported in Tables 4A.3.5-1 through 4A.3.5-8. The containment stress limits for each case are also listed at the bottom of each table. See Chapter 8 for additional analyses of the cask body under hypothetical accident conditions. An additional condition not listed in Tables 3.2-5 or 3.2-9 that was evaluated is cold rain on a hot cask. Only the top of the cask body between the weather cover and neutron shield is directly exposed to rain. The major portion of the cylindrical length is protected by the neutron shield and outer shell. The analysis was based on the conservative assumption that the temperature of the outer skin of the cask body not covered by the outer shell is suddenly chilled to 32°F. The rest of the cask body is at the Off-Normal temperature distribution previously used.

This temperature distribution produces a peak stress of 52,474 psi in the containment flange at location 19. This stress is added to the maximum design stress of 4,389 psi, resulting in a total stress of 56,863. The stress is assumed to vary from 0 to the maximum value shown above; hence the maximum alternating stress is $S_a = 1/2 \times 56{,}863 = 28{,}432$ *psi*. The allowable number of cycles, NA, for this value of S_a is obtained from Figure I-9.1 of the ASME Code Section III Appendices and is 22,000 cycles. Note that the assumptions used to arrive at this value are very conservative.

4A.4 LID BOLT ANALYSES

4A.4.1 BOLT PRELOAD

The lid is secured to the cask body by forty eight 1.5 in. diameter UN-8 bolts. The selected bolt preload is such that the metallic containment seals are properly compressed and the lid seated against the flange with sufficient force to resist the maximum cavity internal pressure and any dead weight loads acting to unseat the lid. The corresponding tensile preload stress in the bolts at temperature is 25,000 psi (for dry bolt) which is less than the stress allowable for the bolt material for Design Conditions. The load per bolt is:

$$F_B = A_B \times 25{,}000$$

$$= 1.492 \times 25{,}000 = 37{,}300 \text{ lb./bolt}$$

The lubricated bolt preload stress of 51,000 psi need not be included in the "Design" condition. Since we have 48 bolts, the total seating force of all 48 bolts is:

$$48\ F_B = 1{,}790{,}400 \text{ lb.}$$

The force required to seat the seals is a line load of 2,198 pounds per inch of seal circumference. The diameter of the outer seal is 75.9 in. and the diameter of the inner seal 74.3 in. The seal seating force is then:

$$F_{seating} = 2{,}198\ \pi(75.9 + 74.3) = 1{,}037{,}164 \text{ lb.}$$

The maximum cask cavity internal pressure is the Design Pressure of 100 psi. The force required to react the pressure load, conservatively assuming the pressure is applied over the outer seal diameter, is:

$$F_{pressure} = 100 \frac{\pi}{4} (75.9)^2 = 452{,}453 \text{ lb.}$$

The TN-40 cask is always oriented vertically during loading, during transfer to the ISFSI and during storage on the pad. Dead weight of the lid and cask contents does not actually load the lid bolts. In fact the lid weight (and external pressure) help seat the lid. However, it is conservative to require that the bolt preload maintain lid seating in any cask orientation.

The weights of the lid, fuel and basket are:

Lid Weight	= 15,599 lb.
Fuel Weight	= 52,000 lb.
Basket Weight	= 15,841 lb.
W_{Total}	83,440 lb.

The total of the seal seating force, pressure load and dead weight loads is:

$F_{seating}$	= 1,037,164
$F_{pressure}$	= 452,453
W_{tota}	= 83,440
	1,573,057 lb.

Therefore the selected bolt preload stress of 25,000 psi provides ample lid seating force. The average bolt tensile stress required to react the lid loadings under Design Conditions is the preload stress of 25,000 psi which is well below the limiting value of S_m (31,900 psi) for the bolt material at 300°F.

4A.4.2 DIFFERENTIAL THERMAL EXPANSION

The 48 lid bolts preload the outer rim of the closure lid against the cask body flange. The 1.5 in. diameter bolts are installed through 1.56 in. diameter clearance holes in the 4.50 in. thick lid periphery. Preloading of the bolts against the lid is accomplished by tightening the bolts so that the shank portions of the bolts within the clearance holes are stretched elastically. The bolt loads will therefore change from the initial installed values if any thermal expansion differences should occur between the lid (through thickness direction) and the bolts.

The bolt material is SA 320 Grade L43 (1 3/4 Ni 3/4 Cr 1/4 Mo). The lid and body flange are both SA 350 Grade LF3 (3 1/2 Ni). The Section III Code Appendices specify the same coefficient of thermal expansion for these materials. The bolts are in intimate contact with the lid and flange and will therefore operate at the same temperature as these components. Therefore there will be no thermal expansion differences between the lid and bolts, and the assembly preload will be maintained under all temperatures.

4A.4.3 BOLT TORSION

The torque required to preload the dry bolt is:

$$T = 0.2\ D_N\ F_B$$

Where

A_B = *Bolt stress area = 1.492 in.*^{2}

D_N = *Bolt nominal dia = 1.5 in.*

F_B = *25,000 psi preload stress* $\times A_B$

The residual torque in the bolt is:

$$T_R = 0.5625T = 0.5625 \times 0.2 \times 1.5 \times 1.492 \times 25{,}000$$

$$= 6{,}294 \text{ in.lb.}$$

The shear stress in the bolt due to the residual torque from preload given by Reference 9:

$$\tau_{torsion} = \frac{T_R \times r}{J}$$

Where *r* and *J* are based on the bolt effective radius for the above stress area.

r = 0.689 in. effective bolt radius

$$J = \frac{\pi r^4}{2} = 0.354^4 \text{ in torsional moment of inertia of threaded bolt}$$

$$\tau_{torsion} = \frac{6_R 294 \times 0.689}{0.354} = 12{,}250 \text{ psi torsional shear}$$

4A.4.4 BOLT BENDING

It is assumed that bolt bending does not occur during seating of the lid against the cask body during assembly. The bolts are rotated as they are torqued so any slight relative movement between lid and body flange during preloading will not result in a net offset between the bolt head and tapped flange holes. In addition, since the lid, flange and bolt materials have the same coefficient of thermal expansion and will operate at essentially the same temperature, differential expansion between components will not produce bolt bending.

As internal pressure is applied to the cask cavity, the lid will bulge slightly and its edge will rotate. In addition the body cylinder radius will increase slightly due to the internal pressure resulting in outward radial movement of the tapped bolt holes in the body flange. Since no net membrane stress is developed in the lid, the lid bolt holes (at the mid surface) will remain at the original location. Rotation of the edge of the lid will, however, produce radial movement of the outer surface of the lid at the bolt head location.

The hoop stress in the cask body cylinder is:

$$S_{hoop} = \frac{PR_i}{t}$$

where *P = 100 psi Design Pressure*

R_i = 36 in. inside radius

t = 9.5 in. thickness

$$S_{hoop} = \frac{100 \times 36}{9.5} = 378.9\,psi$$

The radial deflection at the bolt circle is:

$$\delta_{bolt\ circle} = R_{bc} \times \frac{S_{hoop}}{E}$$

R_{bc} = 39.65 in. bolt circle radius

$$\delta_{bolt\ circle} = \frac{39.65 \times 378.9}{28 \times 10^6} = 0.000537\ in\,(\,outward\ motion\,)$$

When pressure is applied to the lid, the edge rotation can be calculated assuming the lid is simply supported:

$$\theta = \frac{3W(m-1)R}{2\pi Emt^3}\ from\ Reference\,8, Table\ X, Case\ 1$$

where:

θ = edge rotation, radians

W = total applied load

m = 1/Poisson's ratio = 3.33

R = 37.95 in. outer seal radius

t = 4.5 in. lid thickness

$$\theta = \frac{3 \times 100 \times \pi \times 37.95^3 \times 2.33}{2\pi \times 28 \times 10^6 \times 3.33 \times 4.5^3} = 0.002248 \ \ radians$$

Figure 4A.4-1 shows the net movement of the threaded hole and the point on the lid under the bolt head. If it is assumed that the bolt head doesn't slide on the lid surface, the head will be forced from position a to a' as the lid deflects. Point a' under the bolt head moves outward 0.005058 in. while the threaded hole moves only 0.000537 in. outward. The bolt head will be bent laterally by 0.005058 - 0.000537 in. or 0.00452 in. from the threaded end.

The bending model of the bolt is shown in Figure 4A.4-2. The moment on the bolt is calculated assuming the bolt is subjected to affect bending with the head and threaded end prevented from rotating. The model in Figure 4A.4-2 is half of a fixed-fixed beam with length 2l and center load 2P deflected δ at the center. If any end rotation occurred the moment would be reduced for a given deflection. For a cantilevered bolt free to rotate at the head, the bending moment would be reduced by one half. Therefore the assumption of fixed ends is the most conservative and results in the highest stress.

The shear force, *P*, and bending moment, *M*, are:

$P = \frac{12EI\delta}{l^3}$ for a beam subjected to offset bending with ends prevented from rotating

$$M = \frac{6EI\delta}{l^2}$$

Where

P = lateral load to deflect the bolt distance δ, lb.

δ = lateral displacement

= 0.00452 in.

E = Young's modulus, 28 x 10^6 psi @300°F

l = bolt length in bending

= 4.625 in. (including tapped hole chamfer)

I = $\pi r^4/4$

= 0.177 in.^{4} (r based on stress area of 1.492 in.^{2})

Therefore

$$M = \frac{6 \times 28 \times 10^6 \times 0.177 \times 0.00452}{4.625^2} = 6{,}283 \text{ in.-lb.}$$

$$P = \frac{12 \times 28 \times 10^6 \times 0.177 \times 0.00452}{4.625^3} = 2{,}712 \text{ lb.}$$

The bending stress in the bolt is

$$\sigma_b = \frac{Mr}{I} = \frac{6{,}283 \times 0.689}{0.177} = 24{,}457 \ psi$$

The shear stress due to the lateral force is

τ_p = P/A = 2712/1.492 = 1821 psi

4A.4.5 COMBINED STRESSES

The total shear stress is then equal to the residual torsional shear stress plus that due to force P.

$$\tau_{total} = \tau_{torsion} + \tau_p = 12{,}250 + 1{,}821 = 14{,}071\ psi$$

The maximum tensile stress at two locations in the lubricated bolt is the preload stress plus the bending stress.

$$\sigma_{max} = 51{,}000 + 24{,}457 = 75{,}457\ psi$$

The combined stress intensity is:

$$\begin{aligned} SI &= \left(\sigma_{max}{}^2 + 4(\tau_{total})^2\right)^{\frac{1}{2}} \\ &= \left(75{,}457^2 + 4 \times 14{,}074^2\right)^{\frac{1}{2}} \\ &= 80{,}534\ psi \end{aligned}$$

For Level A conditions, the average bolt stress is limited to 2 S_m or 2 x 31,900 = 63,800 psi. The maximum bolt stress is limited to 3 S_m or 95,700 psi. We are well within these limits as well as the yield strength of the bolt material (also 95,700 psi). The lid bolt stresses are no different under Level D conditions than Level A conditions.

4A.5 BASKET ANALYSIS

This section has been deleted from Appendix 4A. See Appendix 4B for the complete Basket Analysis.

4A.6 TRUNNION ANALYSIS

This section provides the structural analysis of the TN-40 storage cask trunnions. The trunnions shown in Figure 4A.6-1 are SA-105 carbon steel forgings. They are attached to the cask body with groove welds. A flat surface is machined on the cask body outer surface at each trunnion location for this purpose.

The two top trunnions are used for lifting the cask and are designed to the requirements of ANSI N14.6 (Reference 11) for lifting devices for use with a single failure proof crane. They can support a loading equal to 6 times the weight of the cask without generating stresses in excess of the minimum yield strength of the material. They can also lift 10 times the weight of the cask without exceeding the ultimate tensile strength of the material.

The lower trunnions are used to rotate the cask from a horizontal orientation to the vertical orientation. The lower trunnions will not be used to lift a loaded cask at Prairie Island. The lower trunnions are conservatively designed to half the load carried by the top trunnions and therefore can satisfy ANSI N14.6 if used with the top trunnions to lift a horizontal cask.

Figure 4A.6-1 shows the basic dimensions of the top and bottom trunnions. The cask total weight used in this calculation is W = 250,000 pounds. Table 4A.6-1 shows the cross sectional areas and moments of inertia at cross sections A-A and B-B of both upper and lower trunnions. In addition the loads applied to these sections (for 6 W and 10 W loading) to evaluate the yield and ultimate limits are listed.

Table 4A.6-2 presents a summary of the stresses at the same locations to compare against the yield and ultimate trunnion strengths. Also listed at the bottom of the table are the allowable stresses (yield and ultimate strengths).

The reported data shows that all of the calculated stresses in both the upper and lower trunnions are acceptable, and that the minimum margin of safety is 0.061 for the yield condition and 0.396 for the ultimate condition. Both minimums occur in the upper trunnion.

4A.7 OUTER SHELL

This section presents the structural analysis of the outer shell of the TN-40 storage cask. The outer shell consists of a cylindrical shell section and closure plates (segmented-welded together) at each end which connect the cylinder to the cask body. 01136263
The normal loads acting on the outer shell are due to internal and external pressure and the normal handling operations. Membrane stresses due to the pressure difference and bending and shear stresses due to the handling loads are determined. These stresses are compared to the allowable stress limits in Section 4.2 to assure that the design criteria are met.

Description

The outer shell is constructed from low-alloy carbon steel and is welded to the outer surface of the cask body gamma shielding. The cylindrical shell section is constructed of 0.50 in. thick partial cylinders joined together with partial penetration welds and welded to the closure plates via partial penetration welds. The closure plates are 0.75 in. thick segments that may be joined together via partial penetration welds. Pertinent dimensions are shown in Fig. 4A.7-1 and Figure 1.3-2. 01136263

Materials Input Data

The outer shell cylindrical section and closure plates are SA 516-GR 55. The material properties are taken from the Reference 2, ASME Code, Section III, Appendices. The yield strength of the material is also obtained from the Appendices at a temperature of 300°F.

Applied Loads

It is assumed that a pressure of 25 psi may be applied to either the inside or outside of the outer shell. This bounding assumption envelops the actual expected pressures described in Section 3.2.5.

The handling loads acting on the outer shell are a result of lifting. The loads applied to the shell as a result of these operations consist of the values given in Section 3.2-5. The weight or inertia g load can include all of the weights of the outer shell, neutron resin shield, and aluminum containers. The most severe Design and Level A Condition load is the 3 g inertia load in the vertical lifting orientation. The shell is also analyzed for 3 g loading when the cask is oriented horizontally to ensure it is not damaged during delivery to Prairie Island.

Method of Analysis

The structural analysis of the neutron shield outer shell, closure rings, and attachment welds utilized a finite element model and the ANSYS code to determine the stresses. The ANSYS results were manually adjusted to conservatively account for the partial penetration welds of the closure plates. The ANSYS finite element model modeled the 0.5 inch outer shell, the 0.75 inch closure plates (modeled as solid plates) the welds attaching the outer shell to the closure plates, the weld attaching the top closure plate to the cask body and included two different weld configurations for attaching the bottom closure plate to the cask body. The weight of the resin and aluminum boxes used as inputs to the calculation agree with those listed in Table 3.2-1. Note that the model did not include the weld that joins the outer shell half cylinders together, see discussion below.

The outer shell is constructed of cylinders joined together via partial penetration welds. The stresses at these weld locations were previously analyzed for internal pressure and horizontal handing loads via conservative hand calculations. Since the welds run axially along the outer shell, there are no loads due to vertical handling of the cask. Since the previously analyzed stresses were calculated via a conservative hand calculation and since they were less than the limiting stresses, a finite element analysis of these welds would calculate stresses less than those reported herein.

As discussed above, the ANSYS finite element model modeled the top and bottom closure plates as solid 0.75 inch plates. To ease assembly during fabrication, these plates may be made of segments welded together with partial penetration welds. The stresses were determined by conservatively assuming that the entire thickness of the plate corresponded to the thickness of the weld material. Since stresses are inversely proportional to the square of the thickness of the plate, the limiting ANSYS results (25 psi + 3g handling in vertical position) were adjusted by the appropriate ratio at the junction between the top plate and the cask body (necessary since the plate was now assumed to be thinner than the original attachment weld), the remainder of the top plate, and the bottom plate.

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4A.8 TOP NEUTRON SHIELD BOLTS

The top neutron shield (or resin disc) is bolted to the outside of the TN-40 lid using four SA-193, Gr B-8 bolts as indicated in Table 1.3-2 and Figure 1.3-2. The overpressure tank is attached to the upper surface of the shield. The weight of the overpressure tank is about 110 lbs and the weight of the top neutron shield is 1,692 lbs (Table 3.2-1) for a total component weight of 1,802 lbs attached through the four bolts.

The top neutron shield is necessary for the TN-40 cask to meet dose rate limits for Design and Level A conditions. Shielding analyses (Table 7A-4) show that the dose rate at the top of the lid without the neutron shield is well below the acceptable accident dose limit. Therefore the analysis below is limited to Design primary loadings. No analysis is needed for accident conditions in Chapter 8.

The neutron shield bolts have 1.25-7 UNC threads with a minor diameter of 1.0725 in. The stress area is at least $(\pi/4)(1.0725^2)$ or 0.9034 in^2. Under Design conditions the assembled and loaded (with fuel) TN-40 never experiences a net upward acceleration or a side load exceeding the 1.0g bounding load listed in Table 3.2-4. The tornado missile load is a Level D load. Nevertheless, a 3.0g upward or lateral load (not simultaneous) is assumed to conservatively evaluate these shield attachment bolts. The bolt stress under the 3.0g loading is equal to 3.0g x attached weight divided by the total bolt area. The stress is 3 x 1,802 lbs/(4 x 0.9034 in^2) = 1,496 psi. This is a tensile stress in the upward load case and a shear stress in the side load case.

These neutron shield attachment bolts are non-containment bolts. Therefore, the non-containment structure stress limits of Table 4.2-6 apply. The yield stress of SA-193, Gr B8 is 22,500 psi at 300°F. The allowable membrane stress is then 0.67(22,500) or 15,075 psi, and the allowable shear stress is 0.5(22,500) = 11,250 psi. The 1,496 psi applied stress (tension or shear) calculated above is well below the allowable and is therefore acceptable.

4A.9 REFERENCES

1. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, 1989.

2. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Appendices, 1989.

3. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code Section II, 1989.

4. De Salvo, G. J. and Swanson, J. A., ANSYS Engineering Analysis System, Users Manual for ANSYS Revision 4.3, Swanson Analysis Systems, Inc., Houston, PA, June 1987.

5. Resilient Metal Seals and Gaskets, Helicoflex Catalog H.001.002, Helicoflex Co., Boonton, N.J., 1983 pp.5-7.

6. WRC Bulletin 107, March 1979 Rev: "Local Stresses in Spherical and Cylindrical Shells Due to External Loadings."

7. Baumeister and Marks Standard Handbook for Mechanical Engineers, Sixth Edition.

8. Roark, R.J.: "Formulas for Stress & Strain", 4th Edition, McGraw-Hill Book Co.

9. Hopper, A.G. and Thompson, G.V. "Stress in Preloaded Bolts," Product Engineering, 1964.

10. Deleted

11. ANSI N14.6, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials," New York.

12. Timoshenko and Woinowsky-Krieger, Theory of Plates and Shells, Second Edition.

13. Safety Evaluation 72-425, TN-40 Cask Construction Issues During Cask 3 Fabrication, October 31, 1995.

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TABLE 4A.2-1

MECHANICAL PROPERTIES OF BODY MATERIALS (NOTE 1)

MATERIAL SPECIFICATION (NOMINAL COMPOSITION)	APPLICATION	MINIMUM YIELD STRENGTH S_y psi	MINIMUM ULTIMATE STRENGTH S_u psi	DESIGN STRESS VALUE, PSI (NOTE 2)	DATA SOURCE (NOTE 3)
ASME SA-350, Grade LF3 (3 ½ Ni)	Flange Containment Lid	37,500	70,000	S=17,500	Table I-7.1 P. 140 (Note 4)
ASME SA-203, Grade A & D	Containment Vessel	37,500	65,000	S=16,200	Table I-7.1
ASME SA-203, Grade E	Containment Vessel	40,000	70,000	S=17,500	P.140
ASME SA-320, Grade L43 (1 ¾ Ni ¾ Cr- ¼ Mo)	Closure Lid Bolts	105,000	125,000	S_m=35,000	Table I-1.3, P. 42
ASME SA-105 (C-Si)	Gamma Shielding Trunnions	36,000	70,000	S_m=23,300	Table I-1.1 P. 9
ASME SA-516, Grade 55 (C-Si)	Weather Cover Outer Shell	30,000	55,000	S=13,700	Table I-7.1 P. 132
ASME SA-516, Grade 70 (C-Mn-Si) (Note 5)	Gamma Shielding	38,000	70,000	S_m=23,300	Table 2A, P. 298 (Note 6)

NOTES

1. Mechanical properties listed are for metal temperatures up to 100°F to provide a baseline comparison of all structural materials. Temperature dependent properties required for structural analysis are provided in Table 4A.2-2.
2. Values listed are the stress parameters which form the basis for structural analysis acceptance criteria.
   S refers to the ASME allowable stress for Class 2 or Class 3 components,
   S_m refers to the ASME design stress intensity for Class 1 components, and S_y refers to minimum yield strength.
3. Data are taken from tables in ASME Section III, Appendix I, 1989 unless otherwise noted.
4. Table I - 7.1 actually lists 40,000 psi for S_y. However, the ASME material specification data sheet (Reference 3) (ASME Section II, Part A, p. 409) lists 37,500. Since the stress allowable for this material is based on ultimate strength (1/4 S_u), this variation in quoted yield strength does not affect selection of an analysis acceptance criteria.
5. Optional plate material used as a substitute for cask shell, bottom and shield plate SA-105 forged material.
6. Data is taken from table in ASME Section II, Part D, Subpart 1, 1992.

TABLE 4A.2-2
SHEET 1 OF 3

TEMPERATURE DEPENDENT MATERIAL PROPERTIES
COEFFICIENTS OF THERMAL EXPANSION (NOTE 1)

MATERIAL	TEMPERATURE, °F										
	100	**150**	**200**	**250**	**300**	**350**	**400**	**450**	**500**	**550**	**600**
SA-350,SA-320,SA-203 (Note 2)	6.27	6.41	6.54	6.65	6.78	6.88	6.98	7.07	7.16	7.24	7.32
SA-105 & SA-516, Gr 55 (Note 2)	5.73	5.91	6.09	6.27	6.43	6.57	6.74	6.89	7.06	7.18	7.28
SA-516, Gr 70 (Note 3) (Note 4)	5.53	5.71	5.89	6.09	6.26	6.43	6.61	6.77	6.91	7.06	7.17

NOTES

1. Values listed are the mean coefficients of thermal expansion x 10^{-6} (in./in. °F) from 70° F to the indicated temperature.

2. Source of data is ASME Section III, Appendix I, Table I-5.0, p. 123-128.

3. Optional plate material used as a substitute for cask shell, bottom and shield plate SA-105 forged material.

4. Source of data is ASME Section II, Part D, Subpart 2, Table TE-1, p. 638, 1992.

TABLE 4A.2-2
SHEET 2 OF 3

TEMPERATURE DEPENDENT MATERIAL PROPERTIES
MODULI OF ELASTICITY, E (NOTE 1)

	TEMPERATURE, °F					
MATERIAL	**70**	**200**	**300**	**400**	**500**	**600**
SA-203, SA-320, SA-350 (Note 2)	27.8	27.1	26.7	26.1	25.7	25.2
SA-516 Grade 55, SA-105 (Note2)	29.5	28.8	28.3	27.7	27.3	26.7
SA-516 Grade 70 (Note 3) (Note 4)	29.3	28.6	28.1	27.5	27.1	26.5

NOTES

1. Values listed are the moduli of elasticity x 10^6 psi for the indicated temperature.
2. Source of data is ASME Section III, Appendix I, Table I- 6.0, p. 129-138.
3. Optional plate material used as a substitute for cask shell, bottom and shield plate SA-105 forged material.
4. Source of data is ASME Section II, Part D, Subpart 2, Table TM-1, p.664, 1992

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TABLE 4A.2-2
SHEET 3 OF 3

TEMPERATURE DEPENDENT MATERIAL PROPERTIES
DESIGN STRESS PARAMETER

MATERIAL	STRESS PARAMETER (NOTE 1)	TEMPERATURE, °F 100	200	300	400	500	600	DATA SOURCE (NOTE 2)
SA-350, Grade LF3, SA-203, Grade E	S	17.5	17.5	17.5	17.5	17.5	17.5	Table I-7.1 p. 140
SA-203, Grade A and D	S	16.2	16.2	16.2	16.2	16.2	16.2	Table I-7.1 p. 140
SA-320, Grade L43	S_y	105.0	99.0*	95.7*	91.8*	88.5*	84.3*	Table I-1.3 p. 42. *(Note 3)
SA-105	S_m S_y S_u	23.3 36.0 70.0	21.9 32.8 70.0	21.3 31.9 70.0	**Not Required**			Table I-1.1 p. 9 Table I-2.1 p. 52 Table I-3.1 p.85
SA-516, Grade 55	S_y S_m	30.0 13.7	27.3 13.7	26.6 13.7	25.7 13.7			Table I-2.1 p. 50 Table I-7.1 p, 132
SA-516, Grade 70 (Note 4)	S_m S_y S_u	23.3 38.0 70.0	23.1 34.6 70.0	22.5 33.7 70.0	**Not Required**			Table 2A p. 300 Table y-1 p. 516 Table U p. 481 (Note 5)

NOTES

1. Values listed are the stress parameters which form the basis for structural analysis acceptance criteria.
   S refers to the ASME allowable stress for Class 2 or Class 3 components,
   Sm refers to the ASME design stress intensity for Class ! components, and
   Sy refers to minimum yield strength.
2. Data are taken from ASME Section III, Appendix I as noted.
3. For bolting materials, Sy≥ 3 Sm. Optional plate material used as a substitute for cask shell, bottom and shield plate SA-105 forged material.
4. Optional plate material used as a substitute for cask shell, bottom and shield plate SA-105 forged material.
5. Data is taken from ASME Section II, Part D, Subpart 1, 1992 as noted.

TABLE 4A.2-3

**REFERENCE TEMPERATURES FOR
STRESS ANALYSIS ACCEPTANCE CRITERIA (TN-40 CASK)**

COMPONENT	MAX. CALCULATED TEMPERATURE °F	SELECTED DESIGN TEMPERATURE °F
Cask Body	303	300
Cask Lid	242	300
Lid Bolts	242	300
Trunnions	242	300

TABLE 4A.3.3-1
BOLT PRELOAD
AND SEAL REACTION (TN-40 CASK)

LOCATION			MERIDIONAL STRESS (psi) OUTER SIGB S	MERIDIONAL STRESS (psi) INNER SIGT S	HOOP STRESS (psi) OUTER SIGB TH	HOOP STRESS (psi) INNER SIGT TH	STRESS INTENSITY SI (psi)
CONTAINMENT VESSEL		**1**		-136		-137	137
		2	-52		-51		52
		3		-2148		-710	2148
		4	1428		363		1428
		5		-343		-102	343
		6	-342		-101		342
		7		-2444		-54	2444
		8	1857		1209		1857
			STRESS COMPONENTS (psi) Sx	**Sy**	**Sz**	**Sxy**	**STRESS INTENSITY SI (psi)**
CONTAINMENT FLANGE & LID		**9**	-17	-2359	-1483	-69	2347
		10	495	2214	1159	815	2369
		11	-1314	-36	-1317	61	1284
		12	2118	43	2061	128	2091
NON	**GAMMA SHIELDING**	**13**	-245	0	-245	0	246
		14	264	-1	263	0	265
		15	-4	55	16	0	59
		16	0	59	14	0	59
	TRUNNION	**17**	1	-20	26	-2	45
Table D	**WELDS**	**18**	150	-384	-146	-97	568
		19	-395	2432	807	-484	2987
		20	-1458	-5454	-2982	-2070	5754

TABLE 4A.3.3-2

INTERNAL PRESSURE (100 PSIG) (TN-40 CASK)

LOCATION			MERIDIONAL STRESS (psi)		HOOP STRESS (psi)		STRESS INTENSITY SI (psi)
			OUTER SIGB S	INNER SIGT S	OUTER SIGB TH	INNER SIGT TH	
CONTAINMENT VESSEL		1		-533		-223	433
		2	297		-14		311
		3		4148		1058	4248
		4	-4385		-1295		4385
		5		636		584	736
		6	590		540		590
		7		305		151	405
		8	915		305		915
			STRESS COMPONENTS (psi)				STRESS INTENSITY SI (psi)
			Sx	Sy	Sz	Sxy	
CONTAINMENT FLANGE & LID		9	142	400	529	186	593
		10	-717	-756	-449	-176	464
		11	-859	-44	-882	23	839
		12	2057	38	2010	122	2034
NON Table E	GAMMA SHIELDING	13	-1164	51	-947	-25	1216
		14	1300	-112	1097	-29	1413
		15	-73	92	389	-1	462
		16	1	76	311	0	310
	TRUNNION	17	0	224	370	13	370
	WELDS	18	444	-1515	-489	-268	2031
		19	259	-767	-221	312	1200
		20	-2402	-399	-756	-789	2549

TABLE 4A.3.3-3

EXTERNAL PRESSURE (25PSIG) (TN-40 CASK)

<table>
<tr><th rowspan="2" colspan="2">LOCATION</th><th rowspan="2"></th><th colspan="2">MERIDIONAL STRESS (psi)</th><th colspan="2">HOOP STRESS (psi)</th><th rowspan="2">STRESS INTENSITY SI (psi)</th></tr>
<tr><th>OUTER SIGB S</th><th>INNER SIGT S</th><th>OUTER SIGB TH</th><th>INNER SIGT TH</th></tr>
<tr><td rowspan="8" colspan="2">CONTAINMENT VESSEL</td><td>1</td><td></td><td>110</td><td></td><td>113</td><td>113</td></tr>
<tr><td>2</td><td>-69</td><td></td><td>-71</td><td></td><td>71</td></tr>
<tr><td>3</td><td></td><td>-1838</td><td></td><td>-537</td><td>1838</td></tr>
<tr><td>4</td><td>1366</td><td></td><td>425</td><td></td><td>1366</td></tr>
<tr><td>5</td><td></td><td>-221</td><td></td><td>-174</td><td>221</td></tr>
<tr><td>6</td><td>-220</td><td></td><td>-174</td><td></td><td>220</td></tr>
<tr><td>7</td><td></td><td>-472</td><td></td><td>-144</td><td>472</td></tr>
<tr><td>8</td><td>51</td><td></td><td>7</td><td></td><td>51</td></tr>
<tr><td rowspan="2" colspan="3"></td><td colspan="4">STRESS COMPONENTS (psi)</td><td rowspan="2">STRESS INTENSITY SI (psi)</td></tr>
<tr><td>Sx</td><td>Sy</td><td>Sz</td><td>Sxy</td></tr>
<tr><td rowspan="4" colspan="2">CONTAINMENT FLANGE & LID</td><td>9</td><td>-49</td><td>-126</td><td>-265</td><td>-60</td><td>249</td></tr>
<tr><td>10</td><td>223</td><td>287</td><td>127</td><td>52</td><td>190</td></tr>
<tr><td>11</td><td>195</td><td>-15</td><td>202</td><td>-4</td><td>216</td></tr>
<tr><td>12</td><td>-558</td><td>-35</td><td>-544</td><td>-34</td><td>527</td></tr>
<tr><td rowspan="8">NON Table A</td><td rowspan="4">GAMMA SHIELDING</td><td>13</td><td>479</td><td>-2</td><td>483</td><td>-2</td><td>485</td></tr>
<tr><td>14</td><td>-562</td><td>-22</td><td>-566</td><td>-2</td><td>544</td></tr>
<tr><td>15</td><td>-7</td><td>-44</td><td>-120</td><td>0</td><td>113</td></tr>
<tr><td>16</td><td>-25</td><td>-40</td><td>-101</td><td>0</td><td>76</td></tr>
<tr><td>TRUNNION</td><td>17</td><td>-25</td><td>-71</td><td>-118</td><td>-3</td><td>93</td></tr>
<tr><td rowspan="3">WELDS</td><td>18</td><td>-163</td><td>432</td><td>125</td><td>82</td><td>617</td></tr>
<tr><td>19</td><td>-136</td><td>297</td><td>48</td><td>-135</td><td>511</td></tr>
<tr><td>20</td><td>596</td><td>58</td><td>173</td><td>194</td><td>663</td></tr>
</table>

TABLE 4A.3.3-4

1 g DOWN – TN-40 CASK STANDING IN A VERTICAL POSITION ON A CONCRETE PAD

LOCATION			MERIDIONAL STRESS (psi) OUTER SIGB S	MERIDIONAL STRESS (psi) INNER SIGT S	HOOP STRESS (psi) OUTER SIGB TH	HOOP STRESS (psi) INNER SIGT TH	STRESS INTENSITY SI (psi)
CONTAINMENT VESSEL		1		198		177	198
		2	-30		-5		30
		3		-872		-167	872
		4	1046		342		1046
		5		-76		-14	76
		6	-76		-14		76
		7		45		-3	48
		8	-163		-65		163
			STRESS COMPONENTS (psi) Sx	**Sy**	**Sz**	**Sxy**	**STRESS INTENSITY SI (psi)**
CONTAINMENT FLANGE & LID		9	-2	-3	21	-2	25
		10	-22	-62	-28	-18	54
		11	27	0	28	-1	28
		12	-58	-1	-56	-3	57
Table G NON	GAMMA SHIELDING	13	390	-21	394	-2	415
		14	-412	-36	-416	-2	380
		15	-1	-19	3	0	22
		16	0	-19	2	0	22
	TRUNNION	17	0	-4	2	0	21
	WELDS	18	-129	325	133	82	483
		19	13	-48	-14	15	68
		20	73	10	23	25	78

TABLE 4A.3.3-5

LIFTING – 3 G VERTICAL-UP (TN-40 CASK)

LOCATION			MERIDIONAL STRESS (psi)		HOOP STRESS (psi)		STRESS INTENSITY SI (psi)
			OUTER SIGB S	INNER SIGT S	OUTER SIGB TH	INNER SIGT TH	
CONTAINMENT VESSEL		1		-644		-597	585
		2	12		-36		48
		3		1155		88	1200
		4	-1788		-717		1788
		5		225		-4	228
		6	217		-6		223
		7		-1067		-86	1067
		8	1572		705		1572
			STRESS COMPONENTS (psi)				STRESS INTENSITY SI (psi)
			Sx	Sy	Sz	Sxy	
CONTAINMENT FLANGE & LID		9	-50	-168	-525	-62	502
		10	572	1084	561	275	752
		11	55	0	58	0	58
		12	-238	-5	-232	-15	235
NON	GAMMA SHIELDING	13	-1598	-55	-1606	4	1551
		14	1698	-9	1707	5	1716
		15	0	248	5	-2	248
		16	0	212	6	-1	218
	TRUNNION	17	9	664	212	81	675
Table B	WELDS	18	471	-1463	-535	-295	2022
		19	-280	980	302	-343	1435
		20	223	44	63	77	235

TABLE 4A.3.3-6

THERMAL STRESSES DUE TO OFF-NORMAL TEMPERATURE DISTRIBUTION (TN-40 CASK)

LOCATION		MERIDIONAL STRESS (psi)		HOOP STRESS (psi)		STRESS INTENSITY SI (psi)
		OUTER SIGB S	INNER SIGT S	OUTER SIGB TH	INNER SIGT TH	
CONTAINMENT VESSEL	1		-2468		-2442	2468
	2	-2814		-2847		2847
	3		-7202		-8267	8267
	4	4736		-4685		9422
	5		-1798		-1654	1798
	6	-566		-1285		1285
	7		1591		-1935	3526
	8	-4033		-3529		4033
		STRESS COMPONENTS (psi)				STRESS INTENSITY SI (psi)
		Sx	Sy	Sz	Sxy	
CONTAINMENT FLANGE & LID	9	175	555	797	191	702
	10	-2324	-3489	-1844	-895	2135
	11	40	49	59	-28	56
	12	1300	-207	808	120	1525
Table F NON CONTAINMENT — GAMMA SHIELDING	13	2526	16	2482	15	2510
	14	26	12	-11	-12	43
	15	-195	-741	33	-20	775
	16	-190	1009	469	64	1206
NON CONTAINMENT — TRUNNION	17	368	417	73	101	424
NON CONTAINMENT — WELDS	18	5918	-4071	2783	793	10113
	19	1418	-2067	-446	878	3903
	20	2031	-265	278	348	2398

TABLE 4A.3.3-7

1 g LATERAL AND 1 g DOWN
BOUNDING LOAD FOR SEISMIC, TORNADO WIND & FLOOD (TN-40 CASK)

LOCATION		MERIDIONAL STRESS (psi) OUTER SIGB S	MERIDIONAL STRESS (psi) INNER SIGT S	HOOP STRESS (psi) OUTER SIGB TH	HOOP STRESS (psi) INNER SIGT TH	STRESS INTENSITY SI (psi)
CONTAINMENT VESSEL	1*		38		58	97
	2*	147		84		147
	3		-4371		-1427	4363
	4	3053		800		3053
	5		-166		27	192
	6	-158		49		207
	7		-113		29	142
	8	-12		60		72
		STRESS COMPONENTS (psi)				**STRESS INTENSITY SI (psi)**
		Sx	**Sy**	**Sz**	**Sxy**	
CONTAINMENT FLANGE & LID	9	16*	4	-89	0	105
	10	30*	-19	-47	-17	82
	11	56*	1	49	-2	55
	12	-125*	-4	-127	-7	123
Table C NON CONTAINMENT – GAMMA SHIELDING	13	-14	-53	-23	7	42
	14	-14	7	-7	-1	22
	15	-8	-127	-15	-2	118
	16	0	-73	91	3	164
NON CONTAINMENT – TRUNNION	17	0	-18	70	-1	88
NON CONTAINMENT – WELDS	18	270	-1891	-286	-81	2167
	19	-23	-81	53	-1	134
	20	179	10	47	61	209

<u>NOTE</u>

Values with * sign are at $\theta=0^{o}$

TABLE 4A.3.4.1-1

***TRUNNION LOADINGS ON TN-40**
FOR USE IN CASK BODY EVALUATION

LOADING DESCRIPTION	INERTIAL LOAD	MAX. TRUNNION LOAD (LOAD SHARED BY 2 TOP TRUNNIONS)
Lifting (Cask Vertical)	3 G	V_L = 375 kips M_L = - 2093 in kips

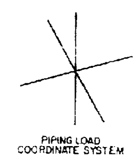
*Based on Figure 3.2-4 of Section 3.2
Note: 1G = weight of cask = 250 kips

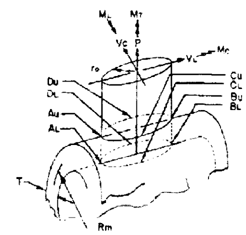
TABLE 4A.3.4.1-2

COMPUTATION SHEET FOR LOCAL STRESSES IN CYLINDRICAL SHELLS

* 1 APPLIED LOADS		2 GEOMETRY		3 GEOMETRIC PARAMETERS
RADIAL LOAD	P = ______ LB.	VESSEL THICKNESS	T = ______ IN.	$\gamma = \frac{Rm}{T} =$ ______
CIRC. MOMENT	Mc = ______ IN-LB.	ATTACHMENT RADIUS	r_o = ______ IN.	$\beta = (0.875)\frac{r_o}{Rm} =$ ______
LONG. MOMENT	ML = ______ IN-LB.	VESSEL RADIUS	Rm = ______ IN.	
TORSION MOMENT	MT = ______ IN-LB.			
SHEAR LOAD	Vc = ______ LB.			
SHEAR LOAD	VL = ______ LB.			

* NOTE: ENTER ALL FORCE VALUES IN ACCORDANCE WITH SIGN CONVENTION





FROM FIG.	READ CURVES FOR	COMPUTE ABSOLUTE VALUES OF STRESS AND ENTER RESULT	STRESSES – IF LOAD IS OPPOSITE THAT SHOWN, REVERSE SIGNS SHOWN Au	AL	Bu	BL	Cu	CL	Du	DL
3C AND 4C	$\frac{N\phi}{P/Rm} =$	$\left(\frac{N\phi}{P/Rm}\right)\cdot\frac{P}{RmT} =$	+	+	+	+	+	+	+	+
1C AND 2C-1	$\frac{M\phi}{P} =$	$\left(\frac{M\phi}{P}\right)\cdot\frac{6P}{T^2} =$	+	−	+	−	+	−	+	−
3A	$\frac{N\phi}{Mc/Rm^2\beta} =$	$\left(\frac{N\phi}{Mc/Rm^2\beta}\right)\cdot\frac{Mc}{Rm^2\beta T} =$					−	−	+	+
1A	$\frac{M\phi}{Mc/Rm\beta} =$	$\left(\frac{M\phi}{Mc/Rm\beta}\right)\cdot\frac{6Mc}{Rm\beta T^2} =$					−	+	+	−
3B	$\frac{N\phi}{ML/Rm^2\beta} =$	$\left(\frac{N\phi}{ML/Rm^2\beta}\right)\cdot\frac{ML}{Rm^2\beta T} =$	−	−	+	+				
1B OR 1B-1	$\frac{M\phi}{ML/Rm\beta} =$	$\left(\frac{M\phi}{ML/Rm\beta}\right)\cdot\frac{6ML}{Rm\beta T^2} =$	−	+	+	−				
ADD ALGEBRAICALLY FOR SUMMATION OF ϕ STRESSES $\sigma\phi$ =										
3C AND 4C	$\frac{Nx}{P/Rm} =$	$\left(\frac{Nx}{P/Rm}\right)\cdot\frac{P}{RmT} =$	+	+	+	+	+	+	+	+
1C-1 AND 2C	$\frac{Mx}{P} =$	$\left(\frac{Mx}{P}\right)\cdot\frac{6P}{T^2} =$	+	−	+	−	+	−	+	−
4A	$\frac{Nx}{Mc/Rm^2\beta} =$	$\left(\frac{Nx}{Mc/Rm^2\beta}\right)\cdot\frac{Mc}{Rm^2\beta T} =$					−	−	+	+
2A	$\frac{Mx}{Mc/Rm\beta} =$	$\left(\frac{Mx}{Mc/Rm\beta}\right)\cdot\frac{6Mc}{Rm\beta T^2} =$					−	+	+	−
4B	$\frac{Nx}{ML/Rm^2\beta} =$	$\left(\frac{Nx}{ML/Rm^2\beta}\right)\cdot\frac{ML}{Rm^2\beta T} =$	−	−	+	+				
2B OR 2B-1	$\frac{Mx}{ML/Rm\beta} =$	$\left(\frac{Mx}{ML/Rm\beta}\right)\cdot\frac{6ML}{Rm\beta T^2} =$	−	+	+	−				
ADD ALGEBRAICALLY FOR SUMMATION OF x STRESSES σx =										
SHEAR STRESS DUE TO TORSION MT		$\tau\phi x = \tau x\phi = \frac{M_T}{2\pi r_o^2 T}$	+	+	+	+	+	+	+	+
SHEAR STRESS DUE TO LOAD Vc		$\tau x\phi = \frac{V_c}{\pi r_o T}$	+	+	−	−				
SHEAR STRESS DUE TO LOAD VL		$\tau\phi x = \frac{V_L}{\pi r_o T}$					+	+	−	−
ADD ALGEBRAICALLY FOR SUMMATION OF SHEAR STRESSES τ =										

		LONGITUDINAL σx		CIRCUMFERENTIAL $\sigma\phi$
PRESSURE STRESS	$\frac{PR-0.4PT}{2ET} =$	= ______	$\frac{PR+0.6PT}{ET} =$	= ______
LONGITUDINAL BENDING STRESS		= ______		
TOTAL MEMBRANE STRESS		= ______ ≤ ______		= ______ ≤ ______
TOTAL SURFACE STRESS		= ______ ≤ ______		= ______ ≤ ______

NOZZLE NO. ______
PIPING LOAD CODE ______
ANALYSIS POINT ______

COMPUTATION SHEET FOR LOCAL STRESSES IN CYLINDRICAL SHELLS (   )

SERVICE:
ITEM NO.
S-R JOB NO.
DATE   BY   SHEET

TABLE 4A.3.4.1-3

STRESSES ON TN-40 CASK BODY DUE TO TRUNNION LOADING

DESCRIPTION	CIRCUMFERENTIAL STRESS (PSI) MEMBRANE	CIRCUMFERENTIAL STRESS (PSI) SURFACE	LONGITUDINAL STRESS (PSI) MEMBRANE	LONGITUDINAL STRESS (PSI) SURFACE
3 g – Lifting Cask Vertical	413	3,208	102	4,546

TABLE 4A.3.5-1

DESIGN (1) LOAD COMBINATION (TN-40 CASK)

LOCATION			BOLT PRELOAD(a) (PSI)	1 g DOWN (PSI)	INTERNAL PRESSURE (PSI)	BOUNDING VERTICAL & LATERAL FORCES (PSI)	ΣS.I.(a) PSI	ΣS.I. (b) PSI
CONTAINMENT VESSEL		1	137	198	433	97	865	1007
		2	52	30	311	147	540	594
		3	2148	872	4248	4363	11631	13865
		4	1428	1046	4385	3053	9912	11397
		5	343	76	736	192	1347	1704
		6	342	76	590	207	1215	1571
		7	2444	48	405	142	3039	5581
		8	1857	163	915	72	3007	4938
CONTMNT FLANGE & LID		9	2347	25	593	105	3070	5511
		10	2369	54	464	82	2969	5433
		11	1284	28	839	55	2206	3541
		12	2091	57	2034	123	4305	6480
NON CONTAINMENT	GAMMA SHIELDING	13	246	415	1216	42	1919	2175
		14	265	380	1413	22	2080	2356
		15	59	22	462	118	661	722
		16	59	21	310	164	554	615
	TRUNNION	17	45	6	370	88	509	556
	WELDS	18	568	483	2031	2167	5249	5840
		19	2987	68	1200	134	4389	7495
		20	5754	78	2549	209	8590	14574
CONTAINMENT ALLOWABLE STRESS						S_m =	16,200	

(a) CORRESPOND TO BOLT PRELOAD STRESS OF 25,000 psi

(b) TOTAL = Σ S.I.$_{(a)}$+BOLT PRELOAD $\times \left(\frac{51}{25}\right) - 1$

TABLE 4A.3.5-2
DESIGN (2) LOAD COMBINATION (TN-40 CASK)

LOCATION			BOLT PRELOAD(a)	INTERNAL PRESSURE	LIFTING 3g UP	BIJLAARD LOCAL MEMBRANE	ΣS.I.(a) PSI	ΣS.I. (b) PSI
			STRESS INTENSITY S.I. (PSI)					
CONTAINMENT VESSEL		1	137	433	585		1155	1297
CONTAINMENT VESSEL		2	52	311	48		411	465
CONTAINMENT VESSEL		3	2148	4248	1200		7596	9830
CONTAINMENT VESSEL		4	1428	4385	1788		7601	9086
CONTAINMENT VESSEL		5	343	736	228		1307	1664
CONTAINMENT VESSEL		6	342	590	223		1155	1511
CONTAINMENT VESSEL		7	2444	405	1067		3916	6458
CONTAINMENT VESSEL		8	1857	915	1572		4344	6275
CONTMNT FLANGE & LID		9	2347	593	502		3442	5883
CONTMNT FLANGE & LID		10	2369	464	752		3585	6049
CONTMNT FLANGE & LID		11	1284	839	58		2181	3516
CONTMNT FLANGE & LID		12	2091	2034	235		4360	6535
NON CONTAINMENT	GAMMA SHIELDING	13	246	1216	1551		3013	3269
NON CONTAINMENT	GAMMA SHIELDING	14	265	1413	1716		3394	3670
NON CONTAINMENT	GAMMA SHIELDING	15	59	462	248		769	830
NON CONTAINMENT	GAMMA SHIELDING	16	59	310	218		587	648
NON CONTAINMENT	TRUNNION	17	45	370	675	11700	12790	12837
NON CONTAINMENT	WELDS	18	568	2031	2022		4621	5212
NON CONTAINMENT	WELDS	19	2987	1200	1435		5622	8728
NON CONTAINMENT	WELDS	20	5754	2549	235		8538	14522
CONTAINMENT ALLOWABLE STRESS			DESIGN (2) LOAD COMBINATION ALLOWABLE STRESS S_m =				16,200	

(a) CORRESPOND TO BOLT PRELOAD STRESS OF 25,000 psi

(b) TOTAL = $\Sigma S.I._{(a)}$+BOLT PRELOAD $\times\left(\frac{51}{25}\right)-1$

TABLE 4A.3.5-3

DESIGN (3) LOAD COMBINATION (TN-40 CASK)

LOCATION			STRESS INTENSITY S.I. (PSI) BOLT PRELOAD(a)	1 g DOWN	EXTERNAL PRESSURE	BOUNDING VERTICAL & LATERAL LOADS	ΣS.I.(a) PSI	ΣS.I. (b) PSI
CONTAINMENT VESSEL		1	137	198	113	97	545	687
		2	52	30	71	147	300	354
		3	2148	872	1838	4363	9221	11455
		4	1428	1046	1366	3053	6893	8378
		5	343	76	221	192	832	1189
		6	342	76	220	207	845	1201
		7	2444	48	472	142	3106	5648
		8	1856	163	51	72	2143	4074
CONTMNT FLANGE & LID		9	2347	25	249	105	2726	5167
		10	2369	54	190	82	2695	5159
		11	1284	28	216	55	1583	2918
		12	2091	57	527	123	2798	4973
NON CONTAINMENT	GAMMA SHIELDING	13	246	415	485	42	1188	1444
		14	265	380	544	22	1211	1487
		15	59	22	113	118	312	373
		16	59	21	76	164	320	381
	TRUNNION	17	45	6	93	88	232	279
	WELDS	18	568	483	617	2167	3835	4426
		19	2987	68	511	134	3700	6806
		20	5754	78	663	209	6704	12688
CONTAINMENT ALLOWABLE STRESS						S_m =	16,200	

(a) CORRESPOND TO BOLT PRELOAD STRESS OF 25,000 psi

(b) TOTAL = Σ S.I.$_{(a)}$+BOLT PRELOAD $\times$ $(51/25 - 1)$

TABLE 4A.3.5-4

LEVEL A (1) LOAD COMBINATION (TN-40 CASK)

LOCATION			STRESS INTENSITY S.I. (PSI)					
					DESIGN (1) LOAD COMBINATIONS(a)	WORST THERMO	ΣS.I.(a) PSI	ΣS.I. (b) PSI
CONTAINMENT VESSEL		1			865	2468	3333	3475
		2			540	2847	3387	3441
		3			11631	8267	19898	22132
		4			9912	9422	19334	20819
		5			1347	1798	3145	3502
		6			1215	1285	2500	2856
		7			3039	3526	6565	9107
		8			3007	4033	7040	8971
CONTMNT FLANGE & LID		9			3070	702	3772	6213
		10			2969	2135	5104	7568
		11			2206	56	2262	3597
		12			4305	1525	5830	8005
NON CONTAINMENT	GAMMA SHIELDING	13			1919	2510	4429	4685
		14			2080	43	2123	2399
		15			661	775	1436	1497
		16			554	1206	1760	1821
	TRUNNION	17			509	424	933	980
	WELDS	18			5249	10113	15362	15953
		19			4389	3903	8292	11398
		20			8590	2398	10988	16972
CONTAINMENT ALLOWABLE STRESS						3.0 S_m =	48600	

(a) CORRESPOND TO BOLT PRELOAD STRESS OF 25,000 psi

(b) TOTAL = WORST THERMO + $\Sigma S.I._{(b)}$ (TABLE 4A.3.5-1)

TABLE 4A.3.5-5

LEVEL A (2) LOAD COMBINATION (TN-40 CASK)

LOCATION			STRESS INTENSITY S.I. (PSI)				ΣS.I.(a) PSI	ΣS.I. (b) PSI
			BOLT PRELOAD(a)	INTERNAL PRESSURE	LIFTING UP	WORST THERMO		
CONTAINMENT VESSEL		1	137	433	585	2468	3623	3765
		2	52	311	48	2847	3258	3312
		3	2148	4248	1200	8267	15863	18097
		4	1428	4385	1788	9422	17023	18508
		5	343	736	228	1798	3105	3462
		6	342	590	223	1285	2440	2796
		7	2444	405	1067	3526	7442	9984
		8	1857	915	1572	4033	8377	10308
CONTMNT FLANGE & LID		9	2347	593	502	702	4144	6585
		10	2369	464	752	2135	5720	8184
		11	1284	839	58	56	2237	3572
		12	2091	2034	235	1525	5885	8060
NON CONTAINMENT	GAMMA SHIELDING	13	246	1216	1551	2510	5523	5779
		14	265	1413	1716	43	3437	3713
		15	59	462	248	775	1544	1605
		16	59	310	218	1206	1793	1854
	TRUNNION	17	45	370	12375*	424	13214	13261
	WELDS	18	568	2031	2022	10113	14734	15325
		19	2987	1200	1435	3903	9525	12631
		20	5754	2549	235	2398	10936	16920
CONTAINMENT ALLOWABLE STRESS			LEVEL A SERVICE LOAD COMBINATION			$3.0S_m$ =	48600	

* INCLUDES BIJLAARD LOCAL MEMBRANE

(a) CORRESPOND TO BOLT PRELOAD STRESS OF 25,000 psi

(b) TOTAL = Σ S.I.$_{(a)}$+BOLT PRELOAD $\times$ $(51/25 - 1)$

TABLE 4A.3.5-6

LEVEL A (3) LOAD COMBINATION (TN-40 CASK)

LOCATION				STRESS INTENSITY S.I. (PSI)				ΣS.I.(a) PSI	ΣS.I. (b) PSI
						DESIGN (3) LOAD COMBINATIONS(a)	WORST THERMO		
CONTAINMENT VESSEL			1			545	2468	3013	3155
			2			300	2847	3147	3201
			3			9221	8267	17488	19722
			4			6893	9422	16315	17800
			5			832	1798	2630	2987
			6			845	1285	2130	2486
			7			3106	3526	6632	9174
			8			2143	4033	6176	8107
CONTMNT FLANGE & LID			9			2726	702	3428	5869
			10			2695	2135	4830	7294
			11			1583	56	1639	2974
			12			2798	1525	4323	6498
NON CONTAINMENT	GAMMA SHIELDING		13			1188	2510	3698	3954
			14			1211	43	1254	1530
			15			312	775	1087	1148
			16			320	1206	1526	1587
	TRUNNION		17			232	424	656	703
	WELDS		18			3835	10113	13948	14539
			19			3700	3903	7603	10709
			20			6704	2398	9102	15086
CONTAINMENT ALLOWABLE STRESS							3.0 S_m =	48600	

(a) CORRESPOND TO BOLT PRELOAD STRESS OF 25,000 psi

(b) TOTAL = WORST THERMO + ΣS.I.$_{(b)}$ (TABLE 4A.3.5-3)

TABLE 4A.3.5-7

LEVEL D (1) LOAD COMBINATION (TN-40 CASK)

LOCATION			STRESS INTENSITY S.I. (PSI)				ΣS.I.(a) PSI	ΣS.I. (b) PSI
					DESIGN (1) LOAD COMBINATION(a)	TORNADO MISSILE		
CONTAINMENT VESSEL		1						
		2						
		3						
		4						
		5						
		6						
		7						
		8						
CONTMNT FLANGE & LID		9						
		10						
		11			2206	40945*	43151	44486
		12			4305	40945*	45250	47425
NON CONTAINMENT	GAMMA SHIELDING	13						
		14						
		15			661	11916	12577	12638
		16			554	11916	12470	12531
	TRUNNION	17						
	WELDS	18						
		19			4389	11916	16305	19411
		20			8590	40945*	49535	55519
CONTAINMENT ALLOWABLE STRESS			MEMBRANE			$2.4S_m$=	38880	
			MEMBRANE+BENDING			$3.6S_m$=	58320	

* THE MEMBRANE PORTION IS 6,140 psi; THE BALANCE IS BENDING

(a) CORRESPOND TO 25,000 psi BOLT PRELOAD STRESS

(b) TOTAL = TORNADO MISSILE +ΣS.I. (b) (TABLE 4A.3.5-1)

TABLE 4A.3.5-8

LEVEL D (2) LOAD COMBINATION (TN-40 CASK)

LOCATION			STRESS INTENSITY S.I. (PSI)				ΣS.I.(a) PSI	ΣS.I.(b) PSI
					DESIGN (2) LOAD COMBINATION(a)	TORNADO MISSILE		
CONTAINMENT VESSEL		1						
		2						
		3						
		4						
		5						
		6						
		7						
		8						
CONTMNT FLANGE & LID		9						
		10						
		11			1583	40945*	42528	43863
		12			2798	40945*	43743	45918
NON CONTAINMENT	GAMMA SHIELDING	13						
		14						
		15			312	11916	12228	12289
		16			320	11916	12236	12297
	TRUNNION	17						
	WELDS	18						
		19			3700	11916	15616	18722
		20			6704	40945*	47649	53633
CONTAINMENT ALLOWABLE STRESS			MEMBRANE			2.4 S_m=	38880	
			MEMBRANE + BENDING			3.6 S_m=	58320	

* THE MEMBRANE PORTION IS 6,140 psi; THE BALANCE IS BENDING

(a) CORRESPOND TO 25,000 psi BOLT PRELOAD STRESS

(b) TOTAL = TORNADO MISSILE +ΣS.I. $_{(b)}$ (TABLE 4A.3.5-3)

TABLE 4A.6-1

TRUNNION SECTION PROPERTIES AND LOADS (TN-40 CASK)

	UPPER TRUNNIONS		LOWER TRUNNIONS	
ITEM	SECT A-A	SECT B-B	SECT A-A	SECT B-B
CROSS SECTION AREA, IN^2	93.415	79.73	54.95	46.0
AREA MOMENT OF INERTIA, IN^4	987.20	755.60	379.69	285.10
YIELD LIMIT LOAD*				
SHEAR FORCE LB.	750,000		375,000	
BENDING MOMENT, IN-LB.	4,185,000	1,312,500	2,092,500	656,250
ULTIMATE LIMIT LOAD **				
SHEAR FORCE LB.	1,250,000		625,000	
BENDING MOMENT, IN-LB.	6,975,000	2,187,500	3,487,500	1,093,750

*TRUNNION LOADS TO SUPPORT 6 TIMES CASK WEIGHT

**TRUNNION LOADS TO SUPPORT 10 TIMES CASK WEIGHT

TABLE 4A.6-2

TRUNNION STRESSES WHEN LOADED BY *6 AND **10 TIMES CASK WEIGHT

LOCATION /STRESS	*YIELD LIMIT		** ULTIMATE LIMIT	
	SECT A-A (NOTE 1)	SECT B-B (NOTE 1)	SECT A-A (NOTE 1)	SECT B-B (NOTE 1)
UPPER TRUNNIONS				
SHEAR STRESS (psi)	8,029	9,407	13,381	15,678
BENDING STRESS	25436	9,771	42,393	16,285
STRESS INTENSITY	30,080	21,200	50,134	35,333
LOWER TRUNNIONS				
SHEAR STRESS	6,824	8,152	11,374	13,589
BENDING STRESS	26,178	10,220	43,629	17,033
STRESS INTENSITY	29,522	19,242	49,203	32,074
ALLOWABLE STRESS	S_y = 31,900 psi		S_u = 70,000 psi	

NOTE

1. Sections A-A and B-B are shown on Figure 4A.6-1.

TABLE 4A.7-1

STRESS IN OUTER SHELL AND CLOSURE PLATES (TN-40 CASK)

01136263

LOCATIONS	STRESS INTENSITIES (1) (psi)
AT JUNCTION TOP PLATE TO VESSEL	12,404
REMAINDER OF TOP PLATE	21,117
AT JUNCTION BOTTOM PLATE TO VESSEL	7,112
REMAINDER OF BOTTOM PLATE	5,304

(1) 25 PSI + 3G handling in vertical position loading

Figure Withheld Under 10 CFR 2.390

Figure 4A.1-1

TN-40 CASK BODY KEY DIMENSIONS

PRAIRIE ISLAND ISFSI SAFETY ANALYSIS REPORT

TN-40 CASK BODY KEY DIMENSIONS			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG4A.1-1_REV_3
	PAGE. NO.	DATE: 04-19-06	

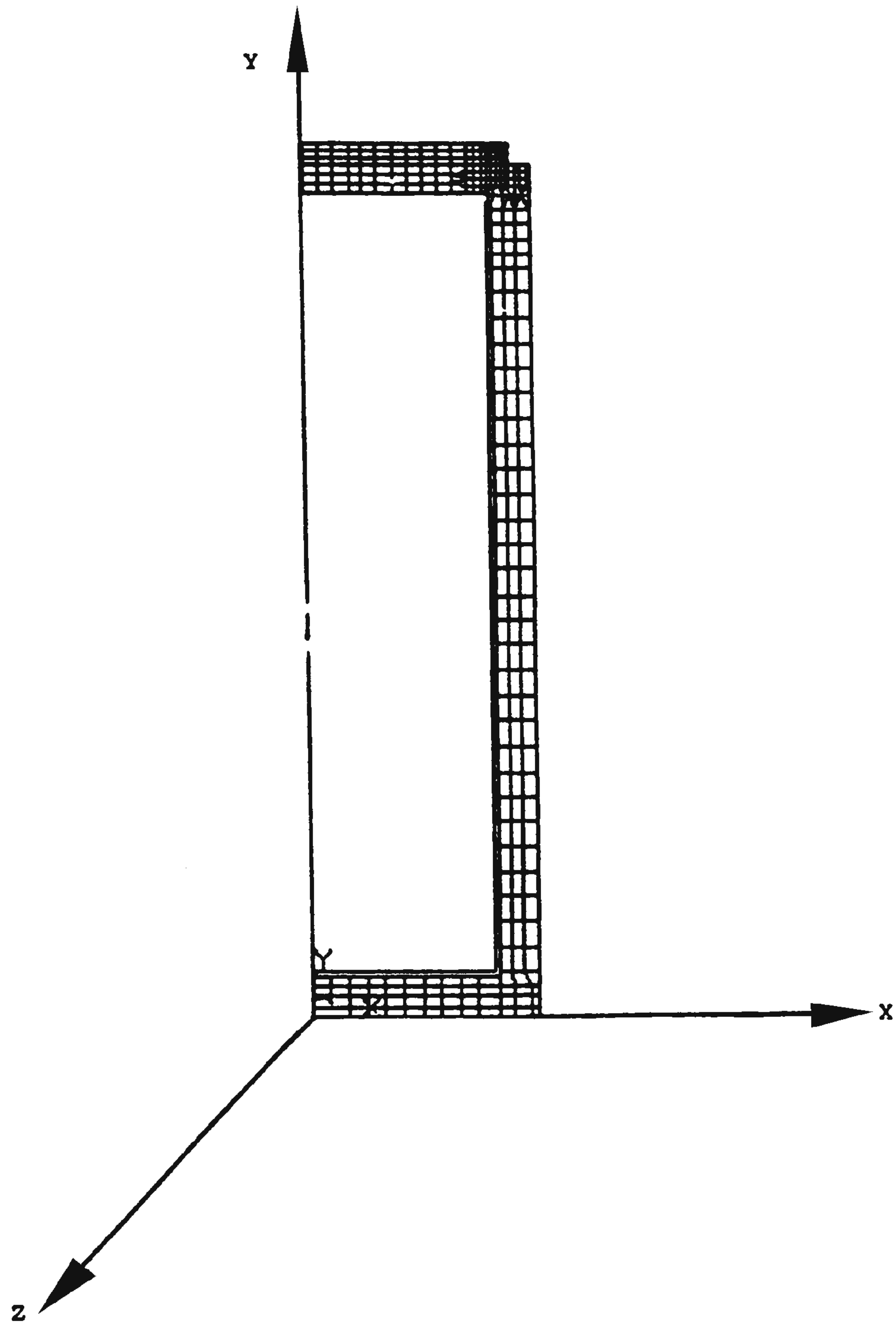


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Figure 4A.3-1

TN-40 CASK BODY - ANSYS MODEL

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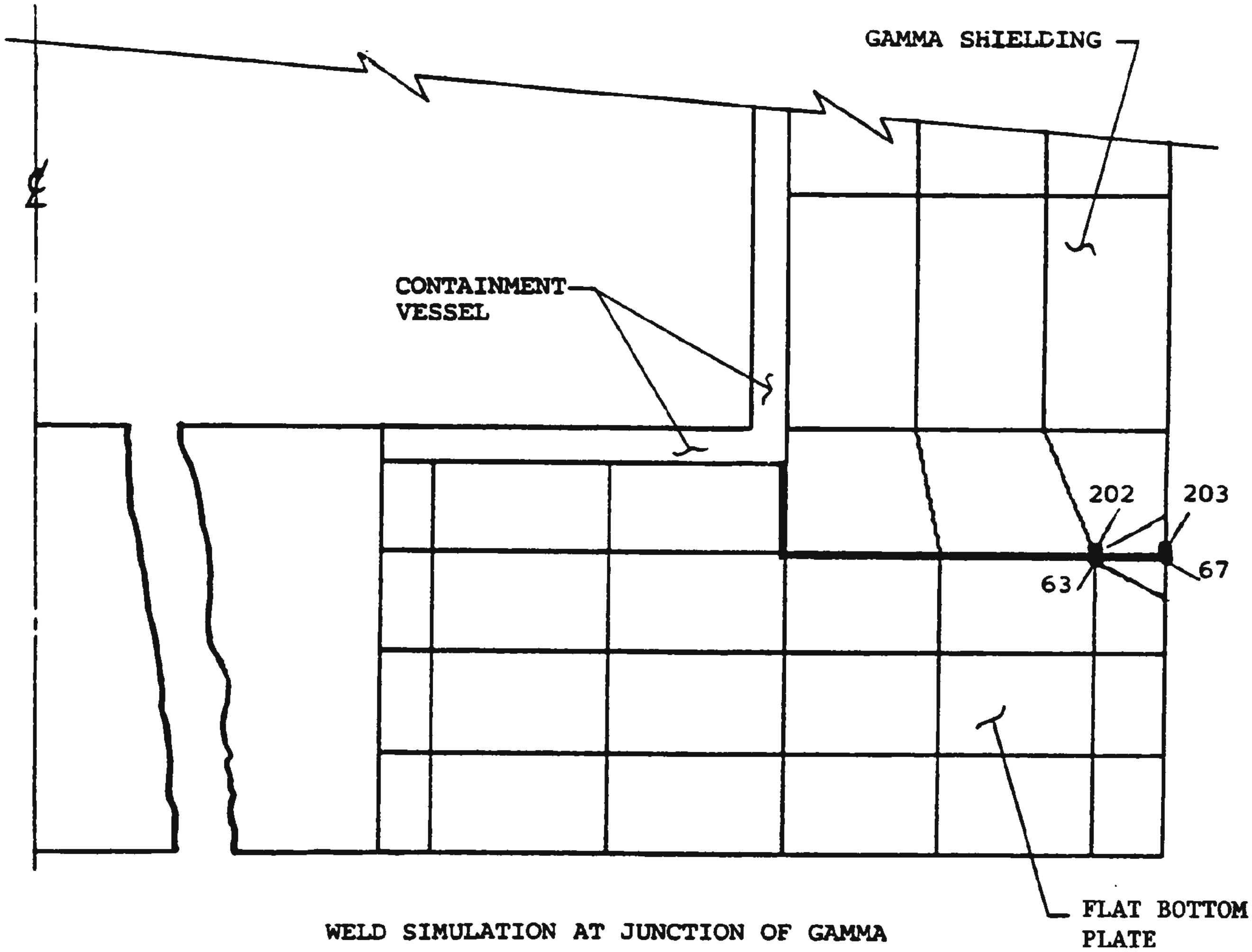
TN-40 CASK BODY - ANSYS MODEL

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG4A.3-1_REV_0
	PAGE. NO.	DATE: 04-19-06	



WELD SIMULATION AT JUNCTION OF GAMMA SHIELDING CYLINDRICAL SHELL TO FLAT BOTTOM BY COUPLING NODES 67-203 AND 63-202

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Figure 4A.3-2

TN-40 CASK BODY BOTTOM CORNER

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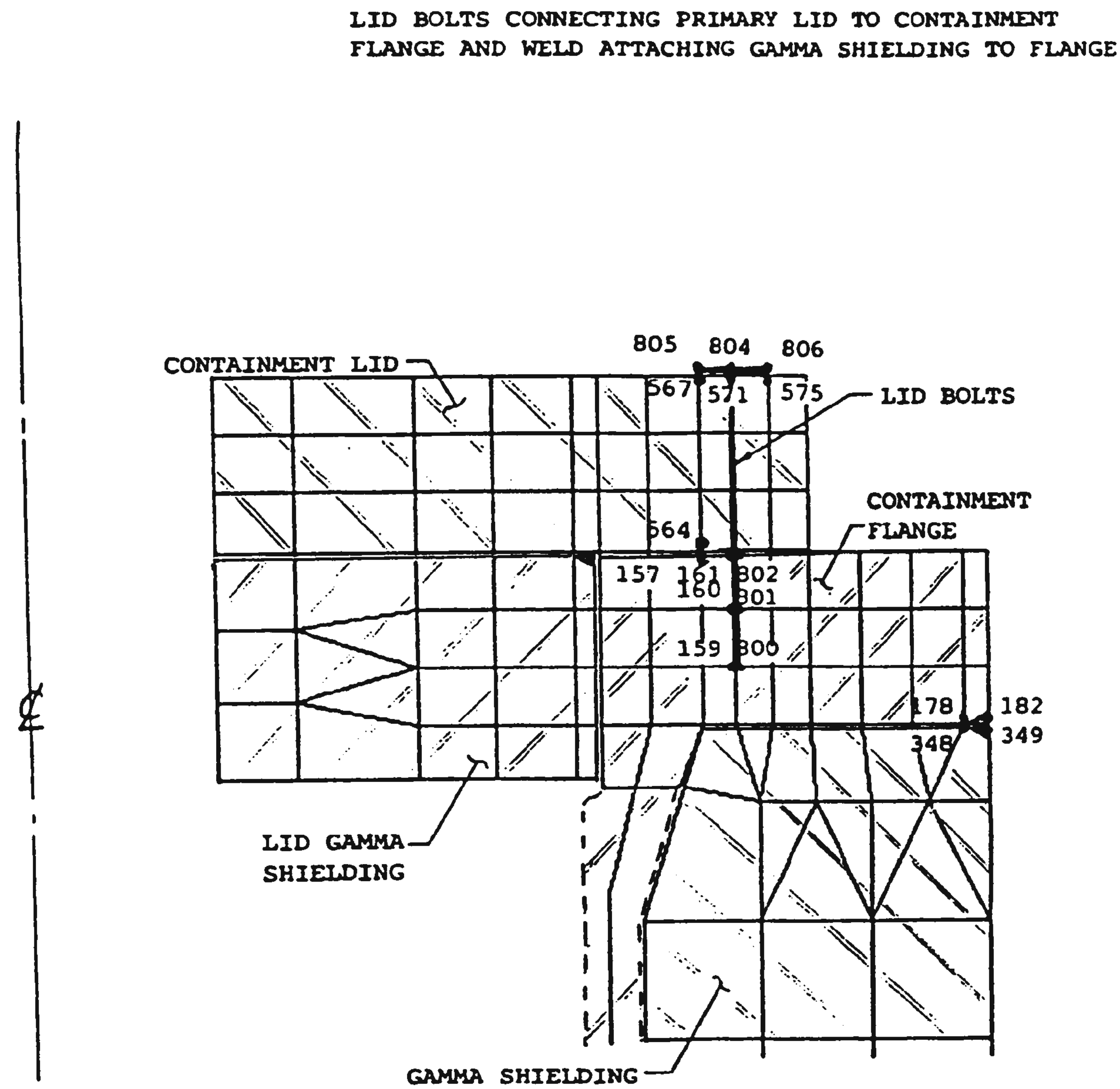

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Figure 4A.3-3

TN-40 CASK BODY TOP CORNER

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TN-40 CASK BODY - TOP CORNER

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 2	FIG4A.3-3_REV_2
	PAGE. NO.	DATE: 04-19-06	

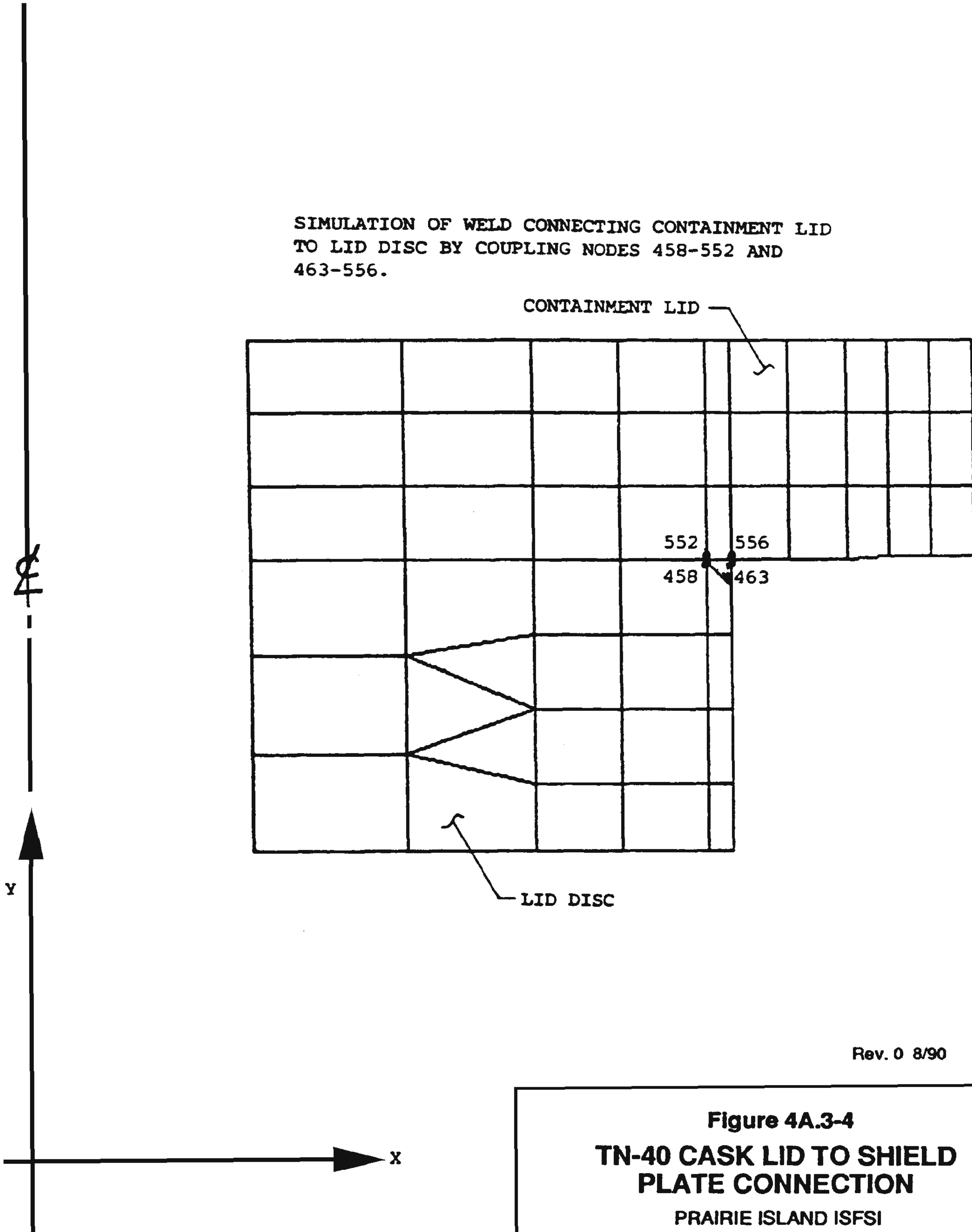

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Figure 4A.3-4

TN-40 CASK LID TO SHIELD PLATE CONNECTION

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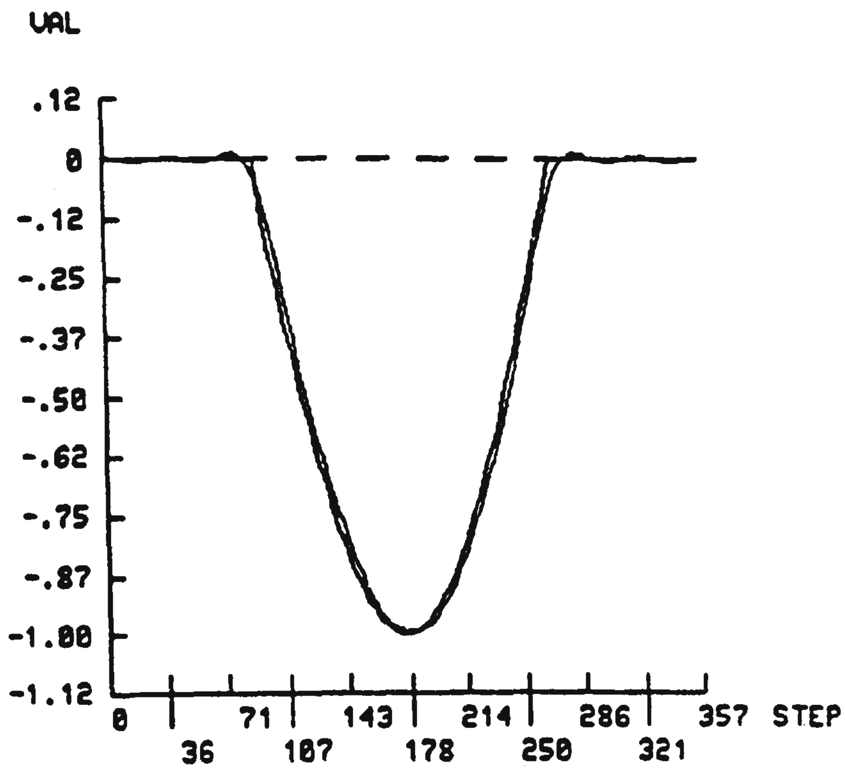
TN-40 CASK BODY LID TO SHIELD PLATE CONNECTION

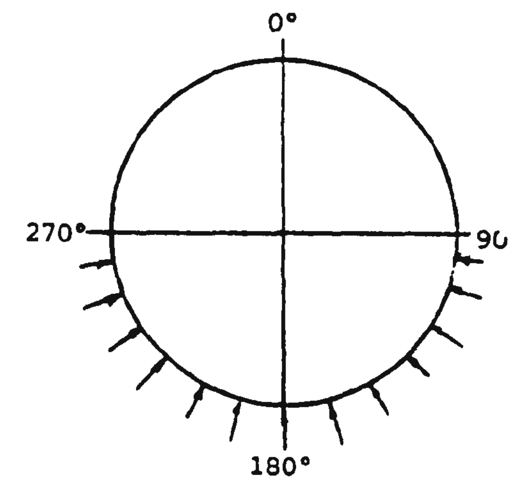
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG4A.3-4_REV_0
	PAGE. NO.	DATE: 04-19-06	

***** FOURIER COEFFICIENTS *****

TERM	MODE*ISYM	COEFFICIENT
1	0	-0.31780103E+00
2	1	0.49931477E+00
3	2	-0.21206110E+00
4	3	-0.22109896E-13
5	4	0.42528724E-01
6	5	0.38194530E-13
7	6	-0.18310180E-01
8	7	-0.27529862E-13
9	8	0.10237762E-01



1PR-9,FOURIER COEFF. GENERATION



FOURIER COEFFICIENTS FOR THE FUNCTION: $\cos\theta \Big|_{90°}^{270°}$

Figure 4A.3-5

FOURIER COEFFICIENTS

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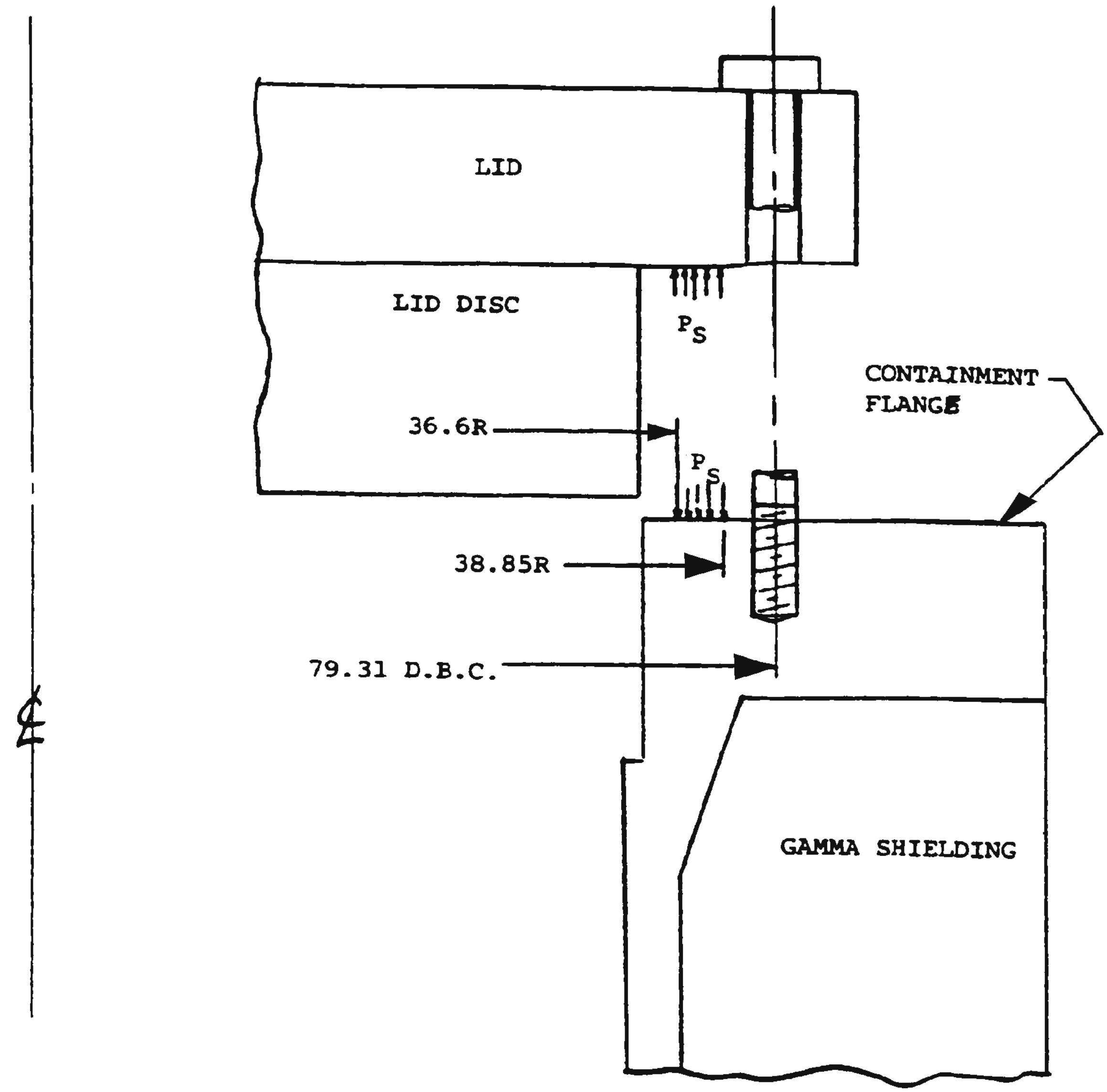
FOURIER COEFFICIENTS

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG4A.3-5_REV_0
	PAGE. NO.	DATE: 04-19-06	



SEAL PRESSURE CALCULATION

SEAL LOAD F_S = 2 x 2198 ~ 4400#/IN. AT D_{AV} = 75.1"

SEAL PRESSURE: $P_S = \dfrac{4400 \times 75.1 \times \pi}{\pi\,(38.85^2 - 36.6^2)} = 1946.48$ **PSI**

BOLT PRELOAD STRESS: 25000 PSI

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Figure 4A.3-6

TN-40 BOLT PRELOAD AND SEAL REACTION

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TN-40 BOLT PRELOAD AND SEAL REACTION

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG4A.3-6_REV_0
	PAGE. NO.	DATE: 04-19-06	

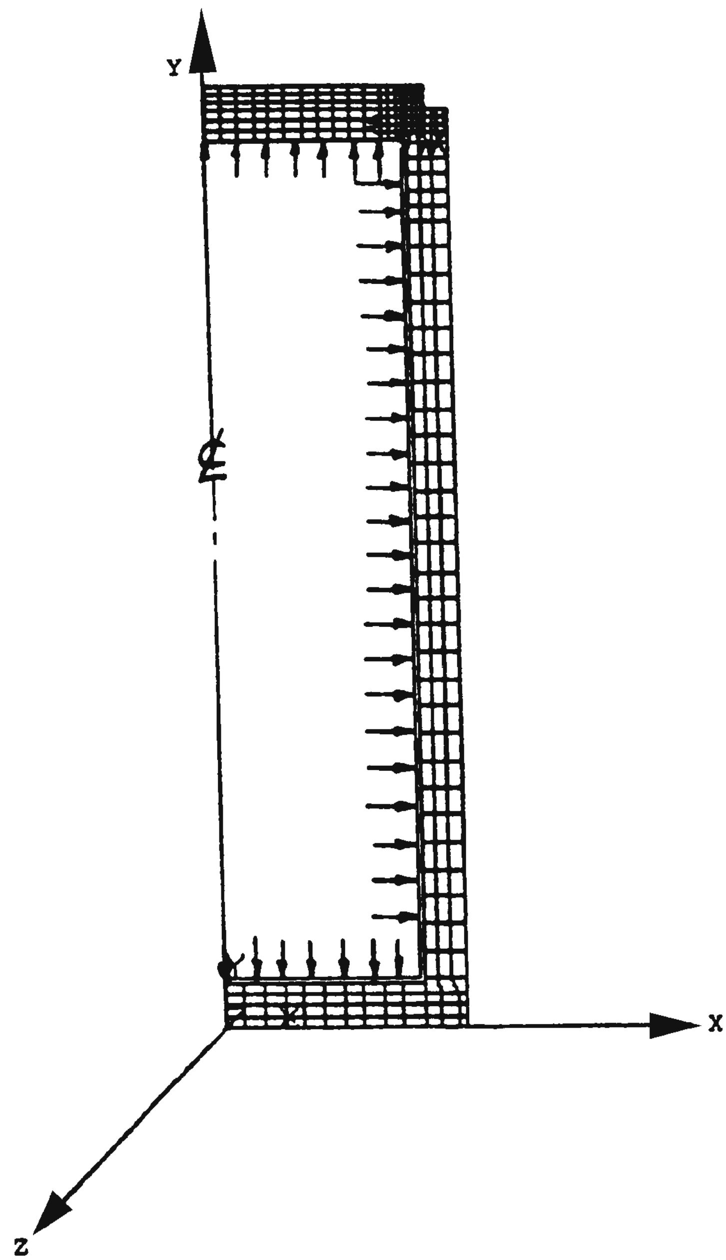

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Figure 4A.3-7

TN-40 DESIGN INTERNAL PRESSURE (100 PSIG)

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TN-40 DESIGN INTERNAL PRESSURE (100 PSIG)

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG4A.3-7_REV_0
	PAGE. NO.	DATE: 04-19-06	

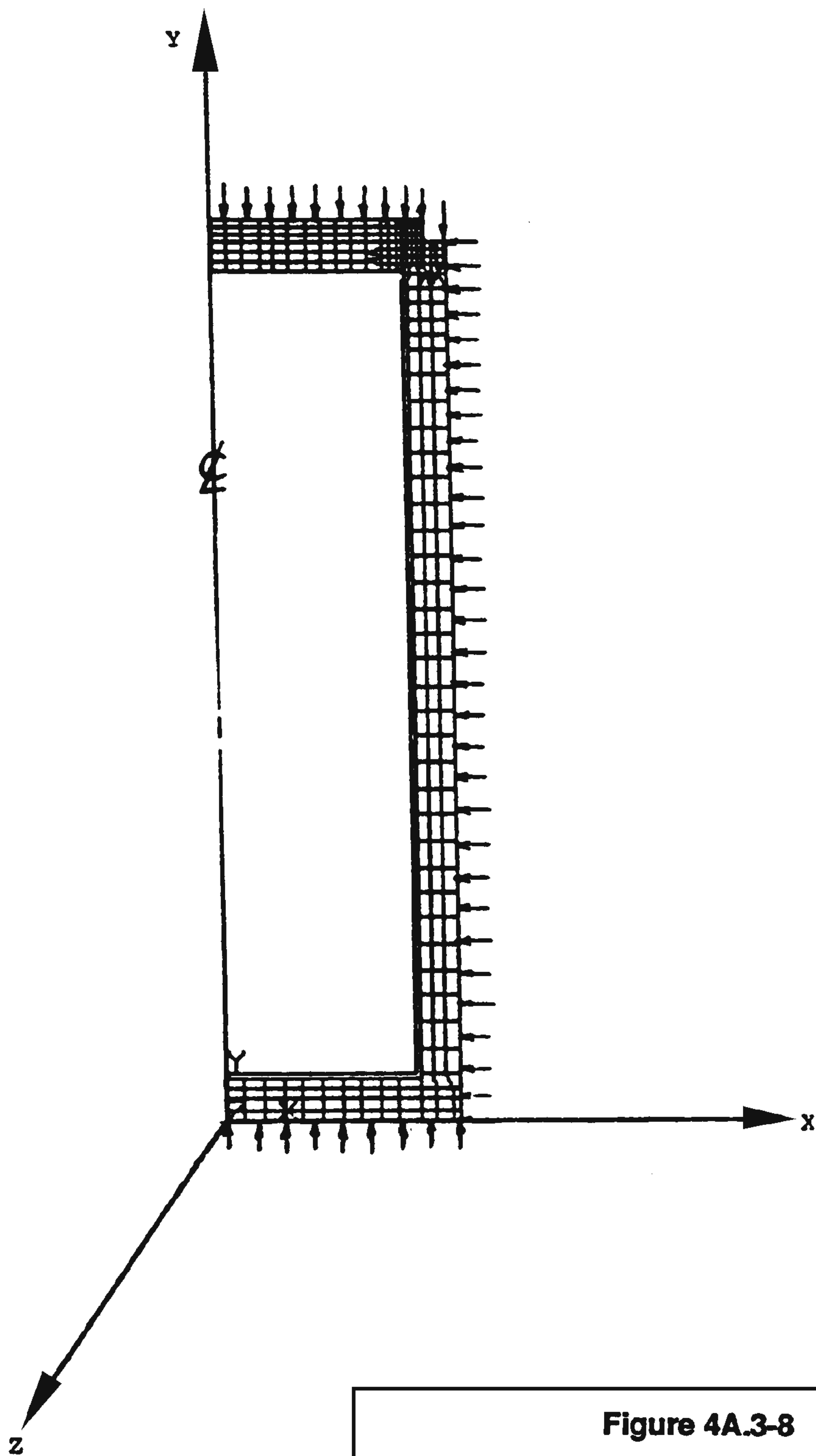

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Figure 4A.3-8

TN-40 EXTERNAL PRESSURE LOADING (25 PSIG)

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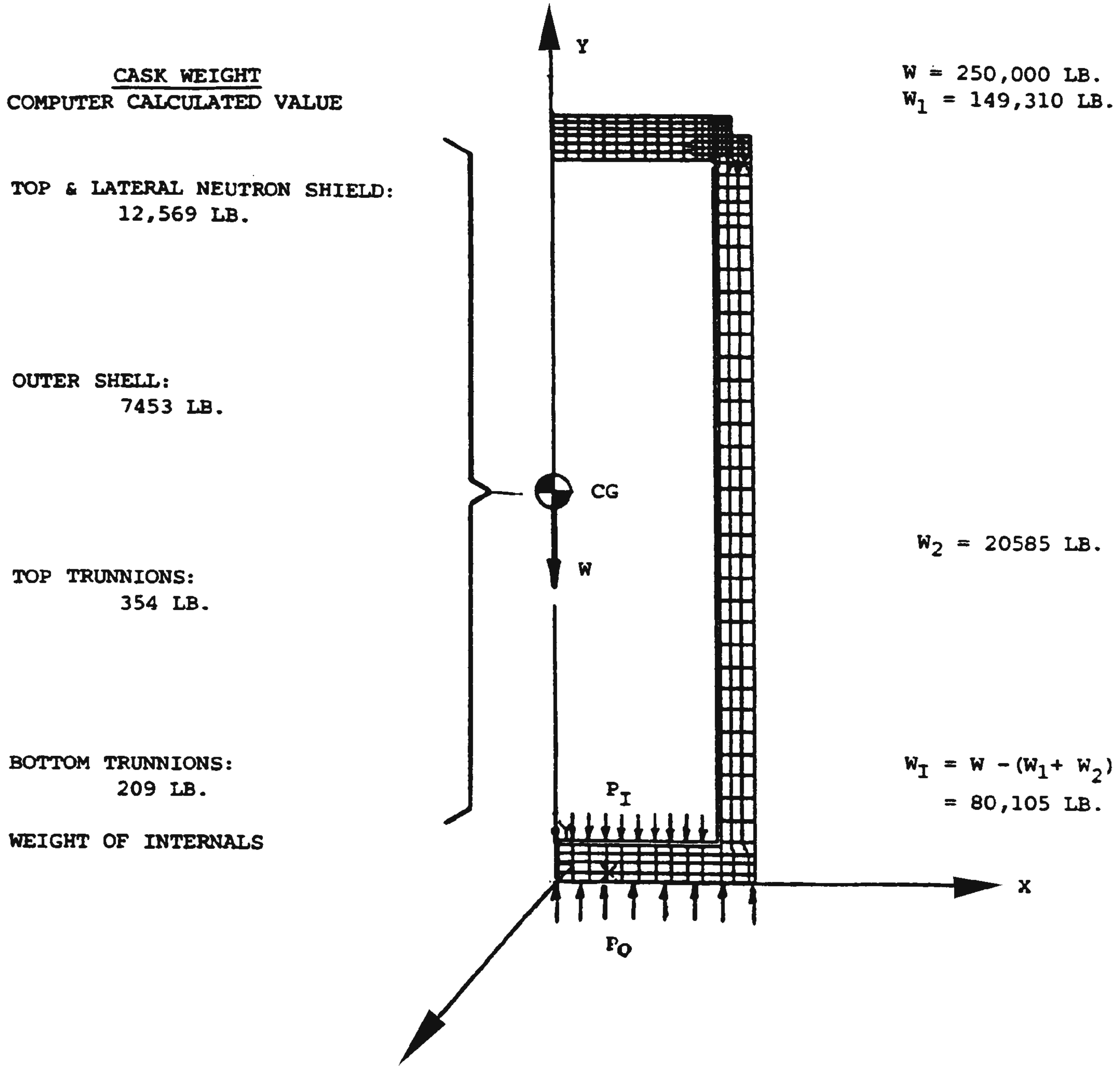

$P_O = \dfrac{250,000}{\pi \times 45.5^2} = 38.44$ PSI PRESSURE ON BOTTOM OUTER SURFACE

$P_I = \dfrac{80,105}{\pi \times 36^2} = 19.67$ PSI PRESSURE ON CONTAINMENT BOTTOM INNER SURFACE

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Figure 4A.3-9

TN-40 1g DOWN LOADING

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$$F_{TR} = \frac{3\,W}{2} = 3 \times \frac{250{,}000}{2} = 375{,}000 \text{ LB.}$$

$$F_B = 3(149{,}310 + 20{,}585) = 509{,}685 \text{ LB.}$$

$$P_I = \frac{80{,}105 \times 3}{\pi \times 36^2} = 59.02 \text{ PSI} \; ; \; 2F_{TR} + F_B + P_I A_I = 0$$

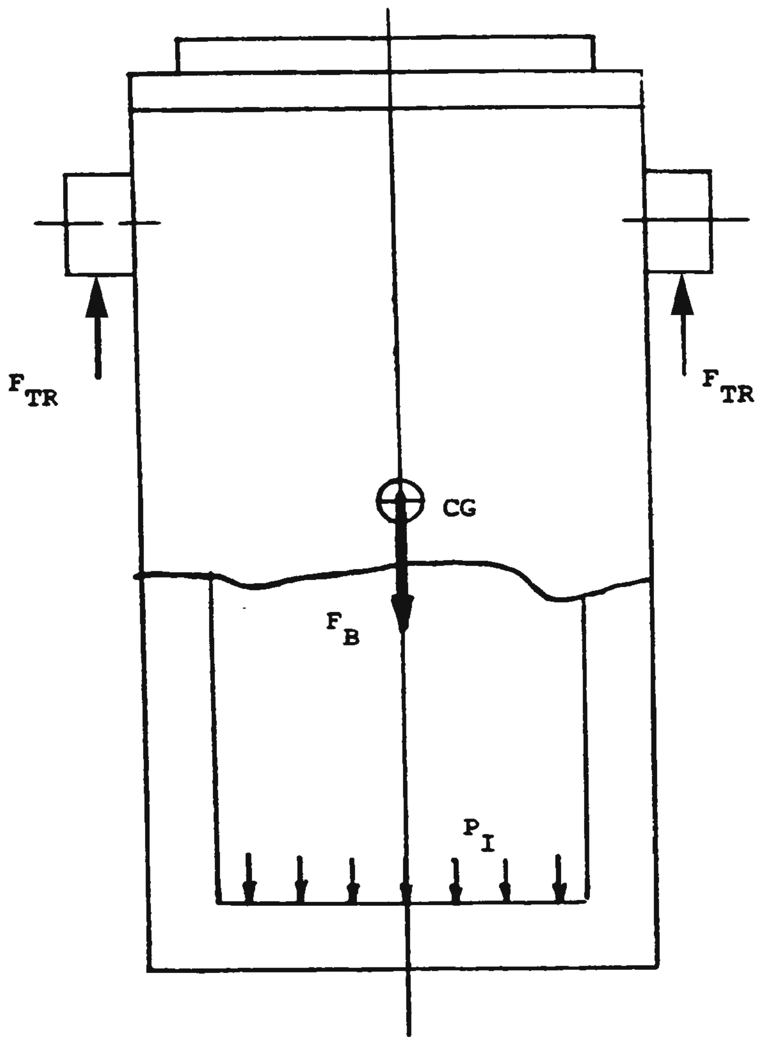

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Figure 4A.3-10

TN-40 LIFTING: 3g VERTICAL UP

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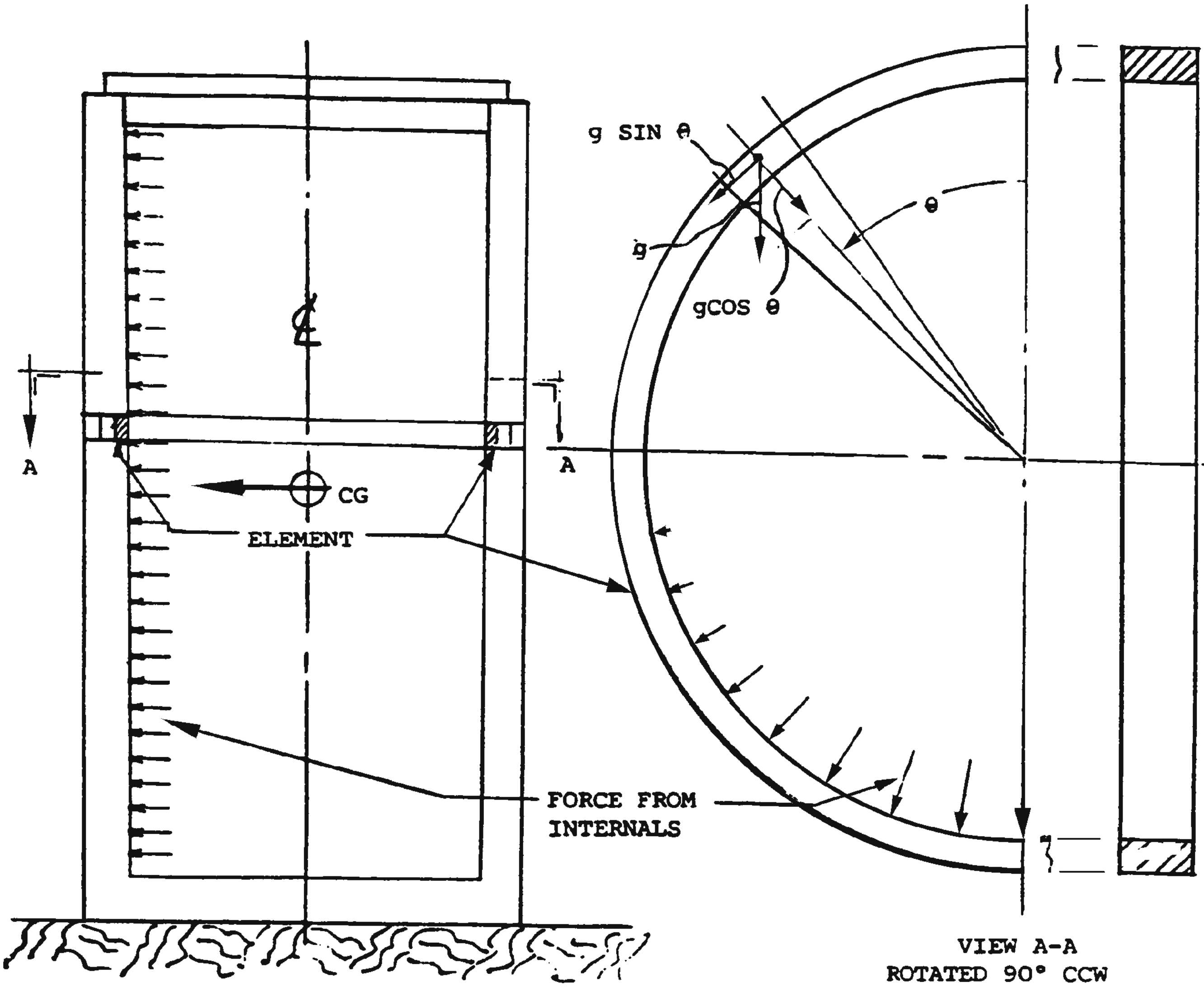

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Figure 4A.3-11

TN-40 1g LATERAL

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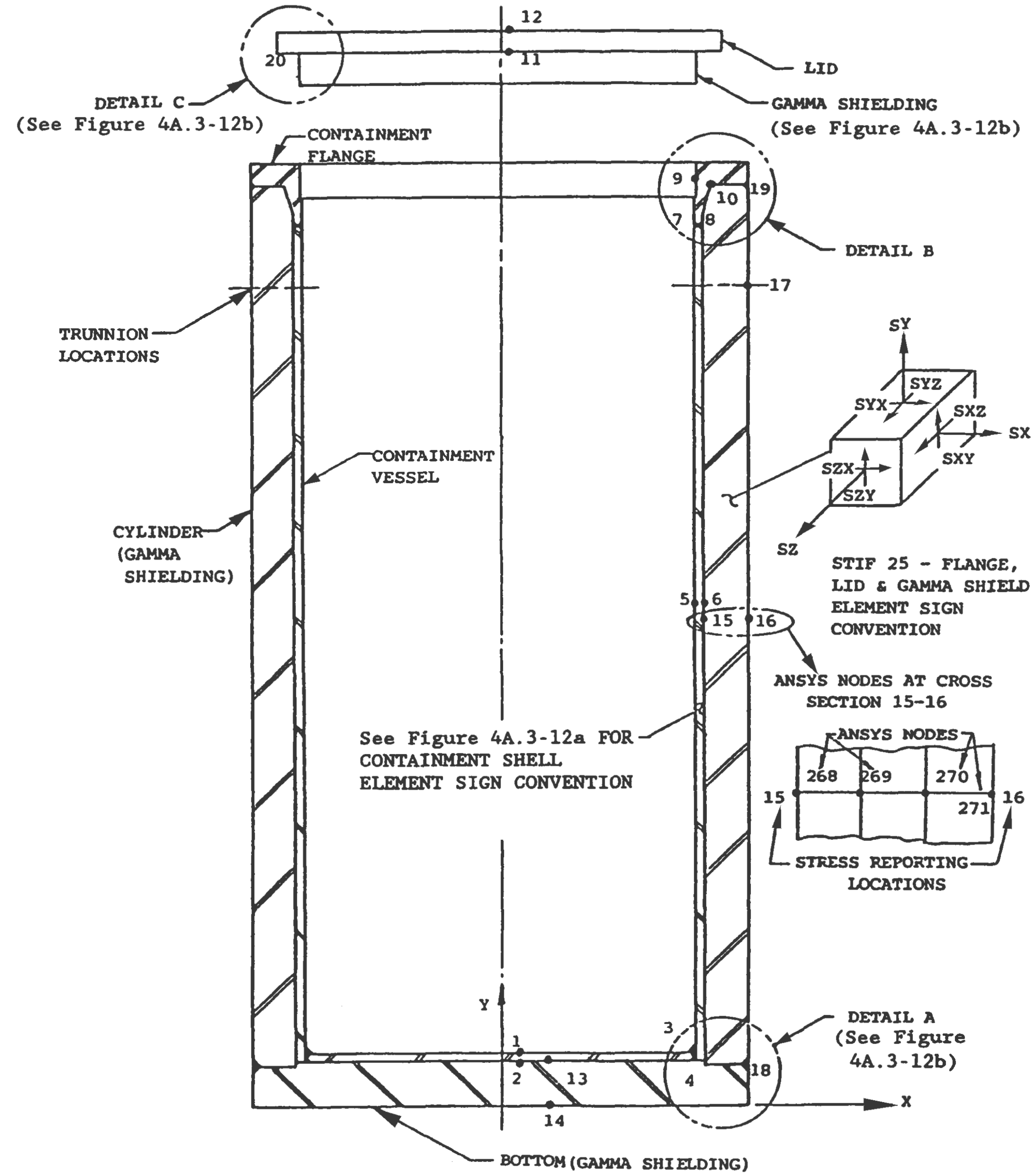
TN-40 1g LATERAL

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG4A.3-11_REV_0
	PAGE. NO.	DATE: 04-20-06	



Rev 3 4/94

Figure 4A.3-12
STANDARD REPORTING LOCATIONS FOR TN-40 CASK BODY
PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

STANDARD REPORTING LOCATIONS FOR TN-40 CASK BODY			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG4A.3-12_REV_3
	PAGE. NO.	DATE: 04-20-06	

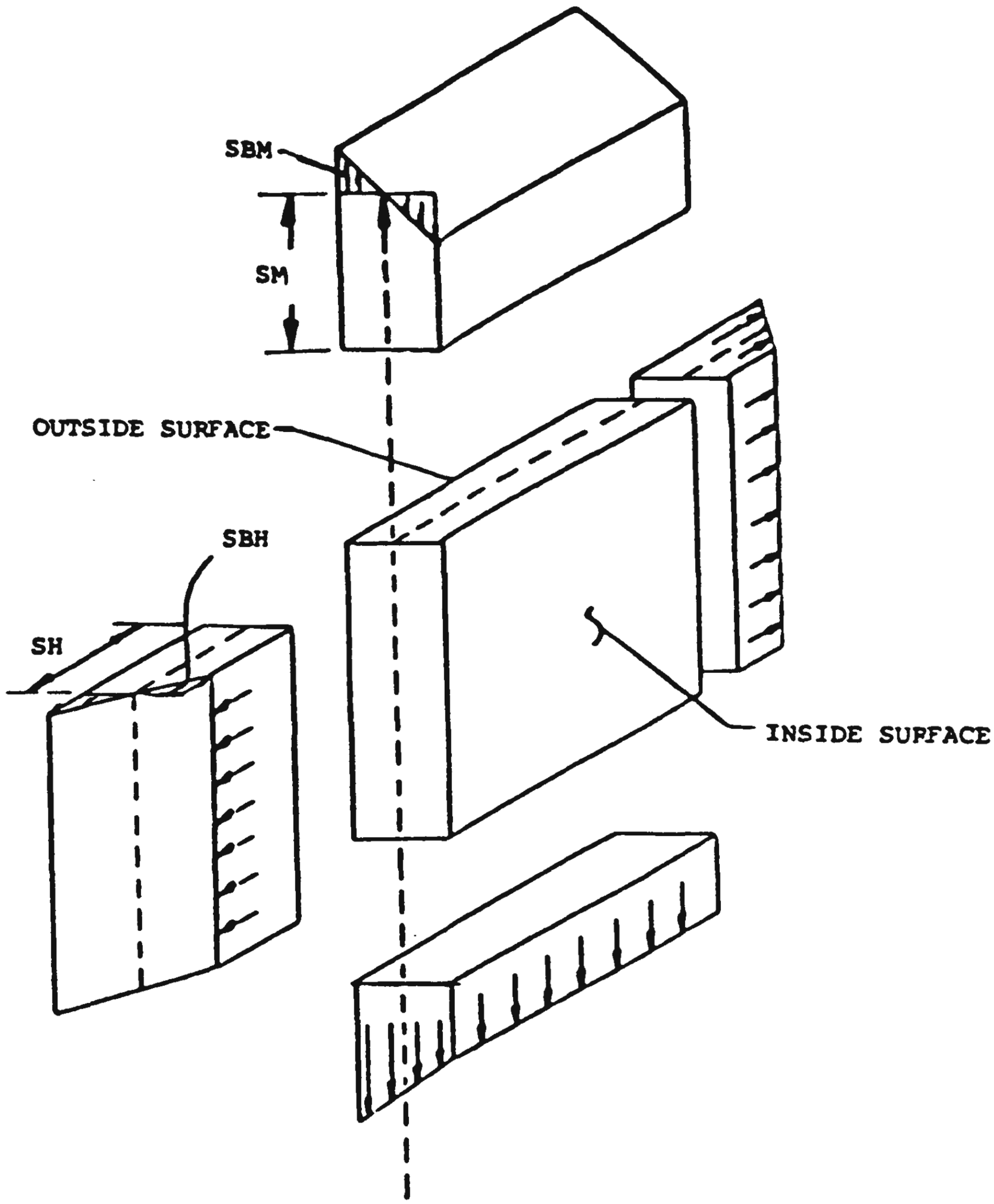


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Figure 4A.3-12a

STIF 61 CONTAINMENT VESSEL ELEMENT SIGN CONVENTION

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STIF 61 CONTAINMENT VESSEL ELEMENT SIGN CONVENTION

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG4A.3-12a_REV_0
	PAGE. NO.	DATE: 04-20-06	

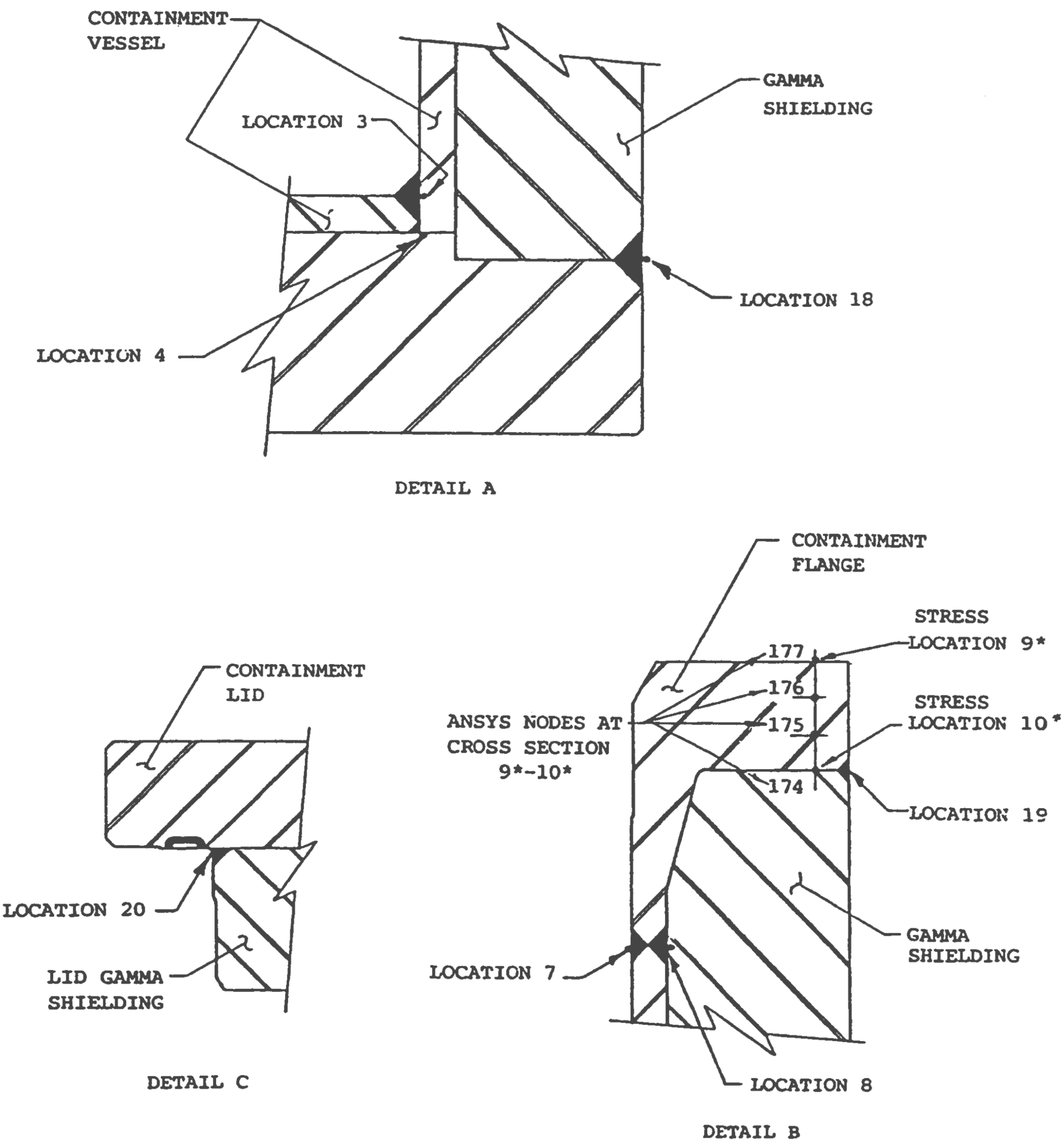

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Figure 4A.3-12b

TN-40 WELD STRESS LOCATIONS

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TN-40 WELD STRESS LOCATIONS

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 2	FIG4A.3-12b_REV_2
	PAGE. NO.	DATE: 04-20-06	

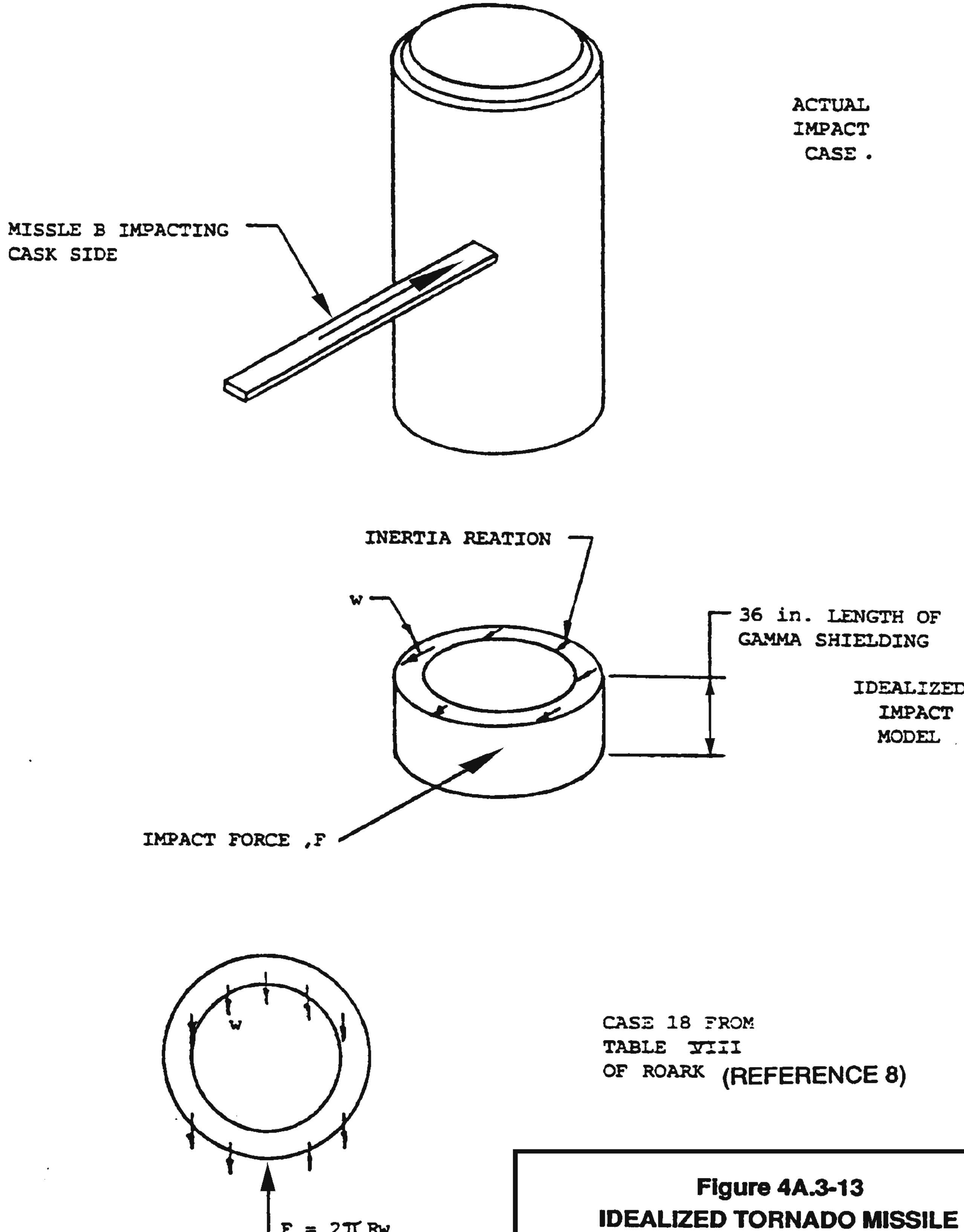

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Figure 4A.3-13

IDEALIZED TORNADO MISSILE IMPACT MODEL OF GAMMA SHIELD CYLINDER

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IDEALIZED TORNADO MISSILE IMPACT MODEL OF GAMMA SHIELD CYLINDER

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG4A.3-13_REV_0
	PAGE. NO.	DATE: 04-20-06	

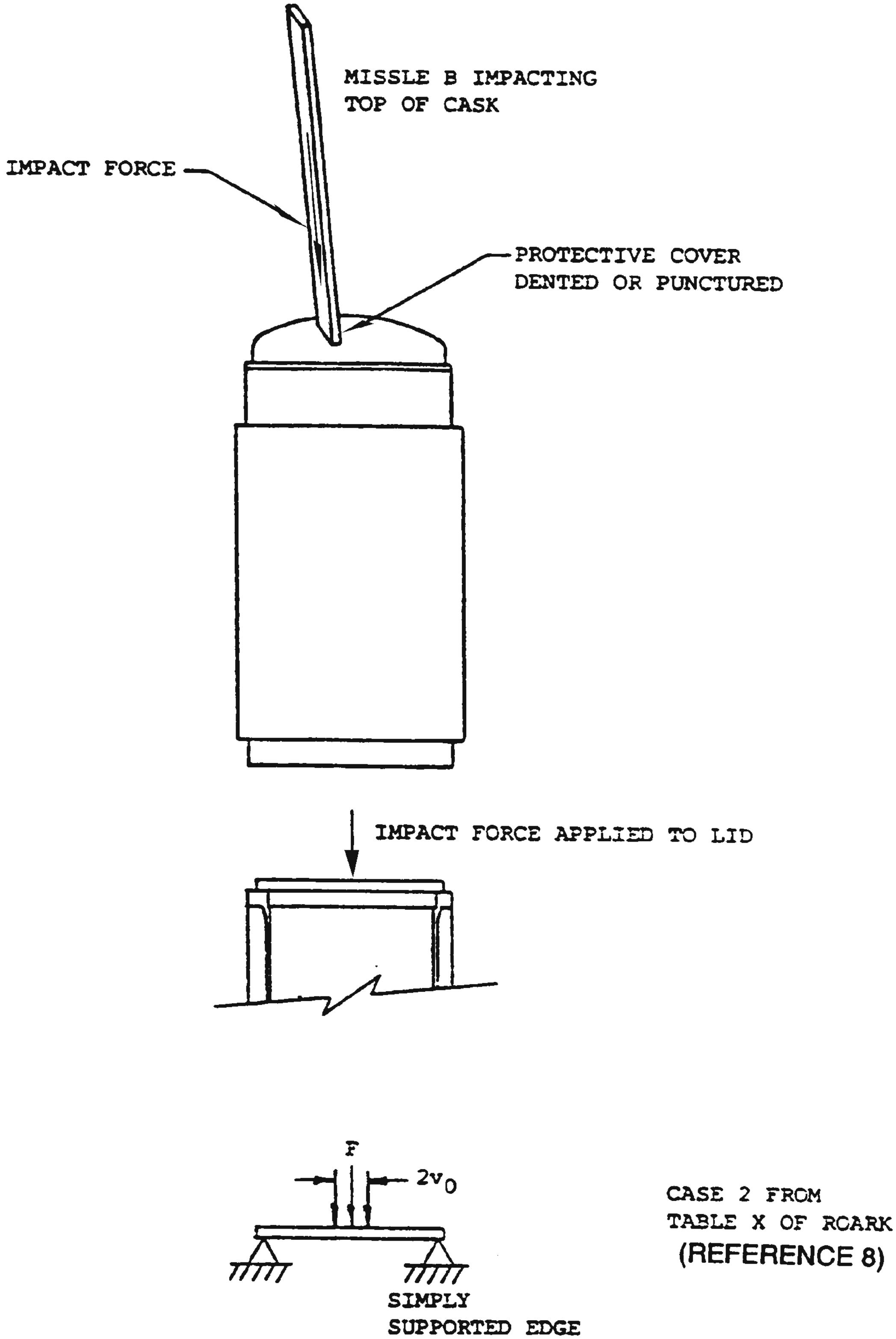

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Figure 4A.3-14

IDEALIZED TORNADO MISSILE IMPACT MODEL OF LID

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IDEALIZED TORNADO MISSILE IMPACT MODEL OF LID

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG4A.3-14_REV_0
	PAGE. NO.	DATE: 04-20-06	

Figure 4A.4-1
SUMMARIZING THE BOLT END MOTIONS DUE TO 100 PSIG PRESSURE IN THE CASK CAVITY

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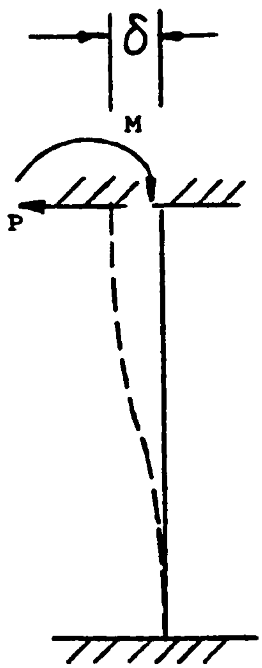
Figure Withheld Under 10 CFR 2.390

SUMMARIZING THE BOLT END MOTIONS DUE TO 100 PSIG PRESSURE IN THE CASK CAVITY			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG4A.4-1_REV_0
	PAGE. NO.	DATE: 04-20-06	



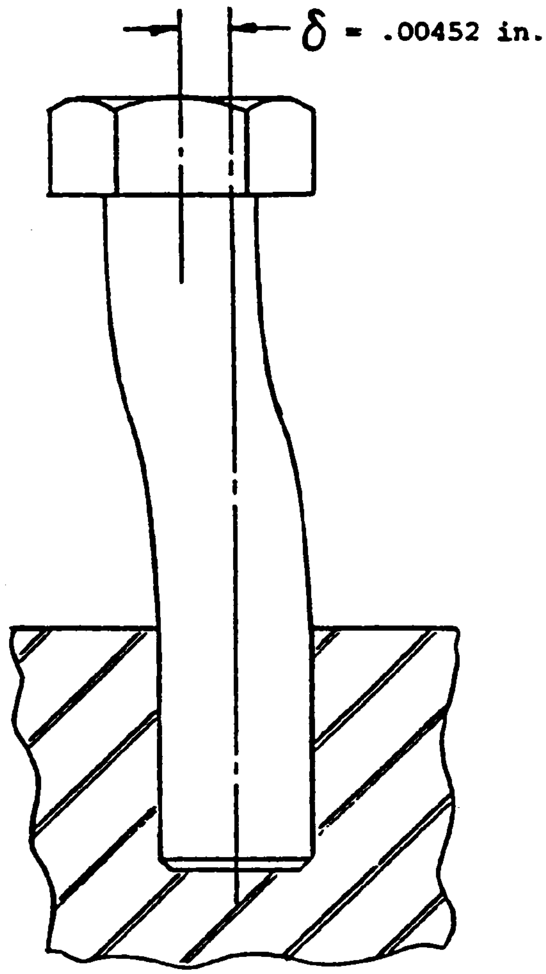


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Figure 4A.4-2
LID BOLT BENDING DUE TO LID EDGE ROTATION UNDER INTERNAL PRESSURE
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LID BOLT BENDING DUE TO LID EDGE ROTATION UNDER INTERNAL PRESSURE

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG4A.4-2_REV_0
	PAGE. NO.	DATE: 04-20-06	

FIGURE 4A.5-1, REV. 1

DELETED

FIGURE 4A.5-2, REV. 1

DELETED

FIGURE 4A.5-3, REV. 1

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FIGURE 4A.5-4, REV. 1

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FIGURE 4A.5-5, REV. 1

DELETED

FIGURE 4A.5-6, REV. 1

DELETED

FIGURE 4A.5-7, REV. 1

DELETED

Figure Withheld Under 10 CFR 2.390

Figure 4A.6-1

TN-40 TRUNNION GEOMETRY

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SAFETY ANALYSIS REPORT

TN-40 TRUNNION GEOMETRY			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: KJF	REVISION: 4	FIG4A.6-1_REV_4
	PAGE. NO.	DATE: 9-22-11	

Figure Withheld Under 10 CFR 2.390

Figure 4A.7-1

TN-40 CASK OUTER SHELL AND CONNECTION WITH CASK BODY

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TN-40 CASK OUTER SHELL AND CONNECTION WITH CASK BODY			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG4A.7-1_REV_0
	PAGE. NO.	DATE: 04-20-06	

Figure Withheld Under 10 CFR 2.390

Figure 4A.7-2

LOAD DISTRIBUTIONS AND MODELS USED FOR ANALYSIS FOR TN-40 OUTER SHELL

PRAIRIE ISLAND ISFSI SAFETY ANALYSIS REPORT

LOAD DISTRIBUTIONS AND MODELS USED FOR ANALYSIS FOR TN-40 OUTER SHELL			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG4A.7-2_REV_0
	PAGE. NO.	DATE: 04-20-06	

APPENDIX 4B

STRUCTURAL ANALYSIS OF THE TN-40 BASKET

4B.1 INTRODUCTION

This appendix presents the structural analysis of the TN-40 fuel support basket. The basket is a welded assembly of stainless steel boxes. The fuel compartment stainless steel box sections are attached together locally by cylindrical stainless steel plugs (that actually pass through the aluminum and boral plates) that are fusion welded to both adjacent box sections. The boral and aluminum plates are thus sandwiched between the stainless steel walls of adjacent box sections. The basket contains 40 compartments for proper spacing and support of the fuel assemblies.

The deformations and stresses induced in the basket structure due to the applied lateral loads are determined using the ANSYS computer program (Reference 1). The most severe loading for which the basket was evaluated is the 50 g vertical loading of the basket evaluated to represent a hypothetical end drop accident. Also a 3 g loading was applied to the basket in both lateral and vertical directions as a bounding load to represent both Design and Level A Conditions. In addition, primary plus secondary (thermal) stresses due to differential expansion were evaluated against Level A limits. The inertial loads of the fuel assemblies were applied to the basket structure as distributed loads applied to the plate surfaces. Quasistatic stress analyses were performed with applied loads in equilibrium with the reactions at the periphery of the basket. The calculated stresses in the basket structure were compared with the stress limits to demonstrate that the established design criteria are met.

Geometry

The details of the TN-40 basket are shown on Figures 1.3-6 and 1.3-7. As described above, the basket structure consists of an assembly of stainless steel boxes or cells joined by fusion welded steel plugs and separated by aluminum and boral poison plates. The stainless, aluminum and boral wall between fuel compartments is effectively a sandwich panel. The panel consists of two 13 gauge (0.088 in.) thick 304 stainless plates and two 0.25 in. thick 6061-T6 aluminum plates surrounding the 0.075 in. thick boral poison plate. The aluminum provides the heat conduction path from the fuel assemblies to the cask cavity wall, and the poison material provides the necessary criticality control.

A representative basket wall panel between fuel compartments is shown in Figure 4B.2-1. The panel plates are fastened together at discrete locations (2 attachments every 8 inches) along their lengths. The adjacent fuel compartment stainless steel walls are fusion welded to cylindrical plugs that pass through holes in the boral and aluminum plates. This method of construction forms a very strong honeycomb-like structure of boxes. The open dimension of each fuel compartment cell or box is 8.05 in. x 8.05 in. which provides a minimum of 1/8 in. clearance around the fuel assemblies. The pitch of the cells is approximately 8.85 in. The overall basket length (160 in.) is less than the cask cavity length to allow for thermal expansion and tolerances.

Several of the aluminum conductor plates are continuous across the diameter of the basket to provide uninterrupted heat conduction paths. Other shorter plates are provided between and perpendicular to these continuous plates. Some of the aluminum plates are as short as one cell dimension in width.

Structural rails oriented parallel to the axis of the cask are attached to the inner cavity wall of the cask body to establish and maintain basket orientation, to prevent twisting of the basket assembly, and to support the edges of those plates adjacent to the rails which would otherwise be free to slide tangentially around the cask cavity wall under lateral inertial loadings.

Weight

A conservative value of 1,330 lb. is assumed for the weight of each fuel assembly. Under lateral inertial loading each assembly is assumed to be uniformly supported across the width and along the length of the basket wall. The inertia of the basket structure (weight of the basket x g load) is also included in the analysis.

Temperature

Thermal analyses were conducted to obtain the temperature distributions in the basket for various conditions. These analyses are presented in Chapter 3. Stress analysis of the basket using the thermal results are described below in Section 4B.6.

Material Properties

The material properties of the 304 stainless steel plates are taken from the ASME Code, Section III Appendices (Reference 2). The material properties of the aluminum alloy (6061-T-6) are also taken from the above reference except at elevated temperature. The elevated temperature properties not available in this reference are obtained from the Aluminum Association data (Reference 3). These properties are listed with specific references in Table 4B.1-1 and 4B.1-2. The material properties used in the finite element model analysis are listed in Table 4B.1-3. The maximum calculated basket temperatures and the temperatures selected for evaluation are listed in Table 4B.1-4. The full strength of the aluminum was considered when performing dynamic impact analyses. For long term sustained loading the aluminum strength is generally neglected under primary loading where it can share the load with the stainless steel.

4B.2 BASKET FINITE ELEMENT MODEL DEVELOPMENT (FOR SIDE IMPACT ANALYSIS)

Approach

The TN-40 basket is a large, complex redundant structure. The basket has 40 fuel compartments 8.05 in. x 8.05 in. x 160 in. long. There are 65 walls between adjacent fuel compartments, 30 exterior compartment walls and numerous radial and circumferential members. In addition, each wall between fuel compartments has 5 different layers of material which are attached to each other locally at 40 locations. Finally, the different layers (or plates) of material are free to slide relative to each other except at the attachment locations. Therefore a careful approach was needed to adequately model the basket while avoiding an excessively large computer model.

Since the major purpose of the analysis was to perform a side impact analysis, a unit slice of the basket cross section perpendicular to the cask longitudinal axis was modeled to represent the basket system. The basket has one axis of symmetry; therefore only half of the slice was modeled. Also, because the basket wall panels are quite complex (5 layers of material all connected differently at the edges and connected together only at the plugs and welds) a simplified wall panel substructure was developed for use in the half slice system model.

Panel Models

Figure 4B.2-1 shows a typical wall panel between adjacent fuel compartments. Note the 5 layers of material: 0.1 in. stainless steel (fuel compartment box side), 0.25 in. aluminum, 0.075 in. boral, 0.25 in. aluminum and another 0.1 in. stainless steel layer (adjacent fuel compartment box side). Note the plugs and welds in the lower enlarged detail. The plugs are spaced 5 in. apart in this section and 8 in. apart along the axis of the basket. Also note that, at the edges of the panel, the 0.1 in. stainless plate edges are actually the corners of the box sections. The aluminum plates sometimes end at the edges of the panel and in some cases continue beyond the panel edges into the next panel. The edge conditions at any panel depend on how these 5 layers or plates are attached to the corresponding plates in the adjacent panels.

First, a detailed wall panel substructure finite element model was developed as illustrated schematically in Figure 4B.2-2. Figure 4B.2-3 is an ANSYS computer plot of this same model showing the elements properly scaled. This 876 element model carefully represents all of the panel components including the plugs and welds. The plates are modeled using ANSYS STIF 63 shell elements and the plugs using STIF 16 pipe elements. The edges of the detailed model terminate at the panel edges (compare Figures 4B.2-1 and 4B.2-3). The finite element model is 8 in. long along the axial direction of the basket and terminates midway between plugs and welds. Symmetry boundary conditions are applied at these front and rear planes (4 in. in each direction from the plugs and welds). The individual plates in the model are connected together in plane only at the plugs and welds (the plugs are rigidly connected to the 0.1 in. stainless steel plates and coupled to nodes in the aluminum and boral plates). The nodes of the various plates are coupled together in the out of plane direction so that they will bend in unison under surface pressure loading. This model is correct and complete and can be used to perform detailed wall panel stress analyses for various loadings simulating fuel assembly inertial reactions applied to the basket if the edge conditions of the various plates can be determined.

Next a simplified wall panel substructure with far fewer elements was developed. The simplified panel was intended for use in the system model of the basket cross section described above. The simplified model is illustrated in Figure 4B.2-4 and a computer plot of the model is provided in Figure 4B.2-5. This simplified model uses ANSYS STIF 91 laminated elements (which have shear deflection capability) to represent all of the various plates in the 5 in. wide region between weld lines and also to represent the aluminum and boral plates outboard of the weld line. The stainless steel box section corners are modeled using STIF 63 shell elements exactly as in the detailed panel model. The weld lines along the axial direction are modeled as rigid members (discrete local plugs and welds are not modeled). Based on the compression load tests described in Appendix 4C, the row of welds and plugs do in fact fix the 0.1 in. stainless steel plates to the aluminum plates along the weld lines.

The reviewer should note that the simplified panel substructure has approximately the same stiffness (stiffness matrix if you prefer) as the detailed model. This could have been accomplished by developing an array of springs or specifying a matrix of numbers to represent the panels in the ANSYS basket system model. The details of the simplified panel model (such as the continuous rigid weld line or the use of laminated elements to represent multiple plates) are not important since this simplified panel model is used only in the system model (described below) and is not used for panel stress analysis.

The detailed and simplified panel substructure models (Figures 4B.2-3 and 4B.2-5) were both subjected to a surface pressure loading simulating a fuel assembly lateral inertial reaction on the basket plates. Cases were run for both models with the edges simply supported and fixed. As would be expected, the simplified model with laminated elements (perfect shear connection) was stiffer and deformed less than the detailed sliding plate model. The stiffness of the simplified panel was decreased until its performance agreed well with that of the detailed panel by decreasing the shear modulus of the aluminum layers in the laminated elements (inducing greater shear deflections in these elements)

Figure 4B.2-6 shows the center deflection, weld line deflection and edge rotation of the simplified panel model plotted vs. laminated element shear modulus for the simply supported case. The horizontal lines at the right edge of the figure show the detailed panel results. Figure 4B.2-7 shows the panel center moment comparison for this simply supported case. The moment is insensitive to shear modulus and there is excellent agreement between both models and classical theory. Figure 4B.2-8 shows the center and edge moments for the fixed edge case. Agreement between models and theory is excellent for a shear modulus range (laminated elements only) of 10,000 to 14,000 psi. A shear modulus of 11,000 psi was selected for subsequent analyses to be performed with the simplified panel model.

Basket System Model

A basket system model, shown in Figure 4B.2-9, was then developed by assembling an array of these simplified panels (with shear modulus of 11,000 psi). This system model is an extremely large and complex ANSYS model. The fuel compartment corners and basket periphery are carefully modeled to define each plate connection. Interface elements are provided between the corner nodes of the stainless steel shell elements to simulate the through thickness support provided by the aluminum. This system model was used to determine the response of the entire basket structure and to obtain the correct panel end conditions for the various loading cases. The panel end conditions were extracted from the system model results (for the highest loaded panels) and used as boundary conditions for the edges of the detailed panel model of Figures 4B.2-2 and 4B.2-3. This detailed panel model was then used directly to determine the stresses in the plates and welds since these components are modeled in detail in that model.

4B.3 DELETED

4B.4 DELETED

4B.5 DELETED

4B.6 BASKET ANALYSIS UNDER SUSTAINED LATERAL LOADINGS

It should be noted that the aluminum plates in the TN-40 basket are primarily heat conductors. The 304 stainless steel members are the primary structural components. The calculated stress intensities presented in this section have been corrected for the 13 gauge (0.088 in.) stainless steel boxes (ANSYS model assumes 0.1 in. SS).

As described in Section 4B.1, for long term sustained loading, the 6061 aluminum strength is generally neglected (except for through thickness strength) under primary loading where the aluminum can share the load with the 304 stainless steel. This analysis approach produces conservatively high calculated values of primary stresses in the stainless steel components. Since the aluminum strength is already neglected in this approach, creep, relaxation, yielding, etc. in the aluminum cannot increase the stainless steel primary stresses above these bounding results. The actual aluminum strength is considered, however, when determining the secondary (thermal) stresses that it can apply to the 304 stainless steel members. Thus the aluminum strength is neglected when it might reduce the calculated primary stresses in the stainless components but it is considered completely effective when it can induce secondary stresses in the stainless steel.

The primary stress analysis of the basket under the bounding loads for Design, Level A and Level D (not impact accidents) conditions described in Sections 3.2.5.4.2 and 4.2.3.4.4 is described below.

3 g Side Load Without Credit for Aluminum Strength

The system analysis was performed with a 3 g lateral inertial load at the 0° orientation shown in Figure 4B.6-1. The elastic modulus of the aluminum and boral were assumed to be small (10,000 psi) to simulate very weak materials. The computer results were scanned and the highest forces and bending moments and their corresponding locations are shown in Figure 4B.6-2. See Figure 4B.3-12 for system model panel locations. Based on the results from this figure, panel location 17 (highest force) and panel location 14 (highest temperature) were selected for analysis using the detailed plate panel model (Figures 4B.2-2 and 4B.2-3). These panel analyses were accomplished by extracting the edge displacements from the system model and applying them to the corresponding edges of the plate panel model. The results were postprocessed to printout the stresses at the 20 standard locations on the basket panel shown in Figure 4B.3-13. Figures 4B.6-3 and 4B.6-4 list the plate stress intensities and plug weld shear stresses for the above two panels. These stresses will be evaluated below to verify that the design criteria are met.

Thermal Stress

The thermal analysis of the basket is described in Section 3.3.2.2. That analysis was performed to determine the basket temperatures for the condition with maximum solar heating, maximum decay heat from the cask contents, and 100°F ambient air. The temperatures from that thermal analysis were used directly in the ANSYS structural models to calculate the basket panel stresses due to differential thermal expansion. Stresses occur due to the differences between the coefficients of thermal expansion of the 304 stainless, the aluminum and boral.

When a panel consisting of aluminum (and boral) and stainless plates is heated, the aluminum expands more than the stainless steel. For example, if a 5 in. long strip (the plug to plug centerline spacing across the panel) of 304 stainless is heated from 70°F to 400°F, it expands 0.0115 in. A 5 in. strip of aluminum expands 0.0223 in., or 0.0108 in. more than the stainless. If the plates were perfectly connected, a compressive stress of 10,240 psi would develop in the aluminum. However, since the connection is only local at a 1 in. diameter plug each 8 in., a simple hand analysis (assuming tight plugs) would result in an aluminum bearing stress of 8 x 10,240 or 81,920 psi, far above the aluminum yield stress of 13,300 psi at 400°F. Therefore local plastic deformation of the aluminum plates in areas adjacent to the plugs will occur (if the plugs are not centered in the holes at assembly) and this effect must be properly considered to accurately determine the 304 stainless secondary stress state.

The first task in determining the thermal stresses was to run the system model. The full elastic behavior of the aluminum was considered in that model to conservatively determine panel edge loads, displacements, etc. The results from the system model were scanned. Since panel location 14 is located in the highest temperature (530°F) area of the basket, the results from this panel were used to run the detailed plate model.

The detailed panel model used for the panel thermal stress analysis is shown in Figure 4B.6-5. It is conservatively assumed that the 1 in. diameter stainless steel plugs that penetrate the 1.12 in. diameter holes in the aluminum (and boral) plates are not centered. The plugs are assumed to be in contact initially (at 70°) with the opposing sides of the two holes in the aluminum (the sides toward the center of the panel) so that the maximum interference of aluminum and steel will occur when the panel is heated. It should be noted that this is a condition that cannot reverse. If the temperature decreases after initial heatup, the plug to aluminum contact will be lost but tension cannot be developed at the plug to aluminum interface.

The holes in the aluminum and boral plates are modeled (see square holes in Figure 4B.6-5). Also the bearing interfaces between the aluminum plates and plugs are approximately modeled using rigid members connected between the edges of the holes and nodes on the axis of the pipe elements representing the plugs. Plasticity was considered in the local regions (only) of the aluminum plates at the plug/aluminum bearing interfaces. In these local areas of the aluminum the ANSYS STIF 63 elastic shell elements were replaced with STIF 48 plastic shell elements. The 304 stainless steel structural members are still modeled elastically. The stress vs. strain curve used for the aluminum was based on Table 4B.1-1 from the Aluminum Association data. The boral sheet was assumed to have the same material properties as the aluminum.

Figure 4B.6-6 lists the thermal stresses from this detailed panel analysis. The reviewer should be reminded that these stresses in the 304 stainless steel structural plates are elastic stresses. The stainless steel is modeled as an elastic material even though local plasticity is considered in the aluminum (which applies differential expansion loading to the stainless steel). These stresses are conservatively combined with the stresses from the 3 g side load by adding stress intensities at the stress reporting locations. The combined primary plus secondary (thermal) stresses are listed in Figure 4B.6-7. The stresses are compared with the specified limits below.

Design Criteria

The primary stress analysis of the basket for sustained Design, Level A and Level D Service Conditions does not take credit for the aluminum conductor plates except for through thickness compression. The aluminum strength is, however, considered when determining secondary stresses in the stainless steel.

The basis for the 304 stainless steel fuel compartment box section stress allowables is Section III of the ASME Code. The primary membrane stress intensity and primary membrane plus bending stress intensities are limited to S_m (S_m is the Code allowable stress intensity) and 1.5 S_m, respectively, at any location in the basket for Design and Level A load combinations. The 3 g bounding load cases described above also bound all Level D loads such as seismic and tornado effects. No special Level D criteria or evaluation are needed for these sustained loadings, however, since these Level D loadings (less than the 3 g bounding loads) are evaluated as Design Conditions.

The ASME Code provides a basic 3 S_m limit on primary plus secondary stress intensity for Level A conditions. That limit is specified to prevent ratcheting of a structure under cyclic loading and to provide controlled linear strain cycling in the structure so that a valid fatigue analysis can be performed. The Code also provides guidance in the application of plastic analyses which can be performed to demonstrate shakedown (absence of ratcheting) and to determine stresses for fatigue evaluation. Ratcheting and fatigue cannot occur in the basket since thermal cycling will not occur and interference loading at the plug/aluminum interfaces cannot reverse.

Evaluation

Figure 4B.6-4 lists the stress intensities for the 3 g side load in the basket at panel location 14. Note that these stresses have been calculated elastically (assuming structurally ineffective aluminum). The highest membrane stress intensity is 465 psi at stress reporting location 20. The highest membrane plus bending stress intensity is 3,594 psi at stress location 5. These stresses are well below the allowable membrane stress intensity (S_m) of 17,000 psi and the allowable membrane plus bending stress intensity (1.5 S_m) of 25,700 psi based on the temperature of 530°F at this panel location.

Figure 4B.6-3 lists the stress intensities for the 3 g side load in the basket at panel location 17. The highest membrane stress intensity is 1,378 psi at stress reporting location 1. The highest membrane plus bending stress intensity is 6,056 psi, also at stress location 1. These stresses are well below the allowable 304 stainless steel membrane stress intensity of 19,000 psi and membrane plus bending stress intensity of 29,000 psi based on the temperature of 330°F at this panel location.

Figure 4B.6-6 lists the thermal stresses due to differential thermal expansion in the basket at panel location 14, and Figure 4B.6-7 lists the combined stress intensities for the thermal stress and 3 g side load. The basic primary plus secondary stress limit at any location on the 304 stainless steel panel is 3 S_m or 51,000 psi at the maximum temperature of 530°F. The maximum primary plus secondary stress intensity of 49,036 psi occurs at stress location 4 and is well below the basic allowable stress. The maximum weld stress intensity is 50,902 psi which is also below the basic limit.

Based on the results of these analyses, it is concluded that:

1. The maximum 304 stainless steel (fuel compartment box) stresses, both in the center and peripheral regions of the basket, are well below the specified allowable stresses under the Design (3 g side load) and Level A (3 g load plus thermal stress) conditions.

2. The maximum shear stresses in the plug welds are low under the 3 g loading above. The stainless and aluminum plates may push against the plugs due to differential thermal expansion if the plugs are not centered in the holes in the aluminum at assembly. In the worst plug misalignment case the weld shear stress could reach a maximum of 25,451 psi. The corresponding stress intensity is 2 x τ or 50,902 psi. This stress intensity is below the basic 3 S_m limit of 51,000 psi. This basic limit ensures that thermal ratcheting of a structure does not occur. This primary plus secondary limit could be exceeded in the TN-40 basket since the stress does not cycle and since the loading cannot reverse.

3. The aluminum plates are generally not considered to have a structural function under Design or Level A conditions. Nevertheless, the primary plus secondary (thermal) stress in the aluminum plates midway between the stainless plugs (stress reporting locations 10 and 12) is no greater than 1,617 psi since this upper bound stress was calculated assuming the stainless plugs were misaligned in the aluminum holes in the worst possible way. This stress is much less than the 10,240 psi discussed above that might develop in a perfectly connected aluminum plate. The 1,617 psi stress is also far below the 8,535 psi allowable compressive stress (based on stability or buckling) in the aluminum from Figure 4B.5-5. Therefore compressive stresses developed in the aluminum cannot cause these plates to buckle.

4. The basket is structurally adequate and it will properly support and position the fuel assemblies.

4B.7 BASKET ANALYSIS UNDER VERTICAL LOADINGS

Under vertical loads, the fuel assemblies and basket are forced against the bottom of the cask. It is important to note that, for any vertical or near vertical loading, the fuel assemblies react directly against the bottom of the cask cavity and not through the basket structure as in lateral loading. The fuel assemblies weigh 52,000 lb and the basket weighs 15,841 lb. Therefore the basket weight is only about 25% of the total cask internals weight (of fuel and basket). Therefore the vertical basket inertial loading is only about 25% of the lateral loading for a given g level.

3 g Vertical Load Without Credit for Aluminum Strength

The analysis of the basket subjected to the 3 g bounding vertical load (bounds all Design, Level A and Level D sustained loads) is presented in Figure 4B.7-1. A full length of compartment wall (160 in. long) with a span length of 8.05 in. is evaluated for compressive load. A maximum compressive force of 431 lb occurs at the bottom of the wall. Stresses are conservatively calculated by assuming all of the load is taken by the 304 stainless steel. Therefore

$$\sigma = \frac{Total\ Compressive\ Load}{Cross\ Section\ Area\ of\ 304SS} = \frac{431\,lb.}{1.42\ in.^2} = 304\ (psi)$$

Based on the above results it is concluded that the stress on the stainless steel panel due to the 3 g vertical load is insignificant and additional analysis is not necessary.

Vertical Load Due to Hypothetical End Drop Accident

Section 8.2.8.2.1 presents the dynamic impact analysis of the TN-40 cask during a hypothetical end drop accident. That section of the SAR recommends that the cask be analyzed for 50 g vertical loading to conservatively evaluate this accident.

Figure 4B.7-1 presents the analysis of the basket due to a 3 g vertical load neglecting the strength contribution from the aluminum. Using the same weight and dimensions, the compressive stress on the 304 stainless steel plates for a 50 g load is 50/3 x 304 or 5,067 psi. This stress level is acceptable for any criteria (Design, Level D or Level D impact). In addition, test data presented in Appendix 4C show that the compressive stress can reach 23,000 psi before buckling failure even without the stiffening effects of the aluminum plates. Therefore the 5,067 psi stress is acceptable based on both stress and buckling criteria.

4B.8 REFERENCES

1. De Salvo, G.J. and Swanson, J.A., ANSYS Engineering Analysis System Users Manual for ANSYS Revision 4.3, Swanson Analysis Systems, Inc., Houston, PA, June 1987.

2. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III Appendices, 1989.

3. Aluminum Standards and Data, Volume 1, The Aluminum Association, 1976.

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TABLE 4B.1-1

MECHANICAL PROPERTIES OF BASKET MATERIALS (NOTE 1)

MATERIAL SPECIFICATION (NOMINAL COMPOSITION)	**APPLICATION**	**MINIMUM YIELD STRENGTH S_y, psi**	**MINIMUM ULTIMATE STRENGTH S_u, psi**	**DESIGN STRESS VALUE,** (NOTE 2)	**DATA SOURCE** (NOTE 3)
ASME SA-240, Type 304 (Stainless Steel)	Basket	30,000	75,000	S_m=20,000	Table I-1.2 p. 25
ASME SB-209, Alloy 6061-T6 (Aluminum)	Basket	35,000	42,000	S=10,500	Table I-8.4 p. 189

NOTES

1. Mechanical properties listed are for metal temperatures up to 100^o to provide a baseline comparison of all structural materials.

   Temperature dependent properties required for structural analysis are provided in Table 4A.2-2.

2. Values listed are the stress parameters which form the basis for structural analysis acceptance criteria.

   S refers to the ASME allowable stress for Class 2 or Class 3 components,
   S_m refers to the ASME design stress intensity for Class 1 components, and S_y refers to minimum yield strength.

3. Data are taken from tables in ASME Section III, Appendix I, 1989 unless otherwise noted

TABLE 4B.1-2

SHEET 1 OF 3

TEMPERATURE DEPENDENT MATERIAL PROPERTIES
COEFFICIENTS OF THERMAL EXPANSION

MATERIAL	TEMPERATURE, °F										
	100	150	200	250	300	350	400	450	500	550	600
SA-240, Type 304	8.55	8.67	8.79	8.90	9.00	9.10	9.19	9.28	9.37	9.45	9.53
SB-209, Aluminum 6061-T6	12.60	12.76	12.91	13.07	13.22	13.37	13.52				

Values listed are the mean coefficients of thermal expansion $\times 10^{-6}$(in./in.°F) from 70°F to the indicated temperature. Source of data is ASME Section III, Appendix I, Table I-5.0, pp. 123-128.

TABLE 4B.1-2

SHEET 2 OF 3

TEMPERATURE DEPENDENT MATERIAL PROPERTIES
MODULI OF ELASTICITY, E

MATERIAL	TEMPERATURE °F					
	70	**200**	**300**	**400**	**500**	**600**
SA-240, Type 304 Stainless Steel	28.3	27.6	27.0	26.5	25.8	25.3
SB-209, 6061-T6 Aluminum	10.0	9.6	9.2	8.7	8.1	
Boral (Based on Brooks Perkins Data Sheet)	5.8					

Values listed are the moduli of elasticity $\times 10^6$ psi for the indicated temperature.

Source of data is ASME Section III, Appendix I, Table I-6.0, pp. 129 –130.

TABLE 4B.1-2

SHEET 3 OF 3

TEMPERATURE DEPENDENT MATERIAL PROPERTIES
DESIGN STRESS PARAMETER

MATERIAL	STRESS PARAMETER (NOTE 1)	TEMPERATURE °F 100	200	300	400	500	600	DATA SOURCE
ASME SA-240,	S_y	30.0	25.0	22.5	20.7	19.4	18.2	Table I-2.2, p. 67 (NOTE 2)
Type 304	S_m	20.0	20.0	20.0	18.7	17.5	16.4	Table I-2.2, p. 25
	psi							
ASME SB-209	S_y	35.0	33.2	27.4	13.3	4.375	---	NOTE 3
Alloy 6061-T6	S_u	42.0	36.7	31.7	12.7	7.0	---	
	ksi							

<u>NOTES</u>

1. Values listed are the stress parameters which form the basis for structural analysis acceptance criteria.
   S_m refers to the ASME design stress intensity for Class 1 components, and
   S_y refers to minimum yield strength.
   S_u refers to minimum ultimate strength.

2. Data are taken from ASME Section III, Appendix I as noted.

3. 87.5% of Reference 3 data as recommended by ASME Subgroup on Non Ferrous Alloys.

TABLE 4B.1-3

MATERIALS PROPERTIES USED FOR TN-40 BASKET FINITE ELEMENT MODEL

MATERIAL	TEMPERATURE (°F)	ULTIMATE STRENGTH (psi)	YIELD STRENGTH (psi)	MODULUS OF ELASTICITY (psi × 10^6)	THERMAL EXPANSION (in./in./°F)
304 S.S.	70	75,000	30,000	28.3	----
	300	66,000	22,500	27.0	9.0×10^{-6}
	400	64.400	20,700	26.5	9.19×19^{-6}
	500	63.500	19,400	25.8	9.37×37^{-6}
*ALUMINUM 6061-T6	70	42,000	35,000	10.0	----
	300	31,700	27,400	9.2	13.22×10^{-6}
	400	17,700	13,300	8.7	13.52×10^{-6}
	500	7,000	4,375	8.1	13.82×10^{-6}

* Boral is assumed to have the same properties as 6061-T6 aluminum except for modulus of elasticity listed in Table 4B.1-2.

TABLE 4B.1-4

REFERENCE TEMPERATURE FOR
STRESS ANALYSIS ACCEPTANCE CRITERIA (TN-40 BASKET)

LOCATION	MAX. CALCULATED TEMPERATURE, °F	SELECTED DESIGN TEMPERATURE, °F
Centerline at Mid Height (max)	530	530
Centerline at Top	392	400
Periphery at Top	305	330

Figure Withheld Under 10 CFR 2.390

Figure 4B.2-1

REPRESENTATIVE BASKET WALL PANEL

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REPRESENTATIVE BASKET WALL PANEL			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 8	FIG4B.2-1_REV_8
	PAGE. NO.	DATE: 04-20-06	

Figure Withheld Under 10 CFR 2.390

Figure 4B.2-2

DETAILED WALL PANEL SUBSTRUCTURE MODEL

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DETAILED WALL PANEL SUBSTRUCTURE MODEL

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 1	FIG4B.2-2_REV_1
	PAGE. NO.	DATE: 04-20-06	

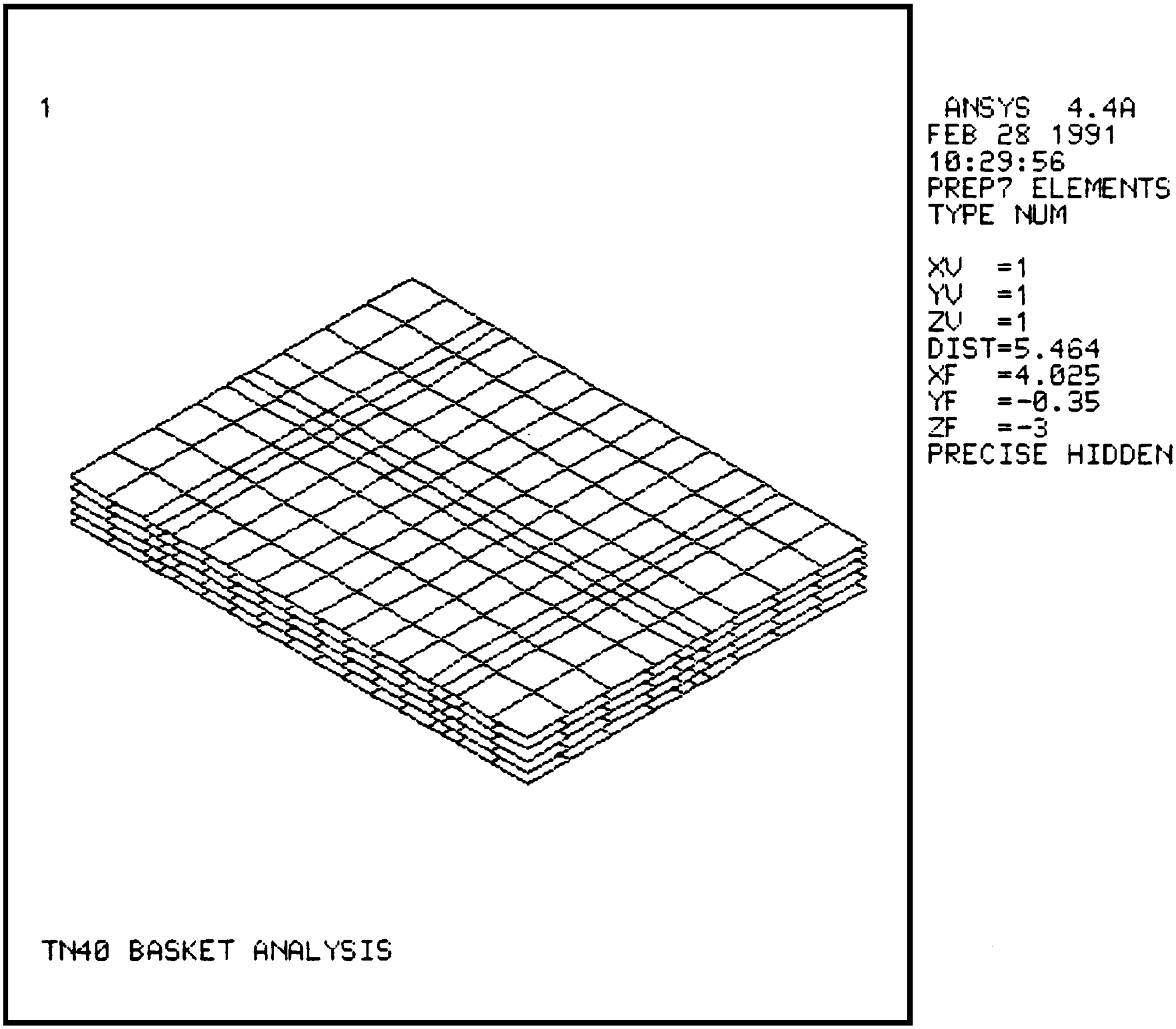

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Figure 4B.2-3

ANSYS COMPUTER PLOT OF DETAILED WALL PANEL SUBSTRUCTURE MODEL

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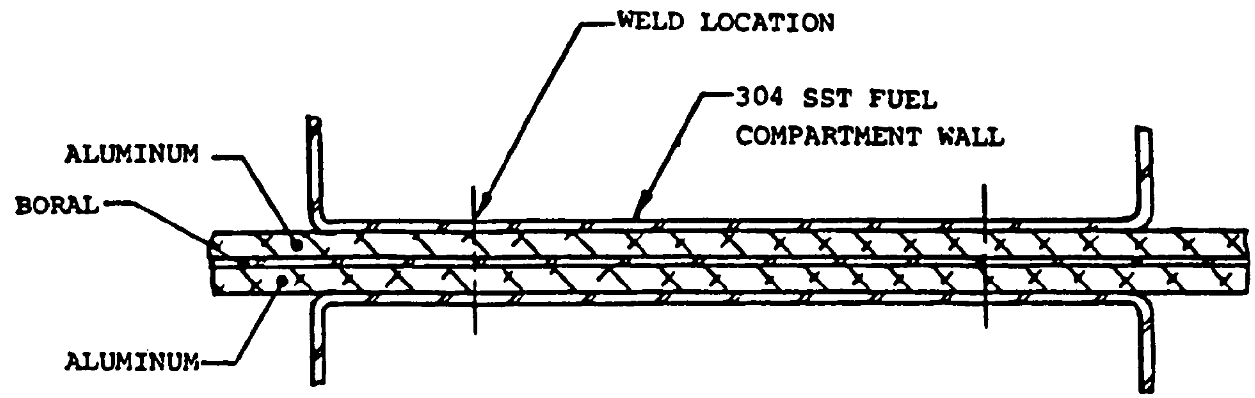
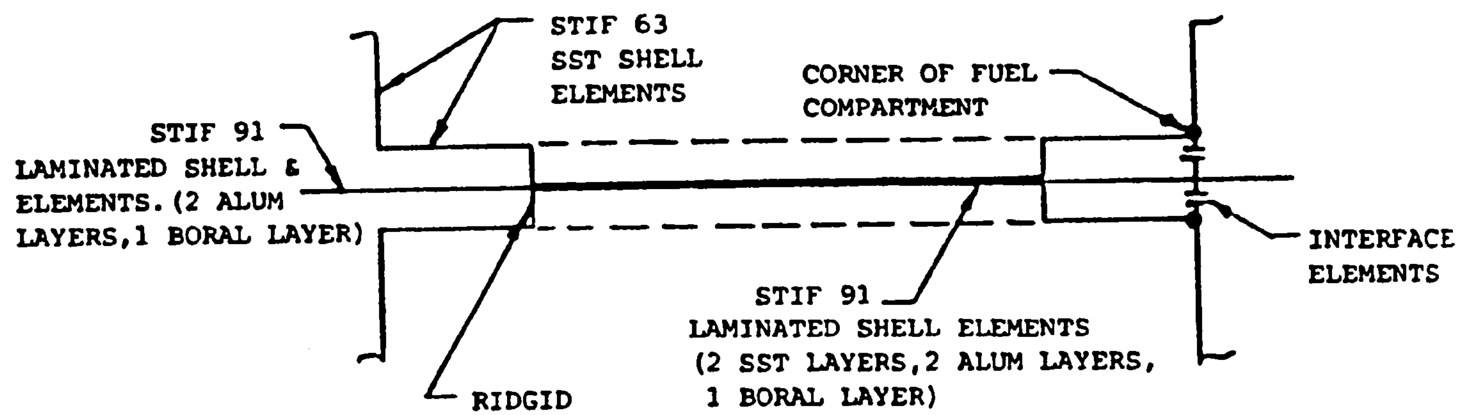
ANSYS COMPUTER PLOT OF DETAILED WALL PANEL SUBSTRUCTURE MODEL

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 1	FIG4B.2-3_REV_1
	PAGE. NO.	DATE: 04-20-06	



a. ACTUAL WALL CONFIGURATION



b. PANEL SIMULATION

REVISION: 1

Figure 4B.2-4

DEVELOPMENT OF SIMPLIFIED PANEL SIMULATION FOR SYSTEM MODEL

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DEVELOPMENT OF SIMPLIFIED PANEL SIMULATION FOR SYSTEM MODEL

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 1	FIG4B.2-4_REV_1
	PAGE. NO.	DATE: 04-20-06	

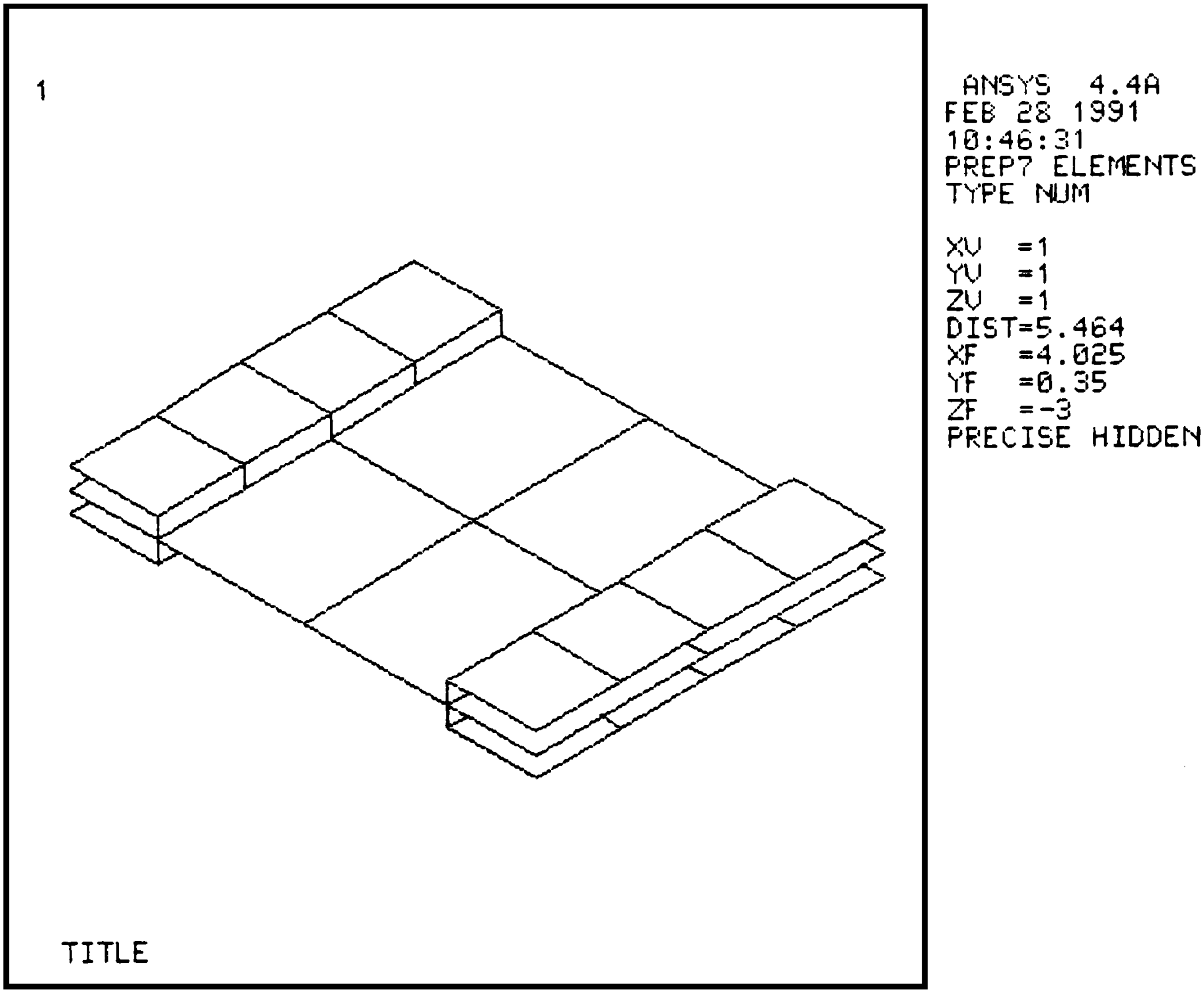

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Figure 4B.2-5

ANSYS COMPUTER PLOT OF SIMPLIFIED WALL PANEL SUBSTRUCTURE MODEL

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ANSYS COMPUTER PLOT OF SIMPLIFIED WALL PANEL SUBSTRUCTURE MODEL

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 1	FIG4B.2-5_REV_1
	PAGE. NO.	DATE: 04-20-06	

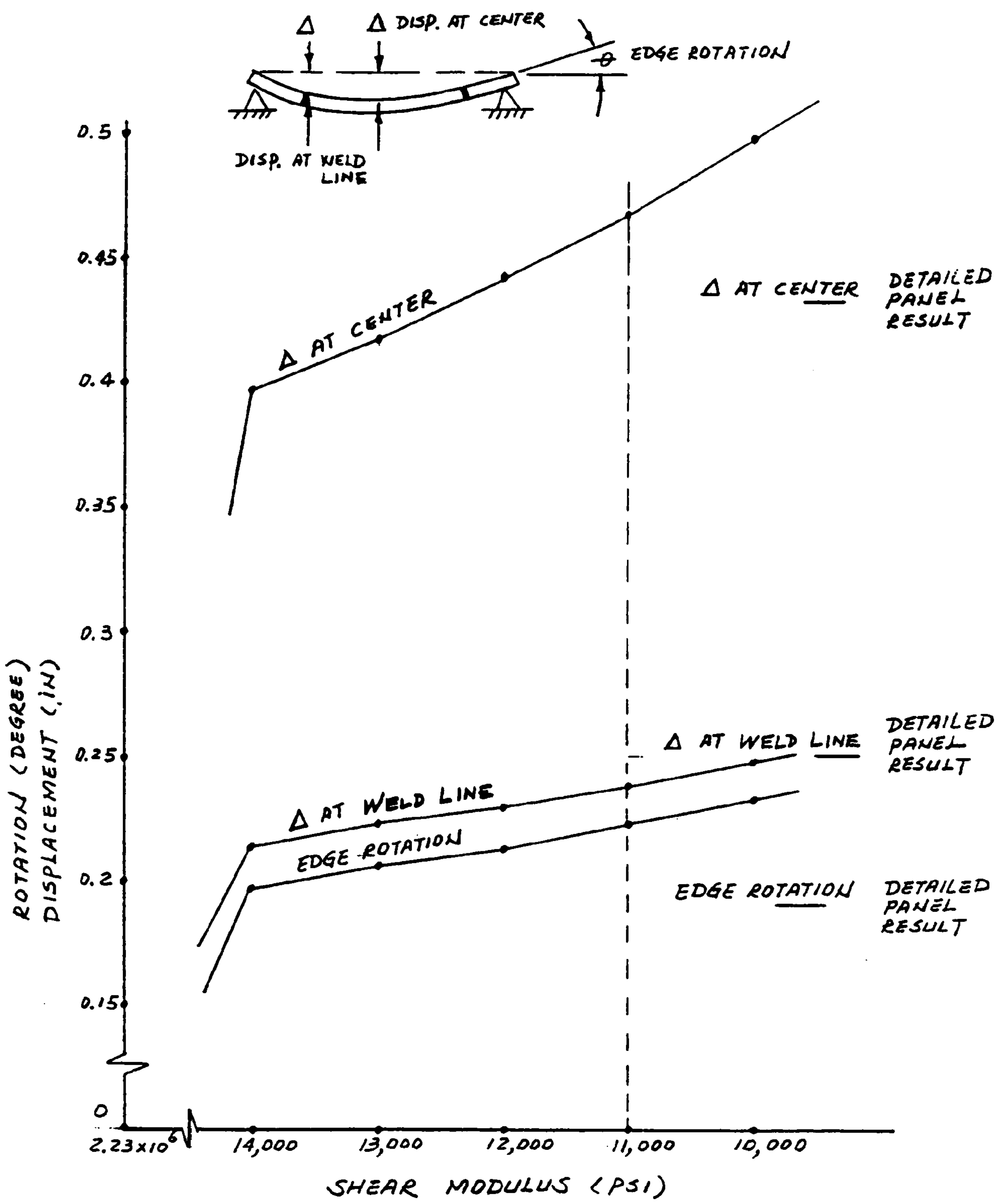

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Figure 4B.2-6

SIMPLY SUPPORTED EDGES-DISP. & ROTATION COMPARISON

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SIMPLY SUPPORTED EDGES - DISP. & ROTATION COMPARISON

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 1	FIG4B.2-6_REV_1
	PAGE. NO.	DATE: 04-20-06	

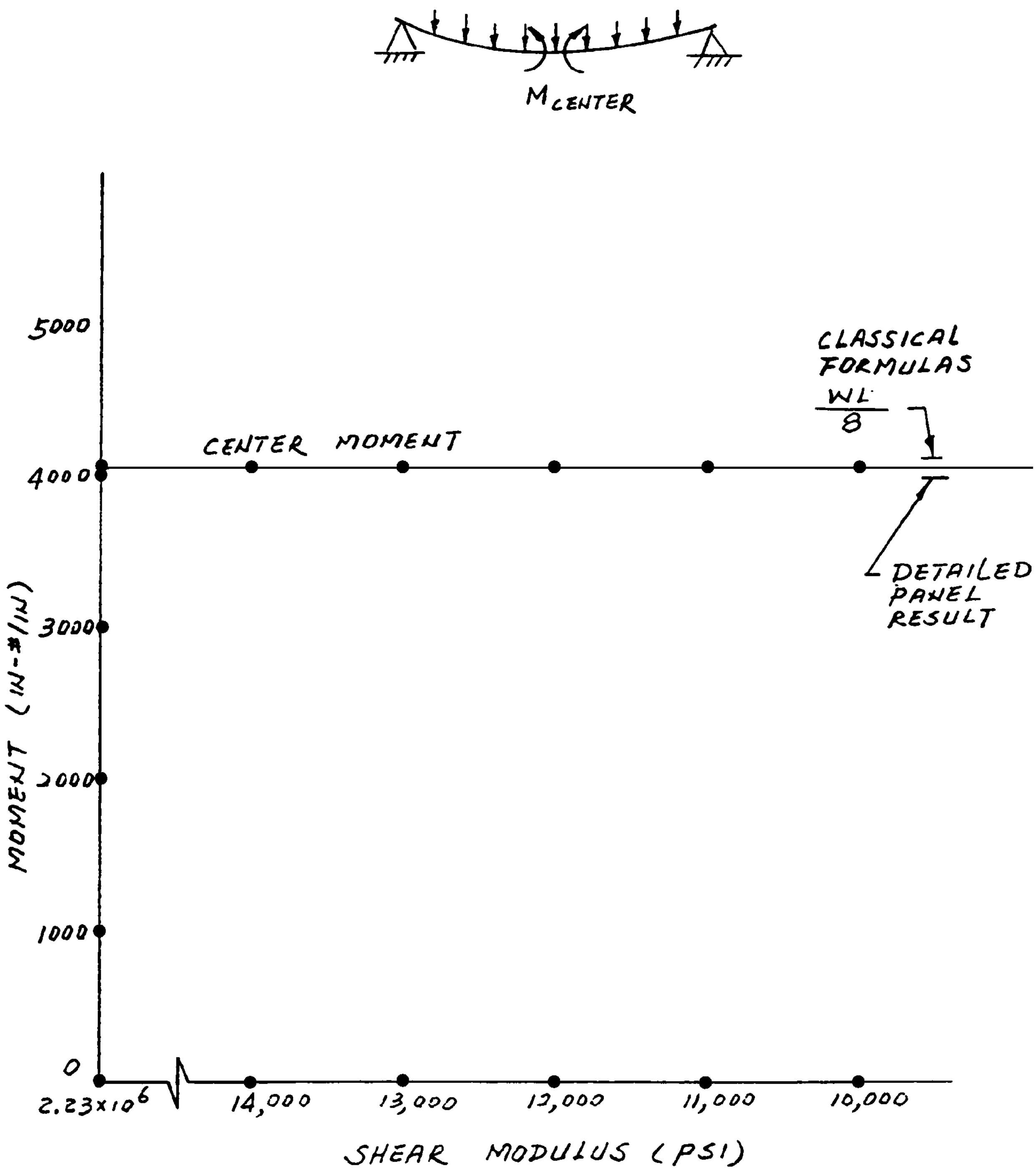

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Figure 4B.2-7

SIMPLY SUPPORTED EDGES-CENTER MOMENT COMPARISON

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SIMPLY SUPPORTED EDGES - CENTER MOMENT COMPARISON

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 1	FIG4B.2-7_REV_1
	PAGE. NO.	DATE: 04-20-06	

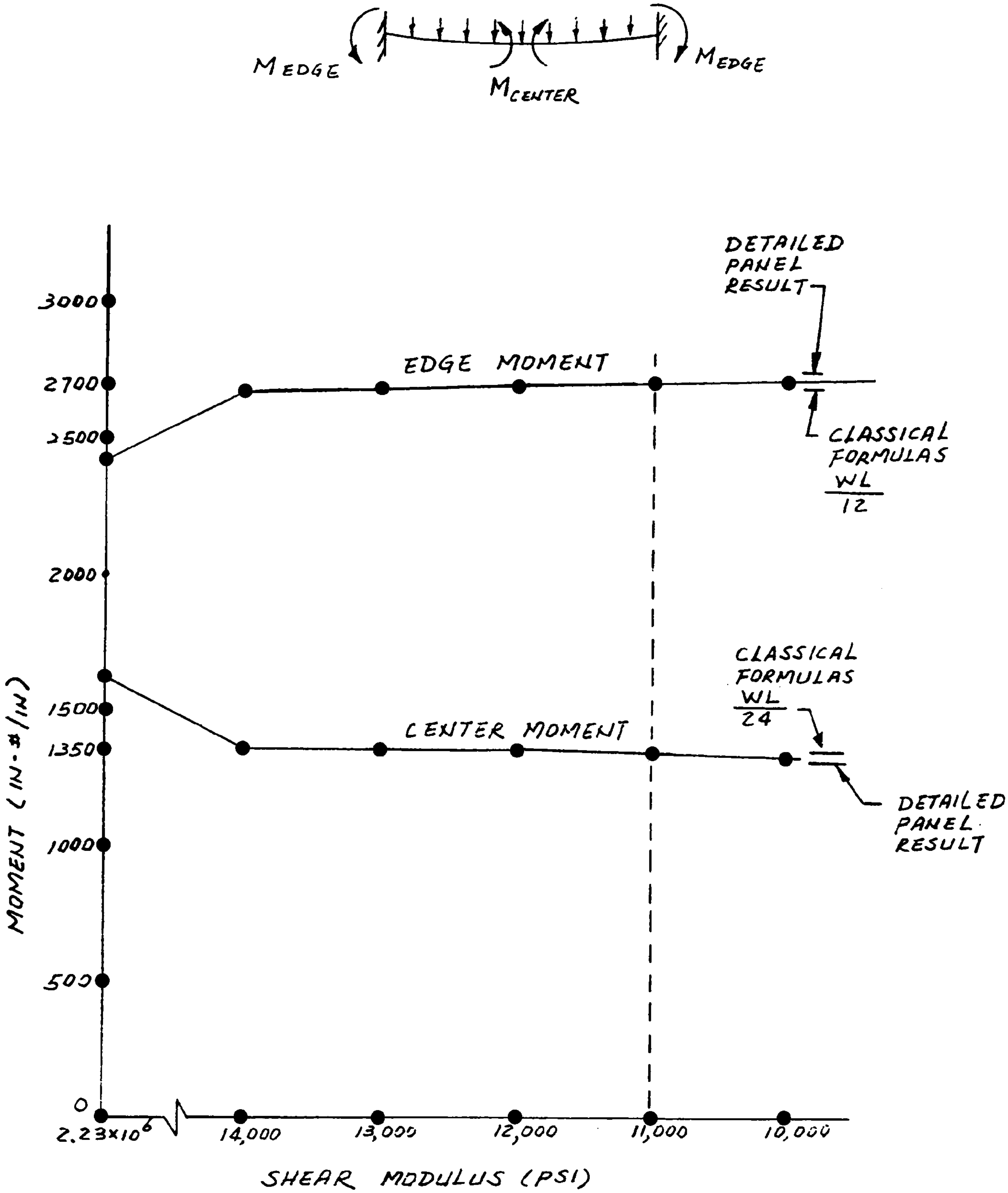

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Figure 4B.2-8

FIXED EDGES-MOMENT COMPARISON

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FIXTED EDGES - MOMENT COMPARISON

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 1	FIG4B.2-8_REV_1
	PAGE. NO.	DATE: 04-20-06	

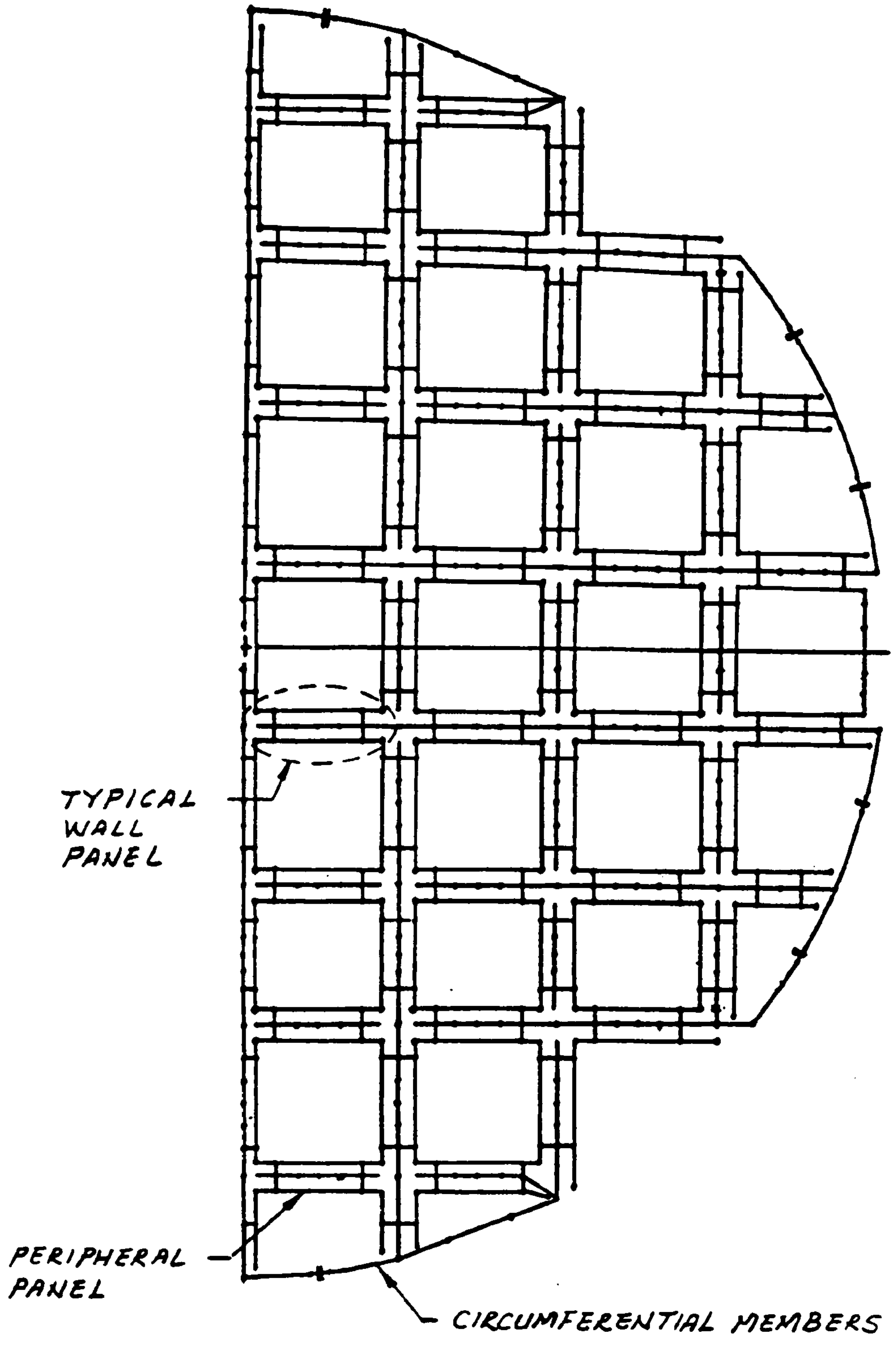

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Figure 4B.2-9

TN-40 BASKET SYSTEM MODEL

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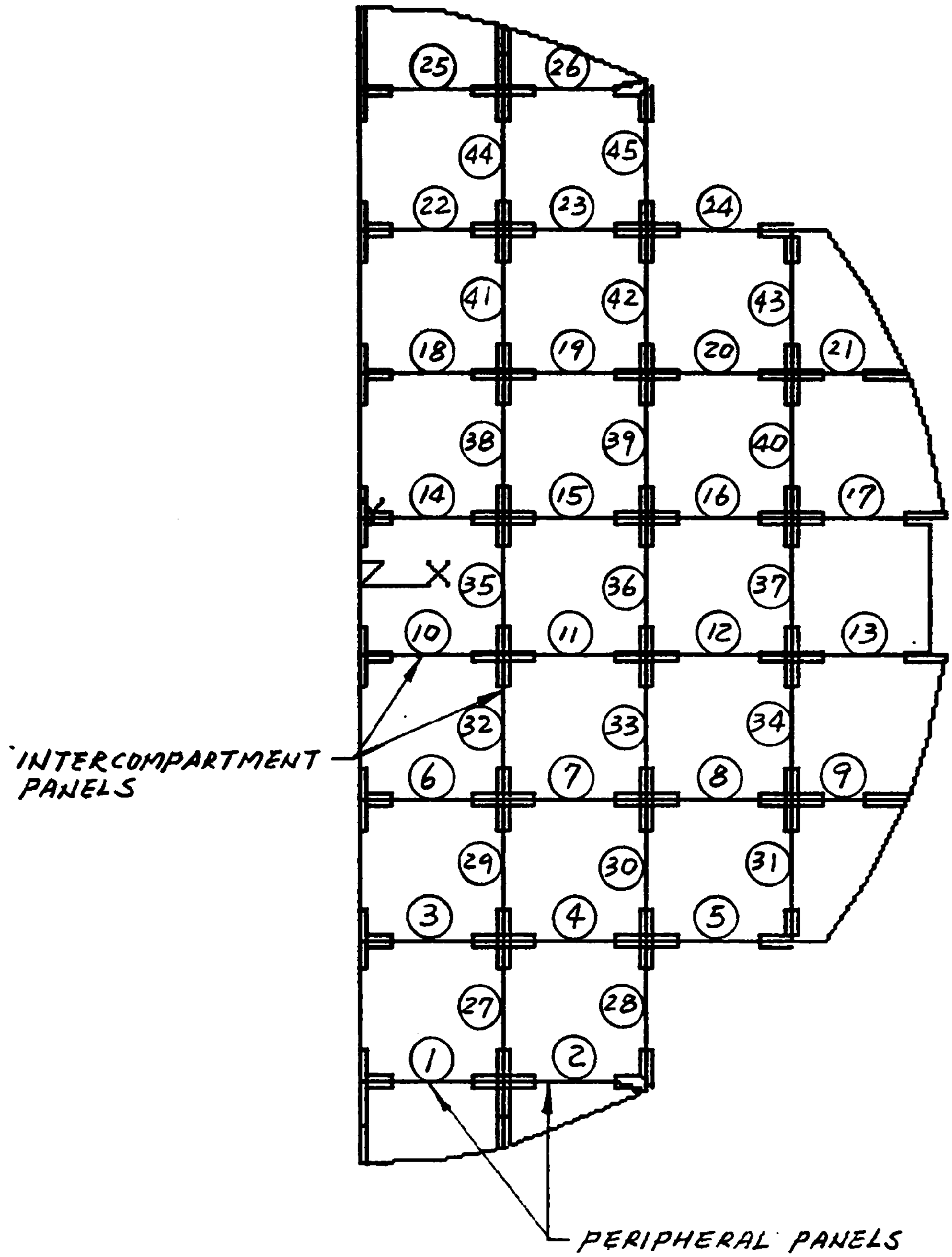

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Figure 4B.3-12

SYSTEM MODEL PANEL LOCATIONS

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SYSTEM MODEL PANEL LOCATIONS

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 1	FIG4B.3-12_REV_1
	PAGE. NO. NA	DATE: 04-21-06	

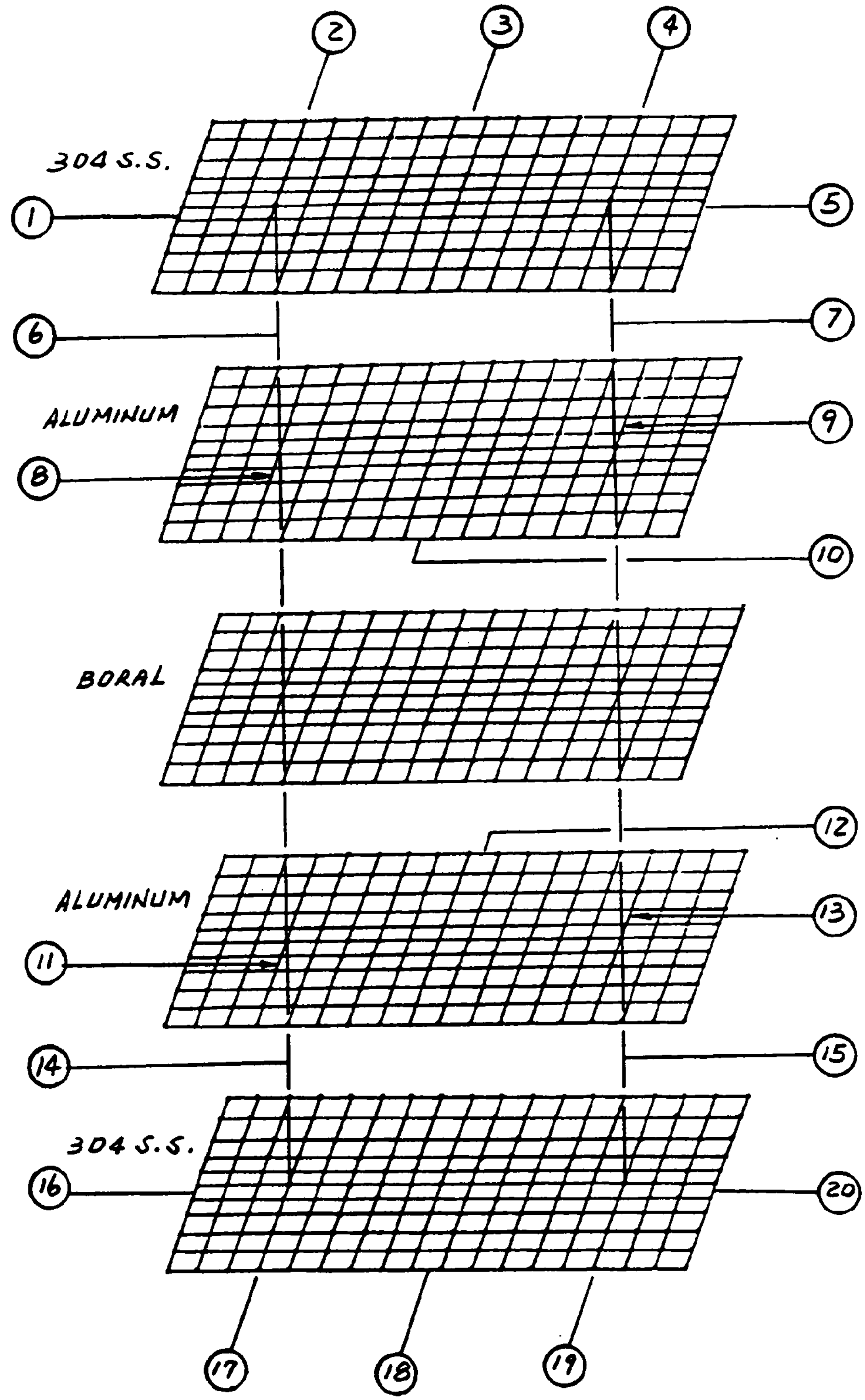

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Figure 4B.3-13

DETAILED WALL PANEL SUBSTRUCTURE STRESS REPORTING LOCATIONS FOR INTERCOMPARTMENT BASKET WALL PANEL

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DETAILED WALL PANEL SUBSTRUCTURE STRESS REPORTING LOCATIONS FOR INTERCOMPARTMENT BASKET WALL PANEL

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 1	FIG4B.3-13_REV_1
	PAGE. NO. NA	DATE: 04-21-06	

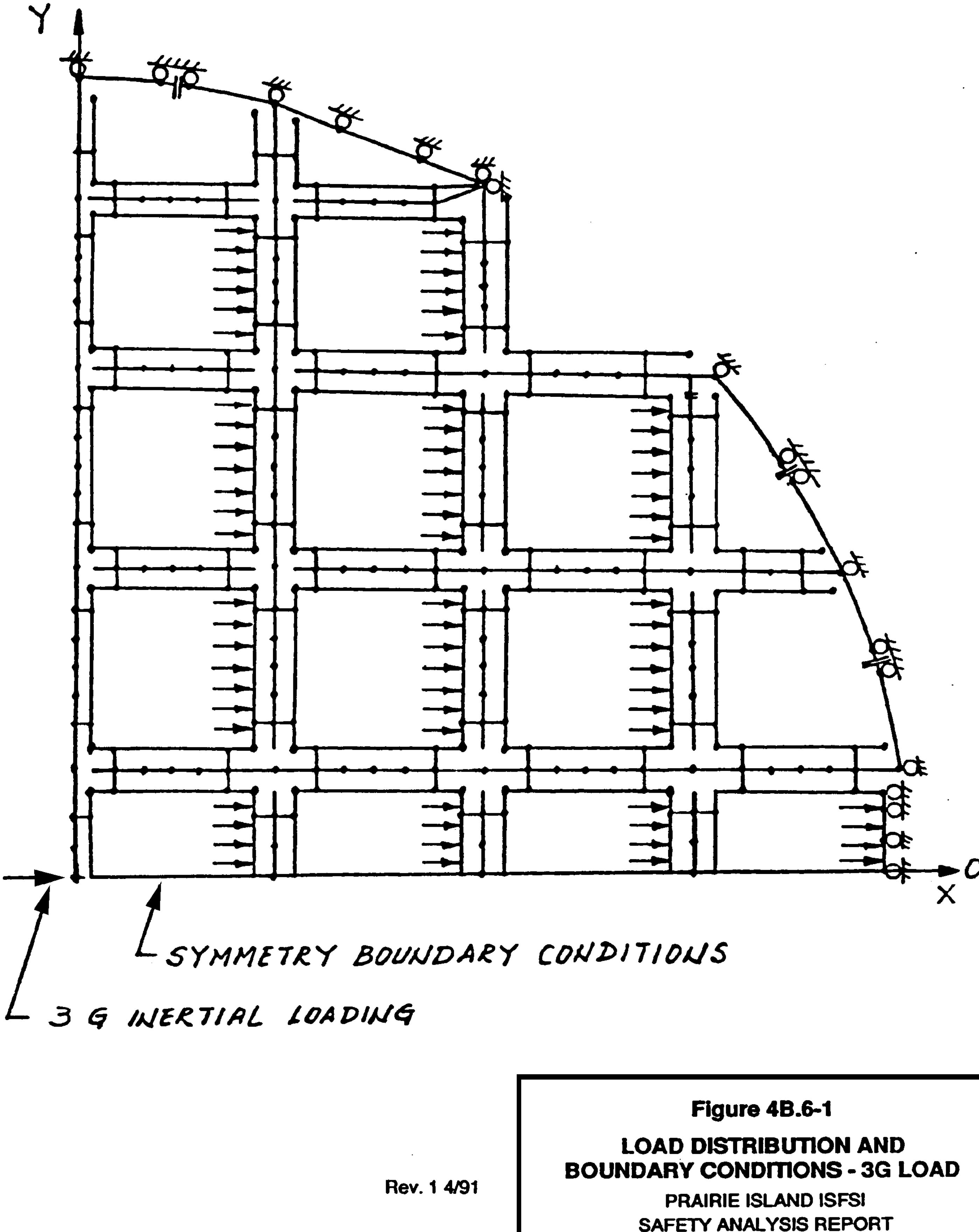

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Figure 4B.6-1

LOAD DISTRIBUTION AND BOUNDARY CONDITIONS - 3G LOAD

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PANEL LOCATION (FIG. 4A.5-5)		Fx (#/IN)	FY (#/IN)	Mz (IN-#/IN)			
HORIZONTAL PANEL	14	70		≈0			
	16	135		≈0			
	17	173		≈0			
	20	20		6			
	24	79		15			
VERTICAL PANEL	35		4	16			
	36		1	15			
	37		86	8			
	39		3	19			
	40		87	15			
	42		3	17			
	43		72	22			

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Figure 4B.6-2

FORCES AND MOMENTS DUE TO 3G SIDE LOAD (TN-40 CASK)

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FORCES AND MOMENTS DUE TO 3G SIDE LOAD (TN-40 CASK)

NORTHERN STATES POWER COMPANY
Xcel Energy
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA

DRAWN BY: VLS	REVISION: 1	FIG4B.6-2_REV_1
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0 Degree Drop (3G) – Location 17

Location (Fig 4B.3-13)	STRESS INTENSITIES (PSI)			Shear Stress at Weld
	Average	Top Surface	Bottom Surface	
1	1571	6056	2694	
2	1365	2096	1760	
3	1451	1736	1634	
4	1360	1382	1722	
5	1470	1318	2010	
6				27
7				85
8				
9				
10				
11				
12				
13				
14				20
15				86
16	1324	2113	4578	
17	1296	2112	1609	
18	1451	1792	1544	
19	1322	1446	1565	
20	1477	1742	1602	

Note: stress intensities shown are for 13 gauge (0.088") stainless steel plate.

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Figure 4B.6-3

BASKET PANEL STRESSES UNDER 3G SIDE LOAD (TN-40 CASK) (PANEL LOCATION 17)

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BASKET PANEL STRESSES UNDER 3G SIDE LOAD (TN-40 CASK) (PANEL LOCATION 17)

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 8	FIG4B.6-3_REV_8
	PAGE. NO. NA	DATE: 04-21-06	

0 Degree Drop (3G) – Location 14

Location (Fig 4B.3-13)	STRESS INTENSITIES (PSI) Average	Top Surface	Bottom Surface	Shear Stress at Weld
1	424	2254	1316	
2	392	428	1171	
3	426	466	502	
4	393	1373	553	
5	448	2599	3594	
6				17
7				70
8				
9				
10				
11				
12				
13				
14				44
15				6
16	449	1020	2012	
17	421	1331	495	
18	458	501	538	
19	424	855	1680	
20	465	2910	1880	

Note: stress intensities shown are for 13 gauge (0.088") stainless steel plate.

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Figure 4B.6-4

BASKET PANEL STRESSES UNDER 3G SIDE LOAD (TN-40 CASK) (PANEL LOCATION 14)

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Figure 4B.6-5

DETAILED PANEL MODEL (THERMAL RUN)

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DETAILED PANEL MODEL (THERMAL RUN)

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 1	FIG4B.6-5_REV_1
	PAGE. NO. NA	DATE: 04-24-06	

Thermal Stress Run – Location 14

Location	STRESS INTENSITIES (PSI) THERMAL (SECONDARY) STRESS - Q			Shear Stress at Weld
	Average	Top Surface	Bottom Surface	
1	14267	32609	8149	
2	17579	26495	35272	
3	19469	21930	22145	
4	17573	29375	48483	
5	9193	30864	7277	
6				25434
7				25378
8	2815	3185	3185	
9	2815	3185	3185	
10	1844	2086	2086	
11	2817	3188	3188	
12	1844	2086	2086	
13	2817	3188	3188	
14				25364
15				25421
16	14242	7891	31786	
17	19027	29830	19640	
18	19432	22118	21868	
19	20211	30932	28741	
20	14231	7327	31308	

Note: stress intensities shown are 13 gauge (0.088”) stainless steel plate.

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Figure 4B.6-6

BASKET PANEL STRESSES DUE TO DIFFERENTIAL THERMAL EXPANSION (TN-40 CASK) (PANEL LOCATION 14)

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Location	STRESS INTENSITIES (PSI) PRIMARY + SECONDARY STRESS (Pm + Pb + Q)			Shear Stress at Weld
	Average	Top Surface	Bottom Surface	
1	13331	34863	9465	
2	17971	26923	36443	
3	19895	22396	22647	
4	17966	30748	49036	
5	9641	33463	10871	
6				25451
7				25448
8	2815	3185	3185	
9	2815	3185	3185	
10	1844	2086	2086	
11	2817	3188	3188	
12	1844	2086	2086	
13	2817	3188	3188	
14				25408
15				25427
16	14691	8911	33798	
17	19448	31161	20135	
18	19890	22619	22406	
19	20635	31787	30421	
20	14696	10237	33188	

Note: stress intensities shown are for 13 gauge (0.088") stainless steel plate.

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Figure 4B.6-7

BASKET PANEL STRESSES - THERMAL + 3G SIDE LOAD (TN-40 CASK) (PANEL LOCATION 14)

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Figure 4B.7-1

BASKET STRESSES DUE TO
3G VERTICAL LOAD
(TN-40 CASK) (PANEL LOCATION 14)

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BASKET STRESSES DUE TO 3G VERTICAL LOAD (TN-40 CASK) (PANEL LOCATION 14)			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 8	FIG4B.7-1_REV_8
	PAGE. NO.	DATE: 04-24-06	

APPENDIX 4C

TESTS PERFORMED TO SUPPORT DESIGN OF THE TN-40 BASKET

4C.1 COMPRESSION TESTS OF THE TN-40 BASKET PANELS

The structural design criteria for the basket under impact accident loading are presented in Section 4B.5 of Appendix 4B, the basket analysis appendix. That section presents both stress and basket wall panel stability criteria. Determination of the panel compressive load limit (analytically) required the analyst to make several key assumptions. The tests described in this section were performed to verify the validity of these assumptions and to demonstrate the safety margin associated with the Section 4B.5 compressive load limit.

Figure 4C.1-1 shows a representative TN-40 basket wall section between fuel compartments. The compression tests were conducted on test specimens or panels designed to simulate this wall section. These test panels were fabricated from the TN-40 basket materials, 6061-T6 aluminum and 304 stainless steel. The panels were made without the 0.075 in. center boral plate. This is conservative since the boral plate slightly increases the compressive load capability of the basket wall section. The various plates in the panels were attached to each other using prototypic stainless steel plugs and fusion welds as shown in the figure (two welds 5 in. apart at various spacings along the basket axial direction).

The compression tests were performed to determine the panel strength when loaded in the 8.05 in. direction (to simulate loading of a bottom outer radial panel of a basket during side impact). Each test panel was 8.05 in. long and 24 in. wide (the basket axial direction). Panels were tested with a weld spacing of 6 in., 8 in. and 12 in. in the 24 in. direction. Panels were also tested at room temperature and at elevated temperature.

The test setup for the panel compression test is shown schematically in Figure 4C.1-2. The loaded edges of the panel were hinged so that no panel edge rotational restraint would be provided. This results in conservatively low panel failure loads since, in the actual basket arrangement, there is some edge rotational restraint at the corner of the stainless steel box sections and at the aluminum plate continuation into adjacent panels. The edges were loaded through a lubricated brass insert free to rotate in the test fixture as shown schematically in Figure 4C.1-2.

Tests were run initially with 3 panels at room temperature. Then three additional elevated temperature tests were run. Thermocouples and strip heaters were mounted on both sides of the panels for the hot test. Soft insulation was wrapped around the panel (and heaters, etc.). The heaters were turned on and the temperature was allowed to stabilize while the panel and fixture were mounted in the test machine. See the top sketch in Figure 4C.1-3.

The tests were performed at a rapid loading rate; the test machine was operated at a head travel speed of 0.2 in./minute. The observed deformation modes are illustrated in the lower three sketches of Figure 4C.1-3.

Initially, the load increased (well beyond 100,000 lbs) with the panel apparently undeformed as in the first sketch. The 0.10 inch steel plates began to separate from each side of the aluminum plates (buckling elastically between welds but not failing) as shown in the second sketch. In the first room temperature test, a load of 290,000 lbs was applied and then released prior to reaching the panel failure load. The steel plates, which had separated, "snapped back" nearly flush with the aluminum plates. Therefore this initial steel separation is almost completely elastic. Also, this demonstrated that there is negligible permanent deformation of the panel at a load approaching 85% of the failure load.

This separation gradually increased as the load increased until both aluminum plates buckled in the center as shown in the third sketch. The load peaked at this time and gradually decreased after the aluminum buckled. There was no sudden load dropoff. Figure 4C.1-4 illustrates a typical load vs. deflection curve. It shows the elevated temperature curve for 8 in. weld spacing.

The panel compression test results are summarized in Figure 4C.1-5. The panel buckling load (maximum load from load vs deflection curve) in the room temperature tests ranged from 13,333 lb/in to 14,166 lb/in. The buckling load at metal temperatures from 365°F to 529°F ranged from 10,542 lb/in to 11,135 lb/in. These loads are very consistent (within 6%) for a given metal temperature. In addition, the deformation mode was uniform from test to test. The buckling load was quite insensitive to weld spacing along the axial direction of the basket. Also, the stainless plates and aluminum plates behaved as though they were fixed along the weld line, as assumed in the criteria section 4B.5. See the lower sketches in Figure 4C.1-3.

The test panels were made from 304 stainless and 6061-T6 aluminum plates procured in accordance with ASME/ASTM material standards. They were slightly thicker than specified for the TN-40 basket. Therefore, it was necessary to determine an allowable compressive load for the thicknesses using the technique described in Section 4B.5. If we repeat the calculations performed in Figures 4B.5-5 and 4B.5-6 for a stainless plate 0.110 in. thick and an aluminum plate 0.256 in. thick (actual thicknesses of plates in test panels), we obtain the results listed in Figure 4C.1-6. The allowable compressive load for this geometry using the TN-40 criteria is 4951 lb/in (vs 4512 lb/in for TN-40). If we compare the minimum measured elevated temperature panel buckling load from Figure 4C.1-5 (10,542 lb/in) with the allowable (4951 lb/in) we can see that the panel buckling load was at least 2.13 times the allowable. This demonstrates that the TN-40 basket panel compressive load criteria has an inherent safety factor greater than 2.

4C.2 AXIAL CRUSH TEST OF STAINLESS STEEL BOX SECTION

Section 4B.7 provides the analysis of the basket under vertical loading due to a hypothetical end drop accident. Under vertical loading the basket inertial loading is reacted by a uniform contact force developed between the basket and the bottom of the cask cavity. The compressive stress developed at the ends of the 304 stainless steel plates is 4750 psi, neglecting any strength contribution from the aluminum plates.

The load vs deflection curve from an axial crush test (performed previously for a different program) on a length of bare, unstiffened 304 stainless steel box section is shown in Figure 4C.2-1. That box section withstood an axial force of 102,000 lbs. before it began to crush. The load peaked at this value and then slowly decreased. Sudden failure did not occur.

The 102,000 lbs corresponds to an axial stress of 23,000 psi in the test specimen (a 9×9×0.122 in box section). This test demonstrated that a bare 304 stainless steel box section can withstand almost five times the stress applied in the end drop accident. This was a severe test since the aluminum plate would react a portion of the load and adjacent box sections would tend to provide additional support.

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Figure 4C.1-1

REPRESENTATIVE BASKET WALL PANEL

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REPRESENTATIVE BASKET WALL PANEL			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 1	FIG4C.1-1_REV_1
	PAGE. NO. NA	DATE: 04-24-06	

Figure Withheld Under 10 CFR 2.390

Figure 4C.1-2

TN-40 BASKET PANEL COMPRESSION TEST SETUP

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TN-40 BASKET PANEL COMPRESSION TEST SETUP			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 1	FIG4C.1-2_REV_1
	PAGE. NO. NA	DATE: 04-24-06	

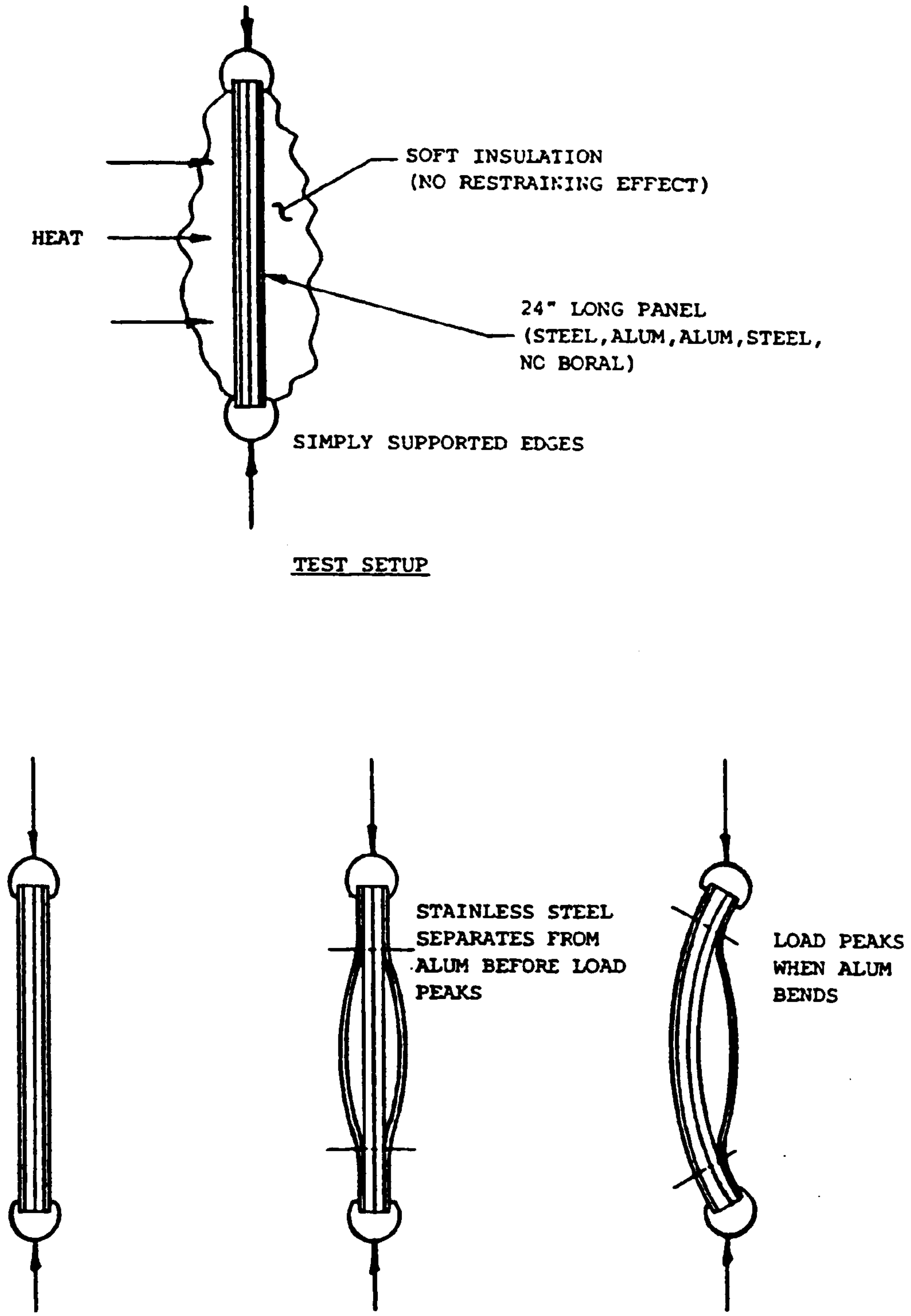

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Figure 4C.1-3

OBSERVED DEFORMATION MODES

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OBSERVED DEFORMATION MODES

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 1	FIG4C.1-3_REV_1
	PAGE. NO. NA	DATE: 04-24-06	

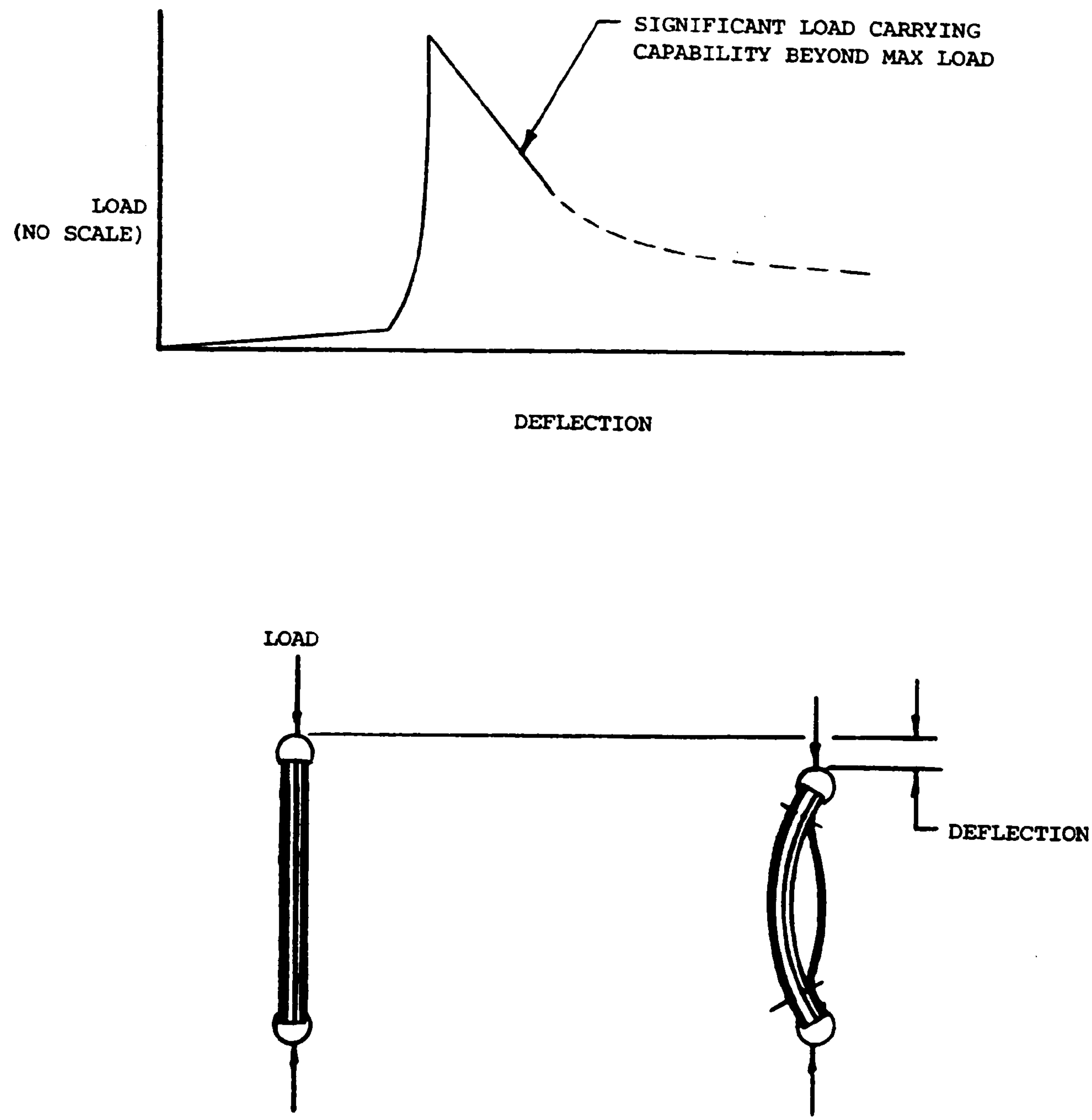

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Figure 4C.1-4

PANEL LOAD VS DEFLECTION CURVE

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PANEL LOAD VS DEFLECTION CURVE

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 1	FIG4C.1-4_REV_1
	PAGE. NO. NA	DATE: 04-24-06	

Panel No.	*Weld Spacing (in.)	Temperature, °F (at Max. Load)	Maximum Load (lb, Total)	Maximum Unit Load (lb/in.)
A1	5x12	RT	340,000	14,166
B1	5x8	RT	332,000	13,833
C1	5x6	RT	320,000	13,333
A2	5x12	529°	253,000	10,542
B2	5x8	405°	267,250	11,135
C2	5x6	365°	261,500	10,896

*Note: The weld spacing in the 8.05 in. load direction is always 5 in. The 12, 8 and 6 in. spacings are along the basket axial direction.

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Figure 4C.1-5

TN-40 BASKET PANEL COMPRESSION TEST RESULTS

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	BASED ON OVERALL PANEL GEOMETRY (CROSS SECTION ASSUMED STABLE)	BASED ON LOCAL PLATE GEOMETRY BETWEEN WELDS (LOCAL CROSS SECTION MAY BE UNSTABLE - PLATES MAY SEPARATE)
304SS PLATES	2814 LB/IN. (2545 FOR TN-40)	2055 LB/IN. (1707 FOR TN-40)
6061 T6 ALUMINUM PLATES	2896 LB/IN. (2805 FOR TN-40)	3234 LB/IN. (3145 FOR TN-40)

TOTAL TEST PANEL ALLOWABLE LOAD = 2055 + 2896 = 4951 LB/IN.
(4512 FOR TN-40)

*NOTE: PLATES USED IN TEST PANELS HAD GREATER THICKNESSES THAN SPECIFIED FOR TN-40 (304 SS THICKNESS WAS .110" VS. 100" AND 6061 T6 ALUMINUM THICKNESS WAS .256" VS. .250").

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Figure 4C.1-6

TEST PANEL ALLOWABLE COMPRESSIVE LOAD* USING TN-40 CRITERIA AND ACTUAL TEST PANEL DIMENSIONS (SLIGHTLY THICKER PLATES THAN TN-40 DESIGN)

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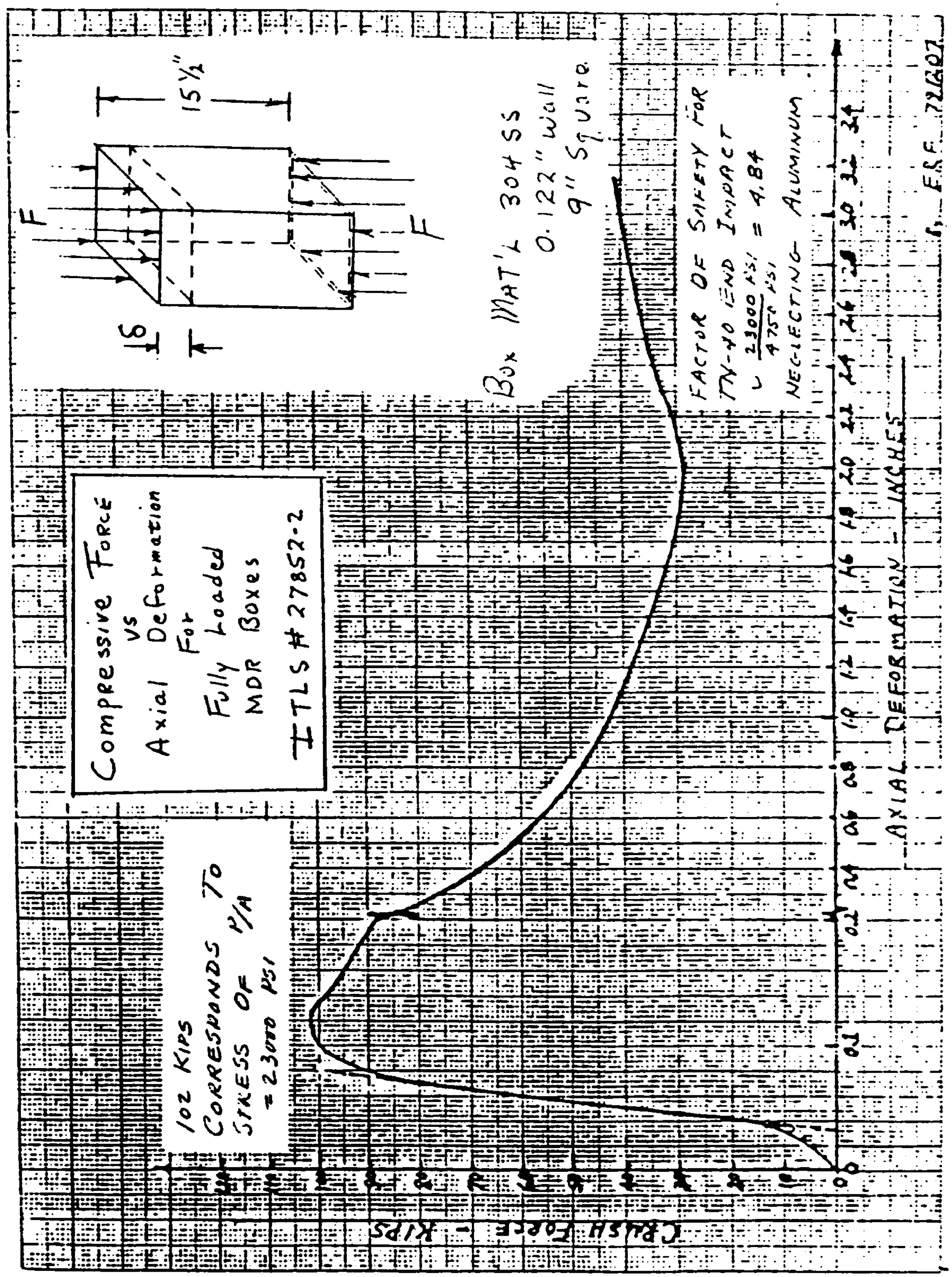

Figure 4C.2-1
AXIAL CRUSH TEST OF "BARE" BOX SECTION
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AXIAL CRUSH TEST OF BARE BOX SECTION

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 1	FIG4C.2-1_REV_1
	PAGE. NO.	DATE: 04-24-06	

SECTION 5

STORAGE SYSTEM OPERATIONS

5.1 OPERATION DESCRIPTION

5.1.1 NARRATIVE DESCRIPTION

Fuel handling and cask loading operations in the Auxiliary Building will be done in accordance with requirements of the Prairie Island Nuclear Generating Plant Operating License. Cask transport and storage at the ISFSI will be subject to requirements of the ISFSI Operating License. Operating activities in the Auxiliary Building and during transport and storage are described below.

The fuel storage operation will commence with the cask being brought into the Auxiliary Building rail bay through the west roll-up door. The cask weather cover and the cask lid will be removed and the cask inspected. A lifting yoke will be attached to the cask and connected to the Auxiliary Building crane hook. The cask will be laterally transferred and lifted by the crane from the basemat, elevation 695 feet, through a large opening in the floor slab at the 755 ft. elevation, laterally transferred and aligned with the access door to the fuel pool area, moved directly north to above the loading and unloading area of Pool No. 1 and lowered to the surface of the pool. The load path for the cask is illustrated in Figures 4.4-1 through 4.4-3. The narrow slot in the ceiling of the pool enclosure prohibits any movement of the cask except in the north-south direction. 01106798

The cask containment lid will be removed and the cask will be lowered into the spent fuel pool. Fuel assemblies will be loaded into the cask using a long handled tool suspended from the spent fuel pool bridge crane hoist and manipulated by an operator standing on the movable bridge over the pool.

After the cask is loaded with spent fuel and the lid is placed on the cask, the cask will be lifted to the pool surface and the lid bolts will be installed. The internal fuel cavity will be drained by displacing the water with air or with a suitable drain pump.

The cask will be returned to the Auxiliary Building rail bay by retracing the load path described above. The cask will be decontaminated in the Auxiliary Building rail bay/cask decontamination area and will be dried by using a vacuum system. The cavity will be filled with helium to design pressure and the cask lid seal will be leak tested. The top neutron shield will be installed on the lid. The overpressure monitoring system will be installed, and the interspaces between the double metallic seals pressurized to equilibrium pressure. Prior to transfer from the Auxiliary Building to the ISFSI, the cask will be monitored for contamination, temperature, radiation dose rates and the proper functioning of the seal tightness monitoring system. The protective cover will be installed and the pressure transducer connections fed through the external fitting. 01106798 01106798

The transport vehicle will be pulled to the ISFSI site by a tow vehicle as described in Section 5.4. The cask will be set in its storage position. The cask will be connected to the cask seal monitoring system and a functional check of the monitoring system will be performed.

5.1.2 FLOW SHEETS

The sequence of operations performed in loading fuel into the TN-40 storage cask and placing the cask into storage at the ISFSI is outlined in Table 5.1-1 and is shown in simplified flowsheet form in Figure 5.1-1.

Details of the number of personnel and the time required for the various operations are given in Table 5.1-2 for use in the radiation exposure determinations developed in Chapter 7. The data are based on Transnuclear's experience with transport cask operations.

5.1.3 IDENTIFICATION OF SUBJECTS FOR SAFETY AND RELIABILITY ANALYSIS

5.1.3.1 CRITICALITY PREVENTION

As discussed in Section 3.3.4, criticality is controlled by utilizing poison materials in the fuel basket and in the spent fuel pool water. These features are only necessary underwater during cask loading in the spent fuel pool. During storage, with the cavity dry and sealed from the environment, no further criticality control measures within the installation are necessary because of the low reactivity of the dry cask and the assurance that no unborated water can enter the cask during storage.

5.1.3.2 INSTRUMENTATION

Due to the totally passive and inherently safe nature of the storage casks, there is no need for any instrumentation to perform safety functions. However, transmitters are utilized to monitor the cask seals for leakage and are described in Section 3.3.3. The transmitters monitor the pressure in an interspace between the double metallic seals to provide an indication of seal failure before any release occurs.

An initial function check is performed at the manufacturer's plant and another function check of the transmitters is performed in preparation for cask storage.

5.1.3.3 MAINTENANCE TECHNIQUES

Because of their passive nature, the storage casks will require little, if any, maintenance over the lifetime of the ISFSI. Typical maintenance tasks involve occasional replacement and recalibration of monitoring instrumentation and recoating of some casks with corrosion-inhibiting coatings. No special maintenance techniques are necessary.

5.2 CONTROL ROOM AND CONTROL AREAS

This is a passive system and there is no need for control room annunciators or other systems to indicate off-normal conditions. A panel outside the ISFSI site will provide an alarm for the cask pressure monitoring equipment.

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5.3 SPENT FUEL ACCOUNTABILITY PROGRAM

Accountability and control of special nuclear materials will be maintained at all times during the loading, transport, and storage of spent fuel assemblies. Accountability records for all fuel assemblies transferred to, stored in or removed from the ISFSI will be maintained for as long as the fuel is stored on the Prairie Island site.

Fuel management records shall be utilized to verify that the initial fuel enrichment and fuel burnup of the fuel assemblies to be stored in the ISFSI are within the cask design criteria limits. Fuel management records will be reviewed to ensure that the assemblies to be put in each cask have not been previously identified as being failed fuel (leakers). The fuel assemblies shall also be visually inspected for physical damage which could potentially cause problems during insertion and/or removal from the storage cask. Each fuel assembly is engraved with a unique identification number and a vendor identification, which is unique to the site for which the fuel assemblies were fabricated. This will allow visual confirmation of the identity of the fuel assemblies placed in the cask.

The visible physical identification on each fuel assembly in combination with the administrative assignment of cell storage locations within the storage cask, and a unique serial number stamped on each storage cask, will allow for the positive identification of the locations of all ISFSI spent fuel assemblies.

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5.4 SPENT FUEL TRANSPORT TO ISFSI

The transport vehicle used to move the casks from the Auxiliary Building to the ISFSI is described in Section 4.3. A tow vehicle, such as a diesel tractor, will be used to pull the transport vehicle.

The casks will be loaded onto the transport vehicle in the Auxiliary Building rail bay. The rail bay is on the basemat of the Auxiliary Building.

The transport vehicle will be pulled out of the rail bay and out to the plant protected area fence along the same route as the railroad spur. The railroad spur (No. 1) is constructed on top of 6 inch by 8 inch railroad ties covered by 1 1/2 inch thick coarse bituminous binder and a 1 1/2 inch thick coarse bituminous surface. The ties are placed on 12 inch thick ballast.

The transport vehicle will be pulled over a newly designed and constructed access road. The access road is constructed of a 14 inch thick MN/DOT Class 5 compacted aggregate and is designed for transport vehicle loads with a loaded cask. The plan and profile of the access road is shown in Figure 5.4-1.

The transport vehicle will be pulled over the access road to the ISFSI site which also consists of a 14 inch thick MN/DOT Class 5 aggregate. The ISFSI site is designed with ample area to allow the transport vehicle to be maneuvered up onto the cask storage pads from any direction. A plan of the site is shown in Figure 1.3-1. The casks will then be unloaded from the transport vehicle onto the pads for storage.

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5.5 SPENT FUEL TRANSFER TO TRANSPORT CASK

Assuming that the spent fuel in the TN-40 cask will be transferred to a licensed transport cask using a normal “in pool” fuel transfer, the sequence of operations described in Section 5.1 and listed in Table 5.1-1 will be essentially performed in reverse.

The cask will be moved from the ISFSI back into the Auxiliary Building rail bay using the cask transport vehicle. The weather cover will be unbolted and removed. The overpressure system will then be removed and the cavity gas sampled through the vent port.

After moving the cask into the fuel pool area, the cavity will be depressurized and the cask lowered into the spent fuel pool. With the cask lid at the pool surface, fill and drain lines will be connected to the lid drain and vent ports. Borated water will be slowly added to fill the cask and to gradually cool the fuel in the cask. When the cask is full, the fill and drain lines will be removed. The cask will then be lowered to the pool bottom where the lid would be removed making the fuel accessible for transfer.

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5.6 REFERENCES

None.

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TABLE 5.1-1

PAGE 1 OF 3

SEQUENCE OF OPERATIONS

A. Receiving

1. Unload empty cask and separately packaged seals at plant site.

2. Inspect the following for shipping damage: exterior surfaces, sealing surfaces, trunnions, seals, accessible interior surfaces and basket assembly, bolts, bolt holes and threads, neutron shield vents.

3. Remove weather shield and install plug in neutron shield vent hole (threaded hole in the top of the steel shell surrounding the resin which contains a pressure relief valve during storage).

4. Remove lid bolts and lid.

5. Install protective plate over cask body sealing area.

6. Obtain lid and lid seal from storage.

7. Attach lid seal to lid by means of six retaining screws.

8. Move to spent fuel pool area.

B. Spent Fuel Pool Area

1. Lower cask into cask loading pool.

2. Load preselected spent fuel assemblies into the 40 basket compartments.

3. Verify identity of the fuel assemblies loaded into the cask.

4. Remove protective plate from cask body flange.

5. Lower lid and place on cask body flange over the two alignment pins.

6. Lift cask to surface of pool and install some lid bolts.
Note: Steps B.7 through B.11 may use a drain pump with a quick-disconnect coupling or suction lance to drain water as an alternate method.

7. Connect drain line to quick-disconnect coupling in the drain port.

8. Bolt special adapter, with quick-disconnect coupling, to vent port bolt holes.

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TABLE 5.1-1

SEQUENCE OF OPERATIONS

9. Connect plant compressed air line to special adapter quick-disconnect coupling.

10. Pressurize cavity to force water from cavity through drain port to the spent fuel pool.

11. Disconnect plant compressed air line and drain line from their quick-disconnect couplings.

12. Move cask to the decontamination area.

C. Decontamination Area (Rail Bay)

1. Decontaminate cask until acceptable surface dose levels are obtained.

2. Install remaining lid bolts and torque lid bolts using the prescribed procedure. 01106798

3. Remove plug from neutron shield vent and install pressure relief valve.

4. Connect Vacuum Drying System (VDS) to vent port.

5. Evacuate cavity to remove remaining moisture using prescribed procedure.

6. Evacuate cavity to 10 millibar and backfill with dry helium gas.

7. Pressurize cavity to about 2 atm with helium.

8. Disconnect VDS at vent port and install vent port cover with seal and bolts.

9. Perform helium leak test of lid seals.

10. Remove overpressure test connector.

11. Load cask on transport vehicle.

12. Check external surface temperatures using infrared camera or equivalent. 01106798

13. Install top neutron shield drum.

14. Torque the bolts using prescribed procedure.

15. Perform leak test on overpressure system.

TABLE 5.1-1

PAGE 3 OF 3

SEQUENCE OF OPERATIONS

16. Pressurize overpressure system (seal interspaces) with helium to a pressure of about 5.5 atm. 01106798

17. Check surface radiation levels.

18. Install protective cover with seal and bolts (could be performed at storage area).

19. Move cask to storage area.

D. Storage Area

1. Position cask in preselected location on storage pad.

2. Unload cask from transport vehicle. 01106798

3. Check for surface defects.

4. Connect pressure instrumentation to cask and to monitoring panel.

5. Check that pressure instrumentation is functioning.

6. Check surface radiation levels.

TABLE 5.1-2

ANTICIPATED TIME AND PERSONNEL REQUIREMENTS FOR CASK HANDLING OPERATIONS

OPERATION	NO. OF PERSONNEL	TIME (MIN)	AVG. DISTANCE (ft) FROM CASK
RECEIVING (Table 5.1-1 [A1 – A8])			
1. Unloading	*	*	*
2. Inspection	*	*	*
3. Transfer to cask	*	*	*
CASK LOADING POOL (Table 5.1-1 [B1 – B12])			
4. Lower cask into pool	*	*	*
5. Load fuel	5	*	*
6. Place lid on cask	5	*	*
7. Lift cask to pool surface	5	30	5
8. Install lid bolts	5	120	3
9. Drain cavity	5	90	6
10. Transfer to decontamination area	3	60	10
DECONTAMINATION AREA (RAIL BAY) (Table 5.1-1 [C1 – C19])			
11. Decontaminate cask	3	120	3
12. Remove vent plugs	2	30	5
13. Drying, evacuating, backfilling	2	480	5

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TABLE 5.1-2

PAGE 2 OF 3

ANTICIPATED TIME AND PERSONNEL REQUIREMENTS FOR CASK HANDLING OPERATIONS

OPERATION	**NO. OF PERSONNEL**	**TIME (MIN)**	**AVG. DISTANCE (ft) FROM CASK**
14. Install top neutron shield	2	15	3
15. Install pressure transducers	2	30	5
16. Pressurize interspace	*	*	*
17. Check leakage	2	30	5
18. Check surface temperature	2	30	5
19. Check surface dose rate	2	30	3
20. Install protective cover	2	30	5
21. Load transport vehicle	3	60	5
22. Transfer to storage area	3	60	10
STORAGE AREA (Table 5.1-1 [D1 – D6])			
23. Unload from vehicle position in location	5	60	5
24. Check surface dose rate	5	30	3
25. Connect pressure instrumentation	5	30	5

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TABLE 5.1-2

ANTICIPATED TIME AND PERSONNEL REQUIREMENTS FOR CASK HANDLING OPERATIONS

OPERATION	NO. OF PERSONNEL	TIME (MIN)	AVG. DISTANCE (ft) FROM CASK
PERIODIC MAINTENANCE			
1. Visual surveillance (NA)	2	15	5
2. Repair surface defects (NA)	2	60	3
3. Instrument testing and calibration	2	180	5
4. Instrument repair (NA)	2	60	3
MAJOR MAINTENANCE (ONCE IN 20 YEARS)			
1. Replace cask lid seals	3	1950 **	8

* No measurable dose associated with this activity. Therefore, the number of personnel, time and distance are not significant.

\# Parenthetical information corresponds to Table 5.1-1 activity numbers.

** Total time to transfer cask to spent fuel pool, replace lid seals, and return cask to ISFSI pad.

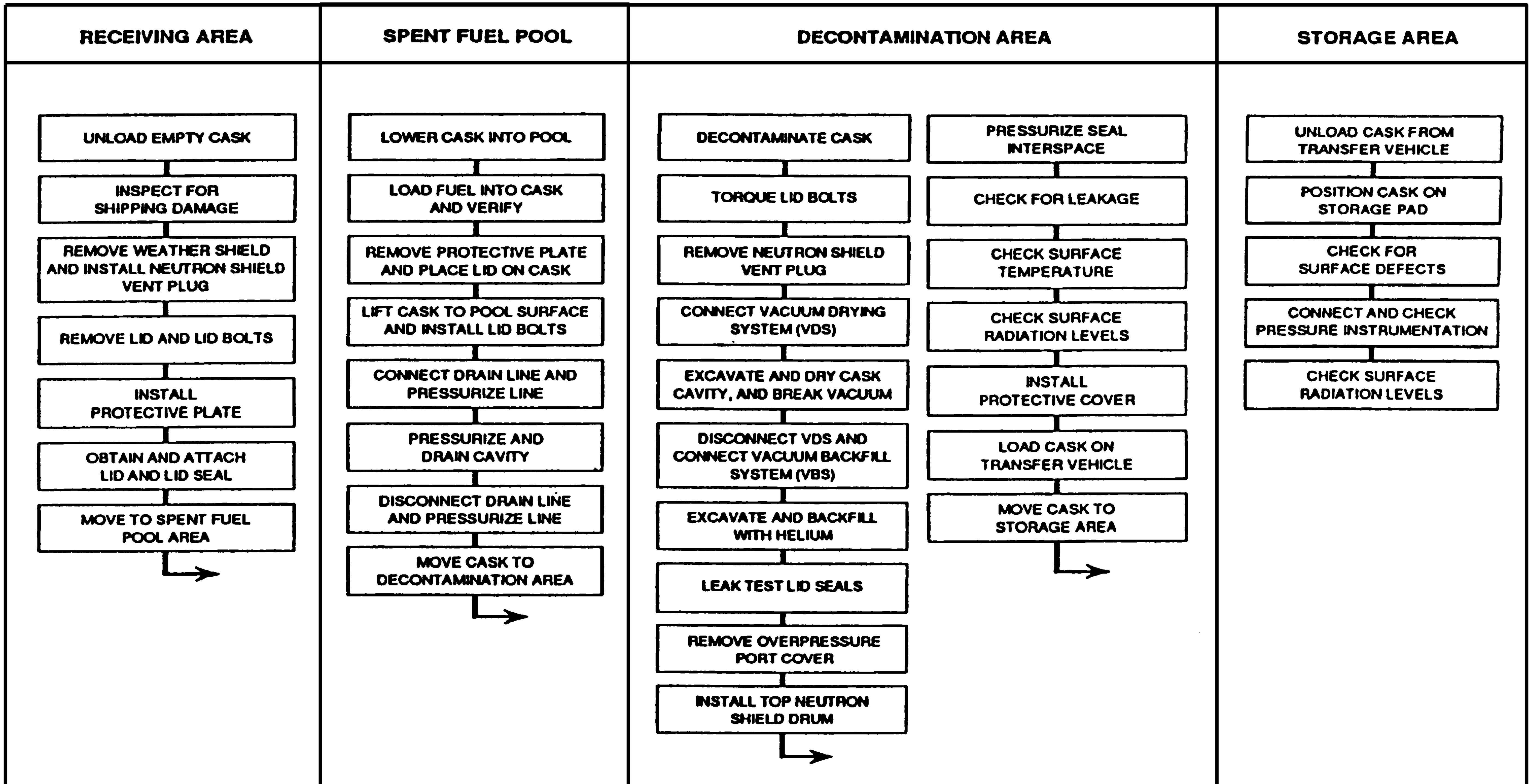

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Figure 5.1-1

SEQUENCE OF OPERATIONS

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

SEQUENCE OF OPERATIONS			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 2	FIG5.1-1_REV_2
	PAGE. NO.	DATE: 04-24-06	

Figure 5.4-1
ACCESS ROAD - PLAN AND PROFILE
PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

Figure Withheld Under 10 CFR 2.390

ACCESS ROAD - PLAN AND PROFILE			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG5.4-1_REV_3
	PAGE. NO.	DATE: 04-24-06	

SECTION 6

WASTE MANAGEMENT

6.1 DESIGN

No radioactive wastes will be generated during storage of the casks at the ISFSI or during cask transport outside of the Auxiliary Building. Radioactive wastes generated during loading operations in the Auxiliary Building will be treated using existing Prairie Island Nuclear Generating Plant radioactive waste control systems as described in Section 9 of the Prairie Island USAR (Reference 1).

Contaminated pool water removed from loaded storage casks will normally be drained back into the spent fuel pool with no additional processing. A small amount of liquid waste will result from storage cask decontamination. The decontamination procedure will result in a small amount of a detergent/demineralized water mixture being collected in the cask decontamination area. Liquid wastes collected in the cask decontamination area are directed to the aerated waste sump tank, where it will be mixed with other plant liquid wastes, treated or held up for decay, and released.

Potentially contaminated air and helium purged from the storage casks following spent fuel loading will be handled by the spent fuel pool ventilation systems as described in the Prairie Island USAR, Section 10.3.7, or the gaseous radwaste system as described in the USAR, Section 9.3. Air in the spent fuel pool area is normally exhausted through roughing and HEPA filters. In the event of a high radiation signal, ventilation is performed by the spent fuel pool special ventilation system, which has roughing, HEPA and activated charcoal filters.

A small quantity of low level solid waste will be generated as a result of storage cask loading operations and transfer cask decontamination. The solid waste generated will be processed as described in the Prairie Island USAR, Section 9.4. This low level waste will consist of disposable anti-contamination garments, tape, blotter paper, rags, etc.

The ISFSI Decommissioning Plan is detailed in Section 4.6.

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6.2 REFERENCES

1. Northern States Power Company, Prairie Island Nuclear Generating Plant Updated Safety Analysis Report, Revision 10, Docket 50-282 (Unit 1) and 50-306 (Unit 2)

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SECTION 7

RADIATION PROTECTION

7.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS REASONABLY ACHIEVABLE (ALARA)

7.1.1 POLICY CONSIDERATIONS AND ORGANIZATION

A radiological protection program will be implemented at the ISFSI in accordance with requirements of 10CFR72.126. The program will be based upon policies in existence at the Prairie Island Nuclear Generating Plant, which are described below.

The Prairie Island site shielding and radiation protection policies are described in Section 12.3 of the Prairie Island USAR (Reference 1). These policies will be applied to the Independent Spent Fuel Storage Installation. NSPM is committed to a strong 01153239
ALARA program in design and operation of nuclear facilities. The ALARA program which is applied to the ISFSI is the same as used at the Prairie Island Nuclear Generating Plant. Plant and design personnel are trained and updated on ALARA practices and dose reduction techniques. Design and implementation of systems and equipment are reviewed to insure ALARA criteria are met on all new and modification projects.

The ALARA program ensures that:

1. An effective ALARA program is administered at the Prairie Island Nuclear Generating Plant that appropriately integrates management philosophy and NRC regulatory requirements and guidance.

2. Facility design features, operating procedures and maintenance practices are in accordance with ALARA program guidelines; and that written reviews of the on-site radiation protection program assure that objectives of the ALARA program are attained.

3. Pertinent information concerning radiation exposure of personnel from other utilities and research work are reflected in design and operation.

4. Appropriate experience gained during the operation of nuclear power stations relative to in-plant radiation control is factored into revisions of procedures to assure that the procedures continually meet the objectives of the ALARA program.

5. Necessary assistance is provided to insure that operations, maintenance, and decommissioning activities are planned and accomplished in accordance with ALARA objectives.

6. Trends in station personnel and job exposures are analyzed in order to permit corrective actions to be taken with respect to adverse trends.

Reports of the findings of the radiation protection staff are also effectively conveyed to management.

Specific responsibilities of station personnel are to ensure that:

1. Activities are planned and accomplished in accordance with the objectives of the ALARA program.

2. Procedures and their revisions are implemented in accordance with the objectives of the ALARA program.

3. The general office radiation protection staff is consulted as necessary for assistance in meeting ALARA program objectives.

The primary goal of the radiation protection and ALARA programs is to minimize exposure to radiation such that the total individual and collective exposure to personnel in all phases of design, construction, operation and maintenance are kept As Low As Reasonably Achievable. This is achieved by integrating ALARA concepts into design, construction, and operation of facilities.

Trained personnel adequate to develop and conduct all necessary radiation protection and ALARA programs are provided. These personnel are trained to assure that all procedures are followed to meet company and regulatory requirements. Training programs in the basics of radiation protection and exposure control are provided to all facility personnel whose duties require working in radiation areas.

The administrative organization is responsible for and has appropriate authority for assuring that the three basic objectives of the radiation protection program are achieved. These objectives are to:

1. Protect personnel

2. Protect the public

3. Protect the facility

<u>Protection of Personnel</u> Includes surveillance and control over internal and external radiation exposure and maintaining the exposure of all personnel within permissible limits and as low as reasonably achievable (ALARA).

<u>Protection of the public</u> includes surveillance and control over all conditions and operations that may affect the health and safety of the public. Included are such activities as radioactive gas, liquid, and solid waste disposal, shipment of radioactive materials, an environmental radioactivity monitoring plan and maintaining portions of the station emergency plan.

<u>Protection of the Facility</u> includes monitoring to warn of possible detrimental changes and exposure hazards, to determine changes or improvement needed, and to note trends for planning future work.

This administrative organization is also responsible for and has appropriate authority for maintaining occupational exposures as far below the specified limits as reasonably achievable by assuring that:

1. Station personnel are made aware of management's commitment to keep occupational exposures as low as reasonably achievable;

2. Formal reviews are performed periodically to determine how exposures might be lowered.

3. There is a well-supervised radiation protection capability with specific defined responsibilities;

4. Station workers receive sufficient training;

5. Sufficient authority to enforce safe station operation is provided;

6. Modification to operating and maintenance procedures and to station equipment and facilities are made where they should substantially reduce exposures at a reasonable cost;

7. The radiation protection staff understands the origins of radiation exposures in the station and seeks ways to reduce exposures;

8. Adequate equipment and supplies for radiation protection work are provided.

The Site Vice President is responsible for the protection of all persons against radiation and for compliance with NRC regulations and license conditions. This responsibility is in turn shared by all Managers. Furthermore, all personnel are required to work safely and to follow the regulations, rules, and procedures that have been established for their protection.

The Radiation Protection and Chemistry Manager is responsible for the Radiation Protection Program, including the program for handling and monitoring radioactive material, that is designed to assure compliance with applicable regulations, technical specifications, and regulatory guides. This person also provides technical guidance and support for conducting this program, reviews the effectiveness and the results of the program and modifies it as required based on experience and regulatory changes, to assure that occupational radiation exposure and exposure to the general public are maintained as low as reasonably achievable.

The Radiation Protection General Supervisor is responsible for radiation safety. This duty includes the authority to measure and control the radiation exposure of personnel; to continuously evaluate and review the radiological status of the station; to make recommendations for control or elimination of radiation hazards; to assure that all personnel are trained in radiation protection; to assist all personnel in carrying out their radiation protection responsibilities; and to protect the health and safety of the public both on-site and in the surrounding area. In order to achieve the goals of the Radiation Protection Program and fulfill these responsibilities for radiation protection, radiological monitoring, survey and personnel exposure control work are performed on a continuing basis for station operations and maintenance including the ISFSI.

7.1.2 DESIGN CONSIDERATIONS

The equipment design takes into account radiation protection considerations, which ensure that occupational radiation exposures are ALARA. The fuel will be stored dry, inside sealed, heavily-shielded casks. The most significant radiation protection design consideration provides for heavy shielding to minimize personnel exposures. To avoid personnel exposure, the casks will not be opened nor fuel removed from the casks while at the ISFSI. Storage of the fuel in dry sealed casks eliminates the possibility of leakage of contaminated liquids. Gaseous releases are not considered credible. The exterior of the casks will be decontaminated before leaving the Auxiliary Building, thereby minimizing exposure of personnel to surface contamination. The storage casks will contain no active components which require periodic maintenance or surveillance. This method of spent fuel storage minimizes direct radiation exposures and eliminates the potential for personnel contamination.

Both concrete storage pads and the Equipment Storage Building at the ISFSI will be constructed prior to ISFSI operation. This will be done to eliminate occupational radiation exposure which would result from additional construction following placement of storage casks in the ISFSI.

An annunciator panel monitoring cask pressure will be located outside of the ISFSI protected area. This will minimize time required for periodic cask surveillance and reduce personnel exposure.

The ISFSI site is within the exclusion area of the Prairie Island site. The location of the ISFSI is of sufficient distance from frequently occupied areas of the Prairie Island Nuclear Generating Plant such that the increased dose to personnel will not be significant.

Regulatory Position 2 of Regulatory Guide 8.8, is incorporated into design considerations, as described below:

- ALARA objective 2a on access control is met by use of a fence with a locked gate that surrounds the ISFSI and prevents unauthorized access.
- Regulatory Position 2b on radiation shielding is met by the heavy shielding of the casks which minimizes personnel exposures.
- Regulatory Position 2c on process instrumentation and controls is met by designing the instrumentation for a long service life and locating readouts in a low dose rate location.
- Regulatory Position 2d on control of airborne contaminants does not apply because no gaseous releases are expected. No significant surface contamination is expected because the exterior of the casks and racks will be decontaminated before they leave the decontamination area in the Auxiliary Building.
- Regulatory Position 2e on crud control is not applicable to the ISFSI because there are no systems at the ISFSI that could transport crud.
- Regulatory Position 2f on decontamination is met because the exteriors of the casks are designed for decontamination. The casks and racks are decontaminated before they are released from the decontamination area in the Auxiliary Building.
- Regulatory Position 2g on radiation monitoring does not apply because the casks are sealed. There is no need for airborne radioactivity monitoring since no airborne radioactivity is anticipated. Area radiation monitors will not be required because the ISFSI will not normally be occupied; however, TLDs will be installed along the controlled access fence.
- Regulatory Position 2h on resin treatment systems is not applicable to the ISFSI because there will be no radioactive systems containing resins.
- Regulatory Position 2i concerning other miscellaneous ALARA items is not applicable because these items refer to radioactive systems not present at the ISFSI.

7.1.3 OPERATIONAL CONSIDERATIONS

The ALARA procedures for the ISFSI will be the same as those used in the radiation protection program for Prairie Island Nuclear Generating Plant. Section 7.1.1 describes the policy and procedures that ensure that ALARA occupational exposures and contamination levels are achieved. Section 7.1.2 describes how the design considerations are ALARA.

Storage of spent fuel in storage casks is expected to involve lower exposures than other alternative methods or designs for onsite storage. For example, storage in a fuel pool would involve use of radioactive water cooling and cleanup systems and filtered HVAC that would result in higher operator exposures during pump, valve, and motor maintenance of these systems, and filter and resin replacement. This alternative would also lead to additional airborne and liquid releases that will not be present at the ISFSI.

Operational requirements for surveillance are incorporated into the design considerations in Section 7.1.2 in that the casks are stored with adequate spacing to allow ease of on site surveillance. In addition, annunciation will be available outside the ISFSI protected area to minimize surveillance time. The operational requirements are incorporated into the radiation protection design features described in Section 7.3 since the casks are heavily shielded to minimize occupational exposure.

The ISFSI contains no systems that process liquids or gases or contain, collect, store, or transport radioactive liquids or solids other than the stored spent fuel and contaminated spent fuel racks. Therefore, the ISFSI meets ALARA requirements since there are no such systems to be maintained, be repaired, or be a source of leaks.

7.2 RADIATION SOURCES

7.2.1 CHARACTERIZATION OF SOURCES

There are five principal sources of radiation associated with cask storage that are of concern for radiation protection. These are:

1. Primary gamma radiation from spent fuel.
2. Primary neutron radiation from spent fuel.
3. Gamma radiation from activated fuel structural materials.
4. Capture gamma radiation produced by attenuation of neutrons by shielding material of the cask.
5. Neutrons produced by sub-critical fission in fuel.

Primary gamma and neutron radiation sources for the stored spent fuel are generated by using the ORIGEN2 computer code (Reference 5). A 3.85% enrichment Westinghouse OFA 14x14 fuel assembly, as described in Section 3.1.1, is modeled in three zones for the ORIGEN2 calculation (Reference 6). The structural material masses are listed in Table 7.2-1. The structural material compositions and fuel impurities are taken from Reference 6. In particular, the cobalt impurities in Inconel, Zircaloy and stainless steel are 0.47%, 0.001% and 0.08% respectively.

The fuel zone is irradiated at a constant specific power of 37.5 MW/MTU to a total burnup of 45,000 MWD/MTU. A conservative three-cycle operating history is utilized with 30 days down time each cycle except for no down time in the last cycle. After the fuel zone irradiation, the flux generated by ORIGEN2 for the fuel zone is used to irradiate the plenum and end zones. However, the methodology of Reference 6 is used to correct the plenum and end zone irradiations. The cobalt, Zircaloy and manganese masses in the structural materials are multiplied by 0.67, 0.40 and 0.80 respectively and the irradiation flux for the plenum and end zones is multiplied by 0.042 and 0.011 respectively. Those factors are used to correct for the spatial and spectral changes of the neutron flux outside of the fuel zone.

Gamma and neutron sources are generated for cooling times from 10 years to 30 years.

Table 7.2-2 shows the total primary gamma source and neutron source. Fission product activities are listed on Table 7.2-3 and activation activities are listed on Table 7.2-3A.

The primary gamma source spectrum, group structure and ANSI/ANS-6.1.1 flux-to-dose factors are listed in Table 7.2-4. The energy groups are those output by ORIGEN2. The fuel zone gamma spectrum contains a contribution of activated materials from the materials listed in Table 7.2-1.

The neutron source consists of spontaneous fission and alpha-n reactions mainly from Cm-244. Spectra for both mechanisms are obtained from Reference 7. However the spectra of Reference 7 must be converted into the proper neutron energy groups of the SCALE 27n-18g library (Reference 8). This is accomplished by apportioning the Cm-244 spectra into the SCALE energy groups. For example, SCALE Group 6 contains neutrons with energies between 0.4 and 0.9MeV. This encompasses two groups and part of a third of the Reference 7 spontaneous fission spectrum (i.e., 0.4-0.6, 0.6-0.8 and 0.8-1.0).

Therefore, the fraction in SCALE Group 6 is:

$$0.07973 + 0.08156 + 0.07056 \times \frac{0.9 - 0.8}{1.0 - 0.8} = 0.1966$$

The spectra from Reference 7 for both spontaneous fission and alpha-n reactions are converted into equivalent neutron spectra in the SCALE energy groups in this manner. A single combined neutron spectrum is then calculated using 93.7% spontaneous fission and 6.3% alpha-n. Those percentages are obtained from the ORIGEN2 output. Table 7.2-5 lists the neutron spectra.

The primary gamma spectrum is also converted into the SCALE 27n-18g energy groups. In addition to simple apportioning of overlapping groups as is done above for the neutrons, the gammas are weighted by the ratio of the ORIGEN2 and SCALE library average group energies because ORIGEN2 weights the photon yields in this fashion. Table 7.2-6 lists this spectrum and ANSI flux-to-dose factors for the SCALE energy groups.

Gamma radiation produced by capture of neutrons in cask shielding materials is computed in the shielding analysis with the coupled SCALE 27n-18g neutron-gamma library. Similarly, neutrons produced from sub-critical multiplication are accounted for in the shielding calculation.

7.2.2 AIRBORNE RADIOACTIVE SOURCES

The quantity of fission gas produced in a typical PWR assembly for the design irradiation condition is approximately 600 liters at STP (Reference 9). Of this quantity, only a small fraction is radioactive with Kr-85 being the dominant nuclide. Most of the fission products generated are retained within the fuel pellet. A small fraction, nominally 30% for Kr-85 and 10% for noble gases, is released into the fuel rod plenums. Table 7.2-7 shows the inventory of fission gases and volatile nuclides contained in 40 of the design basis fuel assemblies. There are the total curves in the fuel assemblies, not just in the plenums.

The radioactive sources resulting from cask storage are safely confined both within the fuel cladding and within the cask containment during storage. Accordingly, airborne provisions for personnel protective measures against airborne sources are not required.

7.3 RADIATION PROTECTION DESIGN FEATURES

7.3.1 STORAGE SYSTEM DESIGN DESCRIPTION

A description of the ISFSI, including layout and characteristics is provided **in Section 4.1.**

The ISFSI has a number of design features which ensure that exposures are ALARA.

- There will be no radioactive systems in the ISFSI other than the storage casks.
- The casks will be loaded, sealed, and decontaminated prior to transfer to the ISFSI.
- The fuel will not be unloaded nor will the casks be opened at the ISFSI.
- The fuel will be stored dry inside the casks, so that no radioactive liquid is available for leakage.
- The casks will be sealed airtight.
- The casks will be heavily shielded to minimize external dose rates.

Also, the ISFSI will not normally be occupied. Therefore, no personnel areas, equipment decontamination areas, contamination control areas, or health physics facilities need be located at the ISFSI. These types of facilities are available at the Prairie Island Nuclear Generating Plant.

Due to the remoteness of the ISFSI from other normally occupied areas and the anticipated occupancy times for performing surveillance activities, the anticipated total effective dose equivalent to any individual member of the public in any period of one calendar year will be well below 0.1 rem in accordance with 10CFR20.1301(a)(1). Accordingly, areas outside the owner-controlled fenced area are unrestricted and the dose to unrestricted areas from the ISFSI cannot exceed 2.00 mrem in any one hour in accordance with 10CFR20.1301(a)(2). Worst case dose rates (assuming design basis fuel in 48 casks) at the outer fence boundary (nuisance fence), which is well within the restricted area boundary, have been calculated to be 1.99 mrem/hr.

7.3.2 SHIELDING

The design of the cask shielding, material of construction, and method of calculating shielding parameters are explained in Appendix 7A. A discussion of the earth berm shielding is contained in Appendix A7A. No special features or remote handling of the casks in the ISFSI are proposed. Portable shielding may be used during cask maintenance, if appropriate. 01244873

7.3.3 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

As indicated in Section 3.3.5.3, there are no credible events which could result in releases of radioactive products or unacceptable increases in direct radiation. Area radiation and airborne radioactivity monitors are therefore not needed at the ISFSI; however, TLDs will be used to record dose rates along the outer (nuisance) fence boundary of the ISFSI

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7.4 ESTIMATED ONSITE COLLECTIVE DOSE ASSESSMENT

Table 7.4-1 shows the estimated design basis occupational exposures to ISFSI personnel during the loading, transport, and emplacement of the storage casks. Dose rates at 1 meter were utilized for all cases except cask transfer, when individuals will typically be at least 2 meters away from the cask.

Table 7.4-2 shows the estimated design basis annual man-rem for surveillance and maintenance activities. Visual surveillance was based on a walkdown of each of the two pads at a distance no closer than 2 meters to the casks. The casks on the opposite pad contribute approximately 1 mrem/hr to the expected dose rate along with contributions from the casks on the near pad. It was assumed that the average dose rate for the surveillance was four times the dose rate calculated at 2 meters from a cask. To estimate the dose rates for operability tests and calibration, the worker was assumed to be located at the monitoring panel at the perimeter fence entrance. During instrument repairs, the worker was assumed to be positioned between two rows of casks. The six surrounding casks (all within 16 feet of the worker) were the predominant dose contributors during repair work, but dose rates from casks on the other pads were also considered.

Both Table 7.4-1 and 7.4-2 provide for each task the estimated time required for the task, number of personnel required, the design basis dose rates, and man-rem.

An evaluation of the additional dose to station personnel from ISFSI operations is contained in Section A7. 01244873

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7.5 OFFSITE COLLECTIVE DOSE ASSESSMENT

An evaluation of the offsite collective dose from ISFSI operations is contained in Section A7. 01244873

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7.6 RADIATION PROTECTION PROGRAM

The ISFSI is located on the site of the Prairie Island Nuclear Generating Plant within the Owner Controlled Area. The Radiation Protection and Chemistry Manager will have responsibility for radiation protection activities at the ISFSI. 01106798

The Radiation Protection and ALARA programs are discussed in Section 7.1.1.

Radiation protection requirements for all radiological work at the Prairie Island Nuclear Generating Plant are governed by existing station directives, and station Radiation Protection procedures. Radiation protection practices for cask loading, transfer, storage, monitoring, and retrieval will also be based on existing procedures, as well as on current and anticipated conditions when the task is to be performed. These procedures include, but are not limited to, the following:

- Procedure for personnel dosimetry issue.
- Issuance, revision, and termination of radiation work permits and standing radiation work permits.
- Procedure for roping off, barricading, and posting of radiation control zones.
- Decontamination procedure for equipment and areas.
- Smear swab sampling, counting, and calculation.
- Procedure for quantifying airborne radioactivity.

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7.7 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

An additional 16 TLDs will be placed around the ISFSI. These will be read as part of the existing radiological environmental monitoring program already in effect at the Prairie Island Nuclear Generating Plant. Since no effluents are expected from the ISFSI, the operation of the ISFSI will have minimal impact on the monitoring program.

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7.8 REFERENCES

1. Northern States Power Company, Prairie Island Nuclear Generating Plant Updated Safety Analysis Report, Revision 10, Docket 50-282 (Unit 1) and 50-306 (Unit 2).

2. Deleted

3. Deleted

4. Deleted

5. Oak Ridge National Laboratory, ORIGEN2 Isotope Generation and Depletion Code, CCC-371, January 1987.

6. A.G. Croff, et. al., Revised Uranium - Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code. ORNL/TM-6051, Oak Ridge National Laboratory, Oak Ridge, Tennessee, September 1978.

7. Bucholz, J.A., Scoping Design Analyses for Optimized Shipping Casks Containing 1-, 2-, 3-, 5-, 7-, and 10-Year-Old PWR Spent Fuel, ORNL/CSD/TM-149, Oak Ridge National Laboratory, Oak Ridge, Tennessee, January 1983.

8. U.S. Nuclear Regulatory Commission, SCALE: A modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation, ORNL/NUREG/CR-0200, Revision 3, December 1984.

9. Transnuclear, Inc., "Extended Fuel Burnup Demonstration Program Topical Report - Transport Considerations for Transnuclear Casks," DOE/ET 34014-11, White Plains, New York, December 1983.

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TABLE 7.2-1

MATERIAL DISTRIBUTION IN WESTINGHOUSE 14x14 OFA FUEL ASSEMBLY

		MATERIAL	MASS kg/MTU	MASS kg/assembly
FUEL ZONE				
	Cladding	Zircaloy-4	220.6	83.8
	Grid spacers and grid-spacer springs	Zircaloy-4 Inconel 718	17.4 3.7	6.6 1.4
	Guide Tubes	Zircaloy-4	19.5	7.4
FUEL-GAS PLENUM ZONE				
	Cladding	Zircaloy-4	10.8	4.1
	Guide Tubes	Zircaloy-4	1.0	0.4
	Spring	SS-304	7.3	2.8
END FITTING ZONE				
	Top End Fitting	SS-304	11.3	4.3
	Hold Down Spring	Inconel 718	1.8	0.7
	Bottom End Fitting	SS-304	11.8	4.5
TOTAL			305.2	116.0

TABLE 7.2-2

GAMMA AND NEUTRON RADIATION SOURCES
WESTINGHOUSE OFA 14x14 3.85 W/O U235
45,000 MWD/MTU, 10 YEAR COOLING TIME

Fission Product Activity (Curie/Assembly)	1.55E5
Neutron Source (n/sec/Assembly)	2.19E8
Fuel Zone Gamma Source (γ/sec/Assembly)	2.44E15
Plenum Zone Gamma Source (γ/sec/Assembly)	1.73E11
End Zone Gamma Source (γ/sec/Assembly)	2.06E11*

* Upper fitting (52.4%), Lower fitting (47.5%)

TABLE 7.2-3

FISSION PRODUCT ACTIVITIES (CURIES/MTU)
WESTINGHOUSE OFA 14x14 3.85 W/O U235,
45,000 MWD/MTU, 10 YEAR COOLING TIME

NUCLIDE	DISCHARGE	10 – YEAR	20 – YEAR
H 3	7.44E+02	4.25E+02	2.42E+02
Kr 85	1.21E+04	6.26E+03	3.33E+03
Sr 90	9.52E+04	7.51E+04	5.92E+04
Y 90	1.01E+05	7.51E+04	5.92E+04
Y 91	1.07E+06	1.74E-13	4.80E-31
Zr 95	1.60E+06	1.05E-11	6.10E-29
Nb 95	1.61E+06	2.32E-11	1.21E-30
Ru106	7.12E+05	7.84E+02	8.09E-01
Rh106	7.90E+05	7.84E+02	8.09E-01
Ag110	2.44E+05	3.67E-03	1.46E-07
Sb125	1.84E+04	1.52E+03	1.24E+02
Cs134	2.57E+05	8.90E+03	3.08E+02
Cs137	1.41E+05	1.12E+05	8.86E+04
Ba137M	1.33E+05	1.06E+05	8.38E+04
Ce144	1.27E+06	1.73E+02	2.34E-02
Pr144	1.29E+06	1.73E+02	2.34E-02
Pm 147	1.29E+05	9.75E+03	6.94E+02
Sm 151	4.79E+02	4.51E+02	4.17E+02
Eu154	1.72E+04	7.70E+03	3.44E+03
Eu155	1.10E+04	2.73E+03	6.74E+02
TOTAL	1.77E+08	4.07E+05	300E+05
	NEUTRON SOURCE (CURIES/MTU)		
Pu238	4.263E+03	4.331E+03	4.002E+03
Am241	1.892E+02	2.418E+03	3.758E+03
Cm244	5.665E+03	3.864E+03	2.635E+03
TOTAL	1.182E+04	1.061E+04	1.040E+04

TABLE 7.2-3a

ACTIVATION ACTIVITIES (CURIES/MTU)

ZONE	NUCLIDE	DISCHARGE	10-YR	20-YR
FUEL	H3	275.5	157.2	89.66
	C14	1.268	1.266	1.264
	Fe55	813.2	56.54	3.932
	Co60	2626	704.7	189.1
	Ni59	1.004	1.004	1.004
	Ni53	160.3	148.6	137.9
	Sbl25	2880	191.6	15.69
ENDS	Fe55	104.6	7.273	0.506
	Co60	27.21	7.303	1.960
	Ni63	2.881	2.672	2.478
PLENUM	Fe55	122.4	8.512	0.592
	Co60	22.6	6.06	1.63
	Ni63	2.352	2.182	2.023
	Sb125	4.549	0.377	0.031

TABLE 7.2-4

PRIMARY GAMMA SOURCE SPECTRUM
ORIGEN2 GROUP STRUCTURE
WESTINGHOUSE OFA 14x14 3.85 W/O U235,
45,000 MWD/MTU, 10 YEAR COOLING TIME

		γ/s/40 ASSEMBLIES			
AVG. ENERGY MeV	FLUX-TO-DOSE FACTOR (mrem/hr/ϕ)	FUEL ZONE	PLENUM ZONE	TOP FITTING	BOTTOM FITTING
0.125	3.26E-4	6.9832E15	1.12E9	1.919E9	1.740E9
0.225	5.66E-4	6.868E15	1.29E10	6.317E8	5.739E8
0.375	9.33E-4	3.191E15	7.65E10	1.769E8	1.607E8
0.575	1.31E-3	6.992E16	9.83E10	1.016E7	9.225E6
0.850	1.76E-3	6.924E15	2.53E7	3.443E9	3.128E9
1.25	2.32E-3	3.524E15	6.75E12	4.303E12	3.909E12
1.75	2.93E-3	8.204E13			
2.25	3.47E-3	1.394E12			
2.75	3.96E-3	9.940E10			
3.50	4.63E-3	1.322E10			

TABLE 7.2-5

NEUTRON SOURCE DISTRIBUTION
WESTINGHOUSE OFA 14x14 3.85 W/O U235
45,000 MWD/MTU, 10 YEAR COOLING TIME

SCALE Group	Alpha-n Spectrum	Spontaneous Fission Spectrum	Combined Spectrum	n/sec*
1	0.0	0.01883	0.01764	1.544E8
2	0.25271	0.20958	0.21231	1.858E9
3	0.52336	0.22657	0.24536	2.147E9
4	0.13084	0.13081	0.13081	1.145E9
5	0.06978	0.17915	0.17223	1.507E9
6	0.01981	0.19657	0.18538	1.622E9
7	0.00349	0.03849	0.03627	3.175E8
	1.0	1.0	1.00000	8.75E9

* For 40 Assemblies

TABLE 7.2-6

PAGE 1 OF 2

PARAMETERS FOR THE SCALE 27N-18G LIBRARY
WESTINGHOUSE OFA 14x14 3.85 W/O U235
45,000 MWD/MTU, 10 YEAR COOLING TIME

Group No.	Max Energy (eV)	Flux-Dose Factor (rem/hr/ϕ)	Primary Gamma* γ/sec)
1	2.000E+07	1.492E-04	0.000E+00
2	6.434E+06	1.446E-04	0.000E+00
3	3.000E+06	1.270E-04	0.000E+00
4	1.850E+06	1.281E-04	0.000E+00
5	1.400E+06	1.298E-04	0.000E+00
6	9.000E+05	1.028E-04	0.000E+00
7	4.000E+05	5.118E-05	0.000E+00
8	1.000E+05	1.232E-05	0.000E+00
9	1.700E+04	3.837E-06	0.000E+00
10	3.000E+03	3.725E-06	0.000E+00
11	5.500E+02	4.015E-06	0.000E+00
12	1.000E+02	4.293E-06	0.000E+00
13	3.000E+01	4.474E-06	0.000E+00
14	1.000E+01	4.568E-06	0.000E+00
15	3.050E+00	4.558E-06	0.000E+00
16	1.770E+00	4.519E-06	0.000E+00
17	1.300E+00	4.488E-06	0.000E+00
18	1.130E+00	4.466E-06	0.000E+00
19	1.000E+00	4.435E-06	0.000E+00
20	8.000E-01	4.327E-06	0.000E+00
21	4.000E-01	4.198E-06	0.000E+00
22	3.250E-01	4.098E-06	0.000E+00
23	2.250E-01	3.839E-06	0.000E+00

TABLE 7.2-6

PAGE 2 OF 2

PARAMETERS FOR THE SCALE 27N-18G LIBRARY
WESTINGHOUSE OFA 14x14 3.85 W/O U235
45,000 MWD/MTU, 10 YEAR COOLING TIME

Group No.	Max Energy (eV)	Flux-Dose Factor (rem/hr/ϕ)	Primary Gamma: γ/sec)
24	1.000E-01	3.675E-06	0.000E+00
25	5.000E-02	3.675E-06	0.000E+00
26	3.000E-02	3.675E-06	0.000E+00
27	1.000E-02	3.675E-06	0.000E+00
28	1.000E+07	8.772E-06	0.000E+00
29	8.000E+06	7.478E-06	0.000E+00
30	6.500E+06	6.375E-06	0.000E+00
31	5.000E+06	5.414E-06	0.000E+00
32	4.000E+06	4.622E-06	1.322E+10
33	3.000E+06	3.960E-06	9.939E+10
34	2.500E+06	3.469E-06	1.394E+12
35	2.000E+06	3.019E-06	5.335E+13
36	1.660E+06	2.628E-06	1.035E+15
37	1.330E+06	2.205E-06	2.502E+15
38	1.000E+06	1.833E-06	4.359E+15
39	8.000E+05	1.523E-06	2.578E+16
40	6.000E+05	1.173E-06	4.904E+16
41	4.000E+05	8.759E-07	2.279E+15
42	3.000E+05	6.306E-07	4.120E+15
43	2.000E+05	3.834E-07	9.211E+15
44	1.000E+05	2.669E-07	0.000E+00
45	5.000E+04	9.347E-07	0.000E+00

*For 40 Assemblies

TABLE 7.2-7

FISSION GAS AND VOLATILE NUCLIDES INVENTORY
(CURIES/40 ASSEMBLIES)
WESTINGHOUSE OFA 14x14 3.85 W/O U235
45,000 MWD/MTU, 10 YEAR COOLING TIME

Nuclide	10 Year Decay	20 Year Decay
H-3	6.46E3	3.68E3
Kr-85	9.67E4	5.06E4
Cs-134	1.35E5	4.68E3
Cs-137	1.70E6	1.35E6

TABLE 7.4-1

DESIGN BASIS OCCUPATIONAL EXPOSURES FOR CASK LOADING, TRANSPORT AND EMPLACEMENT (ONE TIME EXPOSURE)

TASK	TIME REQUIRED (hr)	NO. OF PERSONS	DOSE RATE (rem/hr)	DOSE (man-rem)
Placement in pool (A1-A8)*	2	3	0.005	0.03
Loading Process (B1-B3)	5	5	0.005	0.125
Removal from Pool (B4-B11)	5	5	0.030	0.75
Transfer to Decontamination Area (B12)	1	3	0.030	0.09
Processing of Cask (C2-C11)	6.5	2	0.030	0.39
Helium Leak Test (C12)	2	2	0.030	0.12
Decontamination (C1)	2	3	0.030	0.18
Install Neutron Shield, Pressurize, Test (C13-C22)	3	2	0.030	0.18
Preparation for Transport (C23)	1	3	0.030	0.09
Transfer of Cask to ISFSI (C24)	1	3	0.020	0.06
Final Cask Emplacement (D1-D6)	2	5	0.030	0.30
TOTAL				2.315

* Steps from Table 5.1-1

TABLE 7.4-2

**DESIGN BASIS ISFSI MAINTENANCE OPERATIONS
ANNUAL EXPOSURES**

TASK	TIME REQUIRED (hr)	NO. OF PERSONS	DOSE RATE (mrem/hr)	PERSON-rem
Visual Surveillance of Casks[1]	1	2	78.8	0.16
Instrumentation				
a. Operability Tests[2]	1	2	1.0	0.002
b. Calibration[3]	2	2	1.0	0.002
c. Repairs[4]	1	2	118	0.24
Surface Defect Repair[5]	1	2	118	0.24
Major Maintenance	32.5	3	25.6	2.5

NOTES

1. Assumes 4 yearly surveys, 15 minutes each.
2. Based on two tests per year, 30 minutes each.
3. Based on recalibration of the instruments every 2 years (annualized).
4. Assumes repair of one instrument every year, 1 hour per repair.
5. Assumes repair of one cask every year, 1 hour per repair

TABLE 7.4-3

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TABLE 7.4-4

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TABLE 7.4-6

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FIGURE 7.4-1

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SECTION 7A

TN-40 CASK DOSE ANALYSIS

7A.1 SHIELDING DESIGN FEATURES

Shielding for the TN-40 cask is provided mainly by the thick-walled cask body. For neutron shielding, a borated polyester resin compound surrounds the cask body and a polypropylene disk covers the lid. Additional shielding is provided by the steel shell surrounding the resin layer and by the steel and aluminum structure of the basket.

Geometric attenuation, enhanced by attenuation by air and ground, and the earth filled berm provide additional shielding for distant locations at restricted area and site boundaries. Figure 7A-1 shows the configuration of shielding in the cask. Table 7A-1 lists the compositions of the shielding materials.

7A.2 SHIELDING ANALYSES

Shielding calculations for the primary gamma source are performed with the QAD-CGGP computer code (Reference 1). Neutron and capture gamma calculations are performed with the SCALE modules XSDRNPM and XSDOSE with the coupled SCALE 27n-18g cross section library (Reference 2).

A single QAD model was used for the primary gamma calculations of the top, bottom and side. This model is shown in Figure 7A-2.

Four QAD runs are performed, each utilizing one of the four source areas listed in Table 7.2-4: fuel zone, plenum, upper fitting and lower fitting. The sources are uniformly homogenized over the cavity diameter and the appropriate length, as shown in Figure 7A-2. The fuel basket is also homogenized over the cavity diameter and fuel assembly length and mixed into each source region. The basket material density is reduced by 75% in the top, bottom and plenum source zones to estimate the reduced shielding effectiveness of the basket in the axial direction. The radial resin and aluminum boxes are homogenized into a single composition based on the mass of each component. The steel encased polypropylene disk is also homogenized. The materials input for the QAD model is listed in Table 7A-2.

For the neutron and capture gamma dose on the side of the cask, a cylindrical one-dimensional model is used in XSDRNPM, as shown in Figure 7A-3. The central fuel region is considered to consist of uranium dioxide. The fuel cladding and steel basket are included in the homogenized fuel region. The fuel region is modeled as a cylinder with the actual cavity diameter. Subsequent regions are cylindrical shells corresponding to actual dimensions. The source placed uniformly in the fuel region is the source described in Table 7.2-5. The total source in the cylindrical fuel region is the same as the total source in the actual fuel region.

For doses at the ends of the cask, one-dimensional plane geometry XSDRNPM models are used. The fuel region is assumed to consist of uranium dioxide, Zircaloy and steel basket, as in the cylindrical model described above. In the top end model the plenum and top end fitting are homogenized with 25% basket material density as used in the QAD model and placed above the fuel zone. A polypropylene/steel slab and a steel layer representing the protective cover are placed over the lid. The bottom end model is similar. Both configurations are shown in Figure 7A-4. Atom densities of the materials used in the XSDRNPM calculations are listed in Table 7A-3. A conservative assumption is that the axial distribution of the source is taken as uniform, while in reality, the source will be relatively low near the top and bottom of the fuel region. This is because power shape during operation is non-uniform. The angular fluxes produced by XSDRNPM are processed by XSDOSE to produce dose rates at selected points.

To evaluate doses at long distances from the cask (long by comparison to either radius or height of the cask), a spherical geometry XSDRNPM model is used. This type of model is appropriate because the source region appears to be small and central when viewed from a great distance. This reasoning is confirmed by examination of the analytical solution for dose from a shielded line source of finite height (Reference 3). At large distances, the analytical solution is equivalent to the analytical solution for a point source having the same total source as the line source (line source strength times height of line) and the same thickness of shield. The fuel region of the XSDRNPM model is taken as a sphere of radius such that the volume of the sphere is equal to the volume of the cask cavity. The total primary source contained in the sphere is that listed in Tables 7.2-4 and 7.2-5. Layers of other materials - resin, steel shells - reproduce thicknesses of materials in the actual configuration, as shown in Figure 7A-5.

The spherical geometry model does not account for the presence of the ground. At very large distances from a source, characteristic of the distance to the nominal site boundary, the dose should be reduced by a factor of two because of attenuation by the ground (References 3 and 4). The XSDRNPM model with this correction still is conservative in that attenuation in non-uniform terrain (e.g., hills), natural obstacles (e.g., trees) and manmade structures is neglected. At moderate distances characteristic of the distance to the nominal restricted area boundary, however, the spherical geometry XSDRNPM model should apply without correction because of a balance between attenuation and back-scattering by the ground. Air attenuation is ignored in using XSDOSE to determine the dose rates.

As will be seen below, this modeling, when applied, indicates satisfaction of all regulatory requirements for radiation protection by substantial margins. Side and end contact dose rates and dose rates at short distances are given in Table 7A-4.

These dose rates are modest in view of the short duration of the tasks required in handling of the storage casks. Occupational doses are well within the requirements of 10CFR20 (Reference 5) and are consistent with the 10CFR72 (Reference 6) ALARA guideline. Dose rates at long distances for a single cask are shown in Figure 7A-6.

The models described above pertain to an individual storage cask. The evaluation of the ISFSI dose rates, i.e., from the array of casks, is contained in Appendix A7A 01244873

7A.3 DIRECT RADIATION N-S

The evaluation of the ISFSI dose rates, i.e., from the array of casks, is contained in Appendix A7A. 01244873

7A.4 DIRECT RADIATION E-W

The evaluation of the ISFSI dose rates, i.e., from the array of casks, is contained in Appendix A7A. 01244873

7A.5 INDIRECT RADIATION (SKYSHINE)

The evaluation of the ISFSI dose rates, i.e., from the array of casks, is contained in Appendix A7A. 01244873

7A.6 EARTH BERM

The evaluation of the ISFSI dose rates, i.e., from the array of casks, is contained in Appendix A7A. 01244873

7A.7 DOSE RATE AROUND THE ISFSI

The evaluation of the ISFSI dose rates, i.e., from the array of casks, is contained in Appendix A7A. 01244873

7A.8 EXPERIMENTAL RESULTS

The calculated dose rates for the TN-40 cask are essentially the same as for the TN-24 storage cask, which has a previously approved TSAR. A TN-24 prototype cask (TN-24P) has been fabricated and loaded with 24 Westinghouse 15x15 fuel assemblies. The assemblies had about four years cooling time with a burnup of around 30,000 MWD/MTU. Thermal and shielding tests were conducted at INEL on this loaded cask. The results are detailed in EPRI Report NP-5128 (Reference 8). The measured contact dose rates (mrem/hr) at mid height radially and at the center of the steel lid and bottom (no polypropylene disks or protective cover) were: 17g, 3n; 52g, 30n; and 145g, 90n respectively. At one meter from the surface they became, 10g, 2n; 23g, 12n; and 55g, 25n respectively. Although a direct comparison with the calculated values is difficult because of the different fuel parameters, it can be seen that the dose rates around the TN-24P cask are within expected values and therefore, the expected dose rates around the TN-40 should also compare favorably with the calculated values.

7A.9 REFERENCES

1. "QAD-CGGP - A combinatorial Geometry Version of QAD-P5A, a Point Kernel Code System for Neutron and Gamma-Ray Shielding Calculations Using the GP Buildup Factor," CCC-493, Oak Ridge National Laboratory, Oak Ridge, Tennessee, September 1986.

2. "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," ORNL/NUREG/CR-0200, U.S. Nuclear Regulatory Commission, Revision 3, December 1984.

3. Jaeger, R. G., et. al., Engineering Compendium on Radiation Shielding, Springer-Verlag, New York, 1968.

4. Schaeffer, N. M., Reactor Shielding for Nuclear Engineers, TID-25951, U.S. Atomic Energy Commission, Washington, D.C., 1983.

5. "Standards for Protection Against Radiation," 10CFR Part 20, Rules and Regulations, Title 10, Chapter 1, Code of Federal Regulations - Energy, U.S. Nuclear Regulatory Commission, Washington D.C., January 1987.

6. "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation (ISFSI)," 10CFR Part 72, Rules and Regulations, Title 10, Chapter 1, Code of Federal Regulations - Energy, U.S. Nuclear Regulatory Commission, Washington D.C., January 1987.

7. "SKYSHINE II - Calculation of the Effects of Structure Design on Neutron Primary Gamma-Ray Dose Rates in Air," CCC-289, Oak Ridge National Laboratory, Oak Ridge, Tennessee, March 1982.

8. "The TN-24P PWR Spent-Fuel Storage Cask: Testing and Analysis," EPRI NP-5128, Electric Power Research Institute, Palo Alto, California, April 1987.

TABLE 7A-1

TN-40 CASK SHIELD MATERIALS

COMPONENT	MATERIAL	DENSITY (g/cm^3)	THICKNESS (in.)
Cask Body Wall	Carbon Steel	7.85	9.50
Lid	Carbon Steel	7.85	10.50
Bottom	Carbon Steel	7.85	10.25
Resin	Polyester Resin		
	Styrene		
	Aluminum Hydrate		
	Zinc Borate	1.58	4.26
Aluminum Box	Aluminum	2.7	0.12
Outer Shell	Carbon Steel	7.82	0.50
Basket	Stainless Steel	7.80	2 x 0.10
	Aluminum	2.7	2 x 0.25
	Boral	2.65	0.075
Protective Cover	Carbon Steel	7.85	0.38
Polypropylene Drum	Polypropylene	0.90	4.0
Polypropylene Drum Shell	Carbon Steel	7.85	0.25

TABLE 7A-2

MATERIALS INPUT FOR QAD MODEL (TN-40 CASK)

ZONE	ELEMENT	DENSITY g/cm^3
Fuel Basket	Fe	0.248
	Zr	0.349
	U	1.58
	Al	0.261
Plenum/Basket	Fe	0.062
	Zr	0.378
	Al	0.065
Top Fitting/Basket	Fe	0.802
	Al	0.065
Bottom Fitting/Basket	Fe	1.12
	Al	0.065
Body, Cover, Shell	Fe	7.85
Polypropylene/Steel	C	0.68
	Fe	0.86
Resin/Aluminum	C	0.50
	O	0.59
	Al	0.48

TABLE 7A-3

MATERIALS INPUT FOR XSDRNPM (TN-40 CASK)

ZONE	ELEMENT/ NUCLIDE	LIBRARY NUMBER	DENSITY atom/b.cm
Fuel Basket	U238	92238	3.84E-3
	U235	92235	1.56E-4
	O	8016	8.00E-3
	Zr	40000	2.30E-3
	Fe	26000	2.67E-3
	Al	13027	5.83E-3
Plenum/Top	Fe	26000	9.32E-3
Fitting Basket	Zr	40000	2.50E-3
	Al	13027	2.90E-3
Bottom Fitting/Basket	Fe	26000	1.20E-2
	Al	13027	1.45E-3
Body, Cover, Shell	Fe	26000	8.46E-2
Polypropylene/Steel	Fe	26000	9.27E-3
	C	6012	3.41E-2
	H	1001	7.23E-2
Resin/Aluminum	Al	13027	1.08E-2
	C	6012	2.50E-2
	O	8016	2.23E-2
	H	1001	4.32E-2
	Bl0	5010	1.64E-4

TABLE 7A-4

TN-40 DOSE RATES AT SHORT DISTANCES

LOCATION	DOSE RATE (mrem/hr)		
	GAMMA	NEUTRON	TOTAL
Radial			
Contact	45.1	12.4	57.5
1 m	24.8	5.2	30.0
2 m	16.8	2.9	19.7
3 m	12.0	1.8	13.8
2 m(3)	73.5	160.0	233.5
Top(1)			
Contact	23.0	2.6	25.6
2 m	5.9	0.5	6.4
Top(2)			
Contact	40.8	891	932
Bottom			
Contact	79.8	1195	1275

(1) With polypropylene disc and protective cover in place.

(2) Without polypropylene disc or protective cover.

(3) Without neutron shield and outer shell.

TABLE 7A-5

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TABLE 7A-5a

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TABLE 7A-5b

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TABLE 7A-6

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TABLE 7A-7

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TABLE 7A-8

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Figure Withheld Under 10 CFR 2.390

Figure 7A-1

TN-40 CASK SHIELDING CONFIGURATION

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

TN-40 CASK SHIELDING CONFIGURATION

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG7A-1_REV_3
	PAGE. NO. NA	DATE: 04-25-06	

Figure Withheld Under 10 CFR 2.390

Figure 7A-2

QAD MODEL (TN-40 CASK)

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

QAD MODEL (TN-40 CASK)			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG7A-2_REV_3
	PAGE. NO. NA	DATE: 04-25-06	

Figure Withheld Under 10 CFR 2.390

Figure 7A-3

XSDRN-PM RADIAL MODEL (TN-40 CASK)

PRAIRIE ISLAND ISFSI SAFETY ANALYSIS REPORT

XSDRN-PM RADIAL MODEL (TN-40 CASK)

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: **VLS**	REVISION: 0	FIG7A-3_REV_0
	PAGE. NO. **NA**	DATE: 04-25-06	

Figure Withheld Under 10 CFR 2.390

Figure 7A-4

XSDRN-PM AXIAL MODELS (TN-40 CASK)

PRAIRIE ISLAND ISFSI SAFETY ANALYSIS REPORT

XSDRN-PM AXIAL MODEL (TN-40 CASK)

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG7A-4_REV_0
	PAGE. NO. NA	DATE: 04-25-06	

Figure Withheld Under 10 CFR 2.390

Figure 7A-5

XSDRN-PM TN-40 SPHERICAL MODEL (LONG DISTANCE)

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

XSDRN-PM TN-40 SPHERICAL MODEL (LONG DISTANCE)			
NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: **VLS**	REVISION: 0	FIG7A-5_REV_0
	PAGE. NO. **NA**	DATE: 04-25-06	

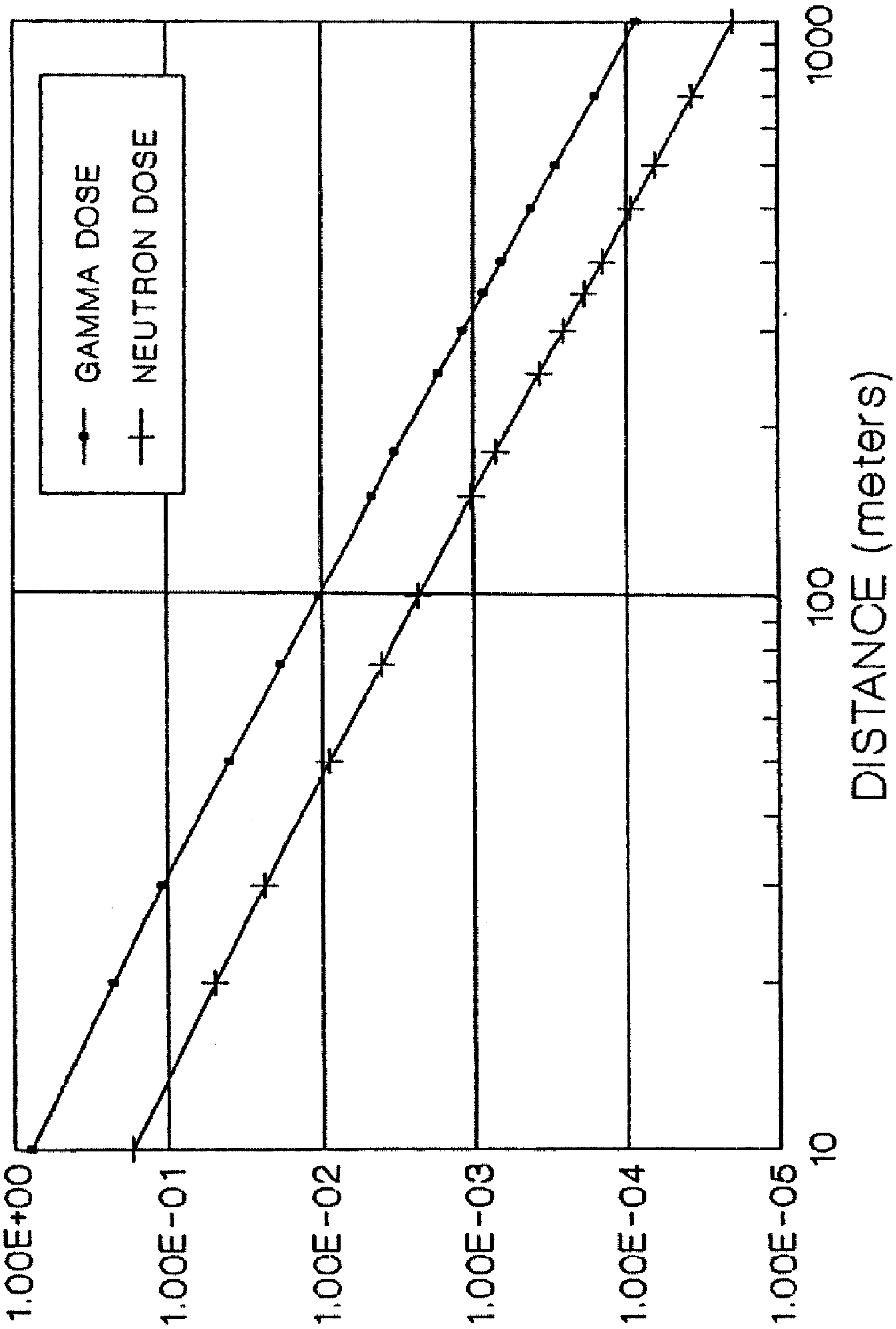

Figure 7A-6
DOSE RATES AT LONG DISTANCES (MREM/HR)
PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

Rev. 2 9/91

DOSE RATES AT LONG DISTANCES (MREM/HR)

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: KJF PAGE. NO. NA	REVISION: 2 DATE: 9-22-11	FIG7A-6_REV_2

FIGURE 7A-7

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FIGURE 7A-8, REV. 2

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FIGURE 7A-9, REV. 2

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FIGURE 7A-10

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FIGURE 7A-11

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APPENDIX 7B

SHIELDING EVALUATION COMPUTER INPUT

PROPRIETARY

PROPRIETARY - TRADE SECRET INFORMATION WITHHELD PURSUANT TO 10 CFR 2.390

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SECTION 8

ACCIDENT ANALYSIS

8.1 OFF-NORMAL OPERATIONS

Off-normal operations are design events of the second type (Design Event II) as defined in ANSI/ANS 57.9 (Reference 1). Design Event II consists of that set of events that, although not occurring regularly, can be expected to occur with moderate frequency or on the order of once during a calendar year of ISFSI operation.

Of the various types of Design Event II conditions described in Reference 1, only the loss of external power supply for a limited duration is considered to be applicable and credible to ISFSI operations. Loss of electric power as an off-normal condition is analyzed below.

8.1.1 LOSS OF ELECTRICAL POWER

A total loss of ac power is postulated to occur in the feeder cabling which supplies power to the ISFSI. The failure could be either an open or a short to ground circuit, or any other mechanism capable of producing an interruption of power.

8.1.1.1 POSTULATED CAUSE OF THE EVENT

A loss of power to the ISFSI may occur as a result of natural phenomena, such as lightning or extreme wind, or as a result of undefined disturbances in the nonsafety-related portion of the electric power system of the Prairie Island Nuclear Generating Plant.

If electric power is lost, the following systems would be de-energized and rendered nonfunctional:

- Area lighting
- Area receptacles
- Cask pressure monitoring instrumentation

8.1.1.2 DETECTION OF EVENTS

A loss of ac power at the Prairie Island Nuclear Generating Plant site would be indicated and/or alarmed in the main control room. If the loss of power were localized solely at the ISFSI, this would be detected during periodic surveillance by noting that area lighting is not operational.

8.1.1.3 ANALYSIS OF EFFECTS AND CONSEQUENCES

This event has no safety or radiological consequences because a loss of power will not affect the integrity of the storage casks, jeopardize the safe storage of the fuel, or result in radiological releases. None of the systems whose failure could be caused by this event are necessary for the accomplishment of the safety function of the ISFSI. The lighting system functions merely for convenience and visual monitoring, and the instrumentation monitors the long-term performance of the storage casks with respect to the cask seals. None of these parameters are expected to change rapidly and their status is not dependent upon electric power.

8.1.1.4 CORRECTIVE ACTION

Following a loss of electric power to the ISFSI, plant maintenance personnel will be informed and will isolate the fault and restore service by conventional means. Such an operation is straightforward and routine for the maintenance personnel of an electric utility.

8.1.2 RADIOLOGICAL IMPACT FROM OFF-NORMAL OPERATIONS

No radiological impact from off-normal operations is postulated.

8.2 ACCIDENTS

Accidents are design events of the third and fourth type (Design Events III and IV) as defined in ANSI/ANS 57.9. Design Event III consists of that set of infrequent events that could reasonably be expected to occur during the lifetime of the ISFSI.

Design Event IV consists of the events that are postulated because their consequences may result in the maximum potential impact on the immediate environs. Their consideration establishes a conservative design basis for certain systems with important confinement features.

8.2.1 EARTHQUAKE

8.2.1.1 CAUSE OF ACCIDENT

The design earthquake (DE) is postulated to occur as a design basis extreme natural phenomenon.

8.2.1.2 ACCIDENT ANALYSIS

Seismic response characteristics of the storage casks are provided in Section 3.2.3 and 4.2.3. Results of these analyses show that cask leak-tight integrity is not compromised and that no damage will be sustained.

8.2.1.3 ACCIDENT DOSE CALCULATIONS

The DE is not capable of damaging the cask. Hence, no radioactivity is released and there is no associated dose.

8.2.2 EXTREME WIND

8.2.2.1 CAUSE OF ACCIDENT

The extreme winds due to passage of the design tornado as defined in Section 3.2.1 are postulated to occur as an extreme natural phenomenon.

8.2.2.2 ACCIDENT ANALYSIS

The effects and consequences of extreme winds on the casks are presented in Section 3.2.1.

8.2.2.3 ACCIDENT DOSE CALCULATIONS

Extreme winds are not capable of overturning these casks nor of damaging their seals. Since no radioactivity is released, no resultant doses will occur.

Local damage to the neutron shield may be caused by tornado missiles. However, Table 7A-4 shows that the dose rate without any shield to be less than the allowable accident dose rate.

8.2.3 FLOOD

8.2.3.1 CAUSE OF ACCIDENT

The probable maximum flood has been calculated to reach a level of 703.6 ft., with wave action to a maximum level of 706.7 ft.

8.2.3.2 ACCIDENT ANALYSIS

The casks are designed to withstand the forces developed by the probable maximum flood without damage to cask integrity or tipping of the casks. The height of the cask seals will be above the level of the probable maximum flood and associated wave action. Accordingly no fuel damage or criticality is postulated to occur as a result of flooding. Analyses are contained in Section 3.2.2.

8.2.3.3 ACCIDENT DOSE CALCULATIONS

The probable maximum flood is not capable of overturning the casks or of damaging their seals. Therefore, no resultant doses are projected.

8.2.4 EXPLOSION

8.2.4.1 CAUSE OF ACCIDENT

A munition barge explosion has been postulated to occur at a location approximately 2600 feet from the ISFSI. This occurrence is described in detail in Section 2.2. The pressure wave of 2.25 psi is estimated to occur at the ISFSI.

8.2.4.2 ACCIDENT ANALYSIS

The cask accident analysis includes the consideration for a 2.25 psi overpressure from the postulated explosion near the ISFSI location, as described in Section 3.2.5.3.4. 01106798

8.2.4.3 ACCIDENT DOSE CALCULATIONS

The cask will not tip as a result of the postulated pressure wave. Accordingly, no cask damage or release of radioactivity is postulated. Since no radioactivity is released, no resultant doses would occur.

8.2.5 FIRE

8.2.5.1 CAUSE OF ACCIDENT

The only combustible materials in the ISFSI are in the form of insulation on instrumentation wiring, and paint on the outside surface of the storage casks. In addition, the tow vehicle will contain a small amount of gasoline or diesel fuel. No other combustible or explosive materials are allowed to be stored on the ISFSI slabs. The ISFSI area is cleared of trees. The entire area surrounding the Equipment Storage Building and concrete pad within the perimeter road is covered with crushed rock. In addition, other equipment in the area are adequately separated from the ISFSI slabs. Therefore, no fires other than small electrical fires are considered credible at the ISFSI.

8.2.5.2 ACCIDENT ANALYSIS

The ability of the cask to withstand postulated fires is addressed in Section 3.2.5.2.9 and 3.3.2.2.2.

8.2.5.3 ACCIDENT DOSE CALCULATIONS

Since no activity is released, no resultant doses would occur.

8.2.6 INADVERTENT LOADING OF A NEWLY DISCHARGED FUEL ASSEMBLY

8.2.6.1 CAUSE OF ACCIDENT

The possibility of a spent fuel assembly, with a heat generation rate greater than 0.675 kw, being erroneously selected for storage in a cask has been considered. The cause of this accident is postulated to be an error during the loading operations, e.g., wrong assembly picked by the fuel handling crane, or a failure in the administrative controls governing the fuel handling operations.

8.2.6.2 ACCIDENT ANALYSIS

The fuel assemblies require several years of storage in the spent fuel pool before the heat generation decays to a rate below 0.675 kw. This accident scenario postulates the inadvertent loading of an assembly not intended for storage in the storage canister, with a heat generation rate in excess of the design basis specified in Section 3.1.1 and 10.1.1.

In order to preclude this accident from going undetected, and to ensure that appropriate rectification actions can take place prior to the sealing of the casks, a final verification of the assemblies loaded into the casks and a comparison with fuel management records will be performed to ensure that the loaded assemblies do not exceed any of the specified limits.

These administrative controls and the records associated with them will be included in the procedures described in Chapter 9 and in the proposed license requirements described in Chapter 10, and will comply with the applicable requirements of the Quality Assurance Program described in Chapter 11.

Therefore, appropriate and sufficient actions will be taken to ensure that an erroneously loaded fuel assembly does not remain undetected. In particular, the storage of a fuel assembly with a heat generation in excess of 0.675 kw is not considered credible in view of the multiple administrative controls.

8.2.6.3 ACCIDENT DOSE CALCULATIONS

The inadvertent loading of a fuel assembly not intended for storage in a storage cask is not considered to be a credible occurrence. Therefore, no doses are postulated.

8.2.7 CASK SEAL LEAKAGE

The storage casks feature redundant seals in conjunction with an extremely rugged body design. Additional barriers to the release of radioactivity are presented by the sintered fuel pellet matrix and the Zircaloy cladding which surrounds the fuel pellets. Furthermore, the interseal gaps are pressurized in excess of the cask cavity. As a result, no credible mechanisms that could result in leakage of radioactive products have been identified. Nevertheless, a complete loss of the storage cask confinement capability is postulated in Section 8.2.9, and the results found to be negligible.

8.2.8 HYPOTHETICAL CASK DROP ACCIDENT

8.2.8.1 CAUSE OF ACCIDENT

The stability of the TN-40 storage cask in the upright position on the ISFSI concrete storage pad is demonstrated in Section 3.2 of this SAR. The effects of tornado wind and missiles, flood water and earthquakes are described in Sections 3.2.1, 3.2.2 and 3.2.3, respectively. It is shown in those sections that the cask will not tip over under the most severe natural phenomena specified in the Prairie Island Updated Safety Analysis Report. Also the cask will not slide on its pad any more than about one inch under any of these loadings.

The cask will be lifted at Prairie Island using a single failure proof crane. The trunnions are designed to the requirements of ANSI N14.6 (Reference 2) for lifting devices for critical loads with increased stress design factors. Safety factors of 6 against the trunnion yield strength and 10 against the ultimate strength are provided. In addition, the cask will be handled by the transport vehicle. The cask will always be in a vertical orientation and never lifted higher than 18 in. Therefore it is extremely unlikely that the cask could be dropped.

However, this section of the SAR considers design events of the third and fourth types (includes accidents) as defined in ANSI/ANS-57.9. The third type of events are those that could reasonably be expected to occur over the lifetime of the ISFSI (does not include dropping of the cask). The fourth type of events include severe natural phenomena (described in Section 8.2.1 through 8.2.5) and man induced low probability events postulated because their consequences could result in the maximum potential impact on the immediate environs. Therefore the cask is examined for a dropping accident, which is a hypothetical impact event that is extremely unlikely to occur.

8.2.8.2 ACCIDENT ANALYSIS

In this section the cask is evaluated under bottom end impact on the ISFSI storage pad after a drop from a height of 18 in. The storage pad is the hardest concrete surface outside of the containment building. The cask is always oriented vertically and is never lifted higher than 18 in. once it leaves the containment building. Therefore this case is an upper bound drop event since impact onto a softer surface would result in lower cask deceleration and a lower impact force.

8.2.8.2.1 DYNAMIC IMPACT LOADS

The impact analysis is based on the methodology of EPRI NP-4830 (Reference 3). This report considers the mass and geometry of the cask but assumes it to be rigid compared to the concrete storage pad. The storage pad properties and the cask geometry are used to determine the pad hardness parameter. The report provides graphs that show the force on the cask as a function of storage pad hardness.

TARGET HARDNESS

The target (or storage pad) hardness parameter, S, for the end drop case is:

$$S_{end} = \frac{2}{w^3} \frac{rAkM_u\sigma_u}{\left(1 - e^{-\beta r} \cos \beta r\right)} = 95{,}680 \quad \text{(formula 4, reference 3)}$$

Where:

r = cask bottom radius = 45.5 in.

A = bottom area = πr^2 = 6,504 in.^{2}

$$k = \frac{\pi E}{1 - {v_s}^2} s = 118{,}200 \text{ psi/in.}$$

E_s = Soil modulus = 30,000 psi

v_s = Poisson's ratio of soil = 0.45

M_u = *is based on pad thickness of 36 in., #14 rebar @ 12 in. spacing (nominal), S_y rebar of 60,000 psi, 2 in. cover (nominal), σ_u of 4,000 psi max. concrete compressive strength*

$= 4.25 \times 10^6$ *in.lb/ft.*

$\beta = 0.02686$

W = 240,000 lb. (slightly low weight gives conservatively high hardness)

DECELERATIONS

Figure 22 for a 20 in. drop height from EPRI NP-4830 (Reference 3) can be conservatively used to determine the cask deceleration after the 18 in. end drop. The upper bound deceleration is 40 g's for a hardness parameter, S, of 95,680. The maximum impact force is then 40 times the weight of the cask. The TN-40 end impact stress analyses below are conservatively performed for a deceleration of 50 g's.

8.2.8.2.2 CASK BODY ANALYSIS

The analysis results for the hypothetical cask drop accident are reported in this section as the 18 in. bottom end drop onto the storage pad. As explained in Section 8.2.8.1, this accident has a very low probability, but in view of its potential impact on the environs, a detailed analysis was performed.

A conservative 50 g bottom drop onto the concrete pad was analyzed. The ANSYS model in Section 4A.3.2 was used to evaluate the stresses in the cask body due to the 50 g bottom drop with the following modifications:

1. Nodes 44-200 and 59-201 (See Figure 8.2-1C) are also coupled. They are coupled in the axial direction but are allowed to slide over each other in the radial and hoop directions.

2. All nodes on the outside bottom surface of the cask are fixed in the axial direction.

3. The internal loading effects are simulated by distributed pressure acting on the inside bottom surface of the cavity, as opposed to nodal forces.

4. A distributed inertial force of 50 g's was applied on each finite element in the model (See Figure 8.2-1D).

Stress results for this individual load case are reported in Figure 8.2-1A. Figure 4A.3-12 shows the locations on the cask body where stress results are reported. This case is combined with the stress results for bolt preload and internal pressure of 100 psi in Figure 8.2-1B. This case will be evaluated against the cask body criteria for a Level D event below.

8.2.8.2.3 LID BOLT ANALYSIS

The lid bolts are analyzed in this section under the loadings selected to bound those for the hypothetical bottom end drop onto the concrete storage pad.

BOTTOM END DROP

The bottom end drop from a height of 18 in. onto the concrete storage pad is analyzed above in Section 8.2.8.2.1. That section indicates that the cask deceleration may reach 40 g. This analysis conservatively examines the effects (if any) of a 50 g quasistatic loading on the lid bolts.

During a bottom end drop, the rim of the lid is forced against the flange of the cask body. The lid is initially seated against the flange by preloading (torquing) the bolts. The bolt preload will not be affected if compressive yielding of the contact bearing area does not occur.

The contact force on the bearing area, conservatively neglecting internal pressure, is the bolt preload force less the seal compression force plus the 50 g inertial force of the lid system. The preload force, from Section 4A.4.1, is 1,790,400 lb.[1] The seal seating force is 1,037,164 lb. The weight of the lid system (including shield plate and resin disk) is 15,397 lb.

Therefore, during a 50 g deceleration in the axial direction the contact force between lid and cask body is:

$$
\begin{aligned}
F_{contact} &= F_{Bolt\ Preload} - F_{seal\ seating} + 50\ (W_{lid\ system}) \\
&= 1{,}790{,}400 - 1{,}037{,}164 + 50(15{,}397) \\
&= 1{,}523{,}086\ lb.
\end{aligned}
$$

Figure 8.2-7A illustrates the bearing interface between lid edge and body flange. The bearing area equals the area within the diameter of the lid raised section (77.25 in.) less that outside of the body chamber (73.29 in.) less the area of the seal groove.

$$A_{bearing} = \frac{\pi}{4}\left(77.25^2 - 76.2^2 + 74.00^2 - 73.29^2\right)$$

$$= 208.7\ in.^2$$

The bearing stress during impact is then equal to:

$$S_{bearing} = \frac{F_{contact}}{A_{bearing}} = \frac{1{,}523{,}086\ lb.}{208.7\ in.^2} = 7{,}229\ psi$$

[1]Based on 25,000 psi bolt preload stress (conservative)

This contact stress is well below the 37,500 psi yield stress of the lid and flange material. The bolt preload will not be affected by the bottom drop. Therefore, this hypothetical accident case will not affect the bolts.

8.2.8.2.4 BASKET ANALYSIS

BOTTOM END DROP

The basket analysis is presented in detail in Appendix 4B. The analysis of the basket under vertical loading is found in Section 4B.7. The fuel assemblies react directly against the bottom of the cask in the vertical load case. They do not load the basket as in the side impact case. The fuel assemblies themselves can withstand more than 80 g as indicated in Reference 6.

The Appendix 4B analysis performed for the hypothetical 50g bottom end drop onto the concrete pad does not take credit for the aluminum strength. It is conservatively assumed that all of the load is taken by the 304 stainless steel. Therefore:

$$\sigma = \frac{Total\ Compressive\ Load}{Cross\ Section\ of\ \ 304SS}$$

On a single wall panel the stress calculated in Section 4B.7 for a 3g load would simply be 50/3 X 304 = 5067 psi. This 5067 psi compressive stress is acceptable since the Section 4B.5 Level D membrane stress intensity limit for 304 stainless steel is 44,900 psi at 400°F (the approximate maximum temperature at the end of the basket). In addition, axial compression tests of unsupported fuel compartment box sections described in Section 4C.2 show that the compressive stress can reach 23,000 psi before failure (even without the stiffening effects of the aluminum plates). Therefore buckling will not occur.

SHEAR STRESS IN ½ IN. FUSION PLUG WELDS

This section describes the analysis of the shear stresses at the basket plug welds due to the combination of differential thermal expansion between aluminum and stainless steel and the bottom end drop.

The thermal analysis of the basket is described in Section 3.3.2.2. The analysis determines the basket temperatures for the condition with maximum solar heating, maximum decay heat from the cask contents, and 100°F ambient air temperature. The basket temperatures were used directly in the ANSYS structural models to calculate the basket panel stresses due to differential thermal expansion. Stresses occur due to differences between the coefficients of thermal expansion of the 304 stainless, aluminum and boral (See Section 4B for detailed ANSYS model descriptions). In order to calculate the maximum shear stresses at the 1/2 in. plug welds, it was conservatively assumed that the 1.38 in. diameter stainless plugs that penetrate the 1.5 in. diameter hole in the aluminum (and boral) plates are not centered. The plugs were assumed to be in contact initially (at 70°F) with the opposing sides of the two holes in the aluminum (the sides toward the center of the panel) so that the maximum interference of aluminum and steel will occur when the panel is heated. In this worst plug misalignment case with the highest temperature of 530°F, seen by any portion of the basket, the weld shear stress could reach a maximum of 25,434 psi as shown on Figure 4B.6-6.

A full length compartment wall (160 in. long) with a span length of 8.05 in. is evaluated for shear stresses at the 1/2 in. plug welds due to a 50 g end drop.

Size of weld = 0.5 dia.

$$\text{Number of welds} = 2 \times 2 \times \frac{160}{8} = 80$$

(each 8" spacing has two rows of plugs, each plug has two welds, one on each side)

$$\text{Shear area} = \pi/4\,(0.5)^2 \times 80 = 15.71\ in^2$$

$$\text{Weight of aluminum} = 2 \times 8.05 \times 0.25 \times 160 \times 0.105 = 67.62\ lbs.$$

$$\text{Weight of boral} = 1 \times 7.50 \times 0.075 \times 160 \times 0.0903 = 8.13\ lbs.$$

Total weight of aluminum and boral = 75.75 lbs.

Assuming the 50 g compressive load is uniformly distributed to all of the 80 welds, the shear stress is

$$\tau = \frac{75.75 \times 50}{15.71} = 241\ psi$$

Appendix F to Section III of the ASME Code (Reference 4) provides a basic 0.42 S_u limit on the average primary shear stress across a section loaded in pure shear for Level D conditions. The combined shear stress in the weld due to differential thermal expansion and the end drop accident is 25,434 + 241 = 25,675 psi which is below the limit of 26,670 psi (0.42 S_u) based on the temperature of 530°F at the worst panel location.

Based on the results of this analysis, it is concluded that the basket is structurally adequate for withstanding the combined loads due to thermal expansion and the 18 inch end drop accident, and will properly support and position the fuel assemblies.

BASKET WELD MATERIAL TOLERANCE

The 18 inch cask end drop is the most limiting postulated event in terms of structural analysis. Achieving a tight tolerance between the aluminum plates and the stainless steel boxes at the bottom edge of the basket was considered. If the aluminum plates are recessed, the stainless steel boxes must support the load due to the end drop. In order to evaluate this possibility, an analysis was performed.

The basket was analyzed first assuming that the entire load is supported by the stainless steel, and then analyzed assuming that the entire load is supported by the aluminum.

The basket is composed of stainless steel boxes, aluminum plates and boral plates held together by plug welds every 8 inches vertically. A cross section of the basket wall is shown in Figure 4B.7-1.

The stainless steel is analyzed as a box section with a thickness of 0.10 inches and inside dimensions of 8.05 inches square. The highest loading occurs at the base of the basket, which is analyzed at a temperature of 300°F. The basket segments below the bottom welds will be stressed the most, since they are taking the entire weight of the basket. The yield stress of Type 304 stainless steel at 300°F is 22,500 psi.

The area of the box section is:

$$A = 8.25^2 - 8.05^2 = 3.26 \text{ in.}^2$$

The radius of gyration is:

$$r = 0.408d = 0.408(8.15) = 3.33 \text{ in.}$$

Assuming hinged ends, the slenderness ratio is:

$$\frac{kl}{r} = \frac{(1)(8.0)}{(3.33)} = 2.4$$

From Section NF-3322.1, equation 6a and Table NF-3523(b)1 of Reference 4, the critical stress is:

$$F_a = 1.5S_y\left[0.47 - (kl/r)/444\right]$$

$$= 1.5(22,500)\left(0.47 - \frac{2.4}{444}\right)$$

$$= 15,680 \text{ psi for level C service conditions.}$$

Level C service conditions are more restrictive than level D accident conditions but are used because Appendix F for level D conditions gives no guidance for austenitic stainless steel.

The critical load applied to the box section is:

$$Pc = F_a \times A = 15{,}680 \times 3.26 = 51{,}116 \text{ lbs.}$$

The weight of the basket per box section is approximately 320 lbs. Therefore the maximum g load sustainable by the stainless steel is 51,116/320 = 160 g's.

This result is conservative as shown by the tests on the bare boxes presented in Section 4C. The test showed that a stress of 23,000 psi was sustainable at room temperature. This is roughly equivalent to 17,000 psi at 300°F, which results in a sustainable g load of 176 g's.

The condition where the aluminum panels are supporting the load was evaluated by assuming a stable cross section with each member bending about a common neutral axis as shown in Figure 4B.5-6. The allowable stresses were taken from Reference 7.

$$A = 0.5 \text{ in}^2/\text{in}$$

$$r = 0.1443 \text{ in.}$$

Assuming hinged ends,

$$l = 8 \text{ in.}$$

$$k = 1.0$$

$$\frac{kl}{r} = 55.44$$

If T = 300°F, the allowable axial compression stress is:

$F_c = 15.8 - 0.096\ kl/r = 10.48$ *ksi for cross sections farther than 1.0 inch from any weld, and*

$F_c = 9.5$ *ksi for cross sections within 1.0 inch of a weld.*

The critical load is $9{,}500 \text{ psi} \times 0.5 = 4{,}750 \text{ lb/in.}$

The critical load per panel is $4{,}750 \times 8.05 = 38{,}237 \text{ lbs.}$

This is equivalent to a g load of $\frac{38{,}237}{153} = 250\ g's$

Based on this analysis, the stainless steel members can withstand a 160 g impact load of the entire basket during the end drop. The aluminum members can withstand a 250 g impact load due to the entire weight of the basket.

Based on the analysis in Section 8.2.8.2.2 showing that the maximum g load during the end drop is less than 50 g's, the stainless steel and the aluminum panels can each withstand the end drop loading separately. Therefore the tolerance on the lengths of the steel boxes and aluminum panels is not critical.

8.2.8.3 ACCIDENT DOSE CALCULATIONS

Cask drop will not breach the cask confinement barrier. No radioactivity will be released and no resultant doses will occur. Table 7A-4 shows the calculated dose rate assuming the neutron shield and outer shell are removed.

8.2.9 LOSS OF CONFINEMENT BARRIER

8.2.9.1 CAUSE OF ACCIDENT

The following postulated accident scenario is not considered to be credible. It is hypothesized solely to demonstrate the inherent safety of the ISFSI by subjecting it to a set of simultaneous multiple failures, any one of which is far beyond the capability of natural phenomena or man-made hazards to produce. A simultaneous failure of all protective layers of confinement is postulated to occur by unspecified nonmechanistic means in the cask.

8.2.9.2 ACCIDENT ANALYSIS

In this accident, the confinement function is nonmechanistically removed. Heat removal and radiation shielding functions operate in the normal passive manner.

This is equivalent to breaking the cask seal barriers (no release), removing the closure lids (no release), failing all the cladding in all the loaded fuel assemblies (gap activity release), and finally, failing the fuel pellets themselves such that the remaining Kr-85 is released from the fuel matrix.

8.2.9.3 ACCIDENT DOSE CALCULATIONS

Table 7.2-7 lists the nuclides present in a cask containing 40 design basis fuel assemblies. The only nuclide listed in Table 7.2-7 which naturally occurs in the gaseous state, which could escape from the cask following a postulated breach of cask confinement barrier and which would be a significant dose contributor, is Kr-85.

All of the Kr-85 gas is conservatively assumed to be instantaneously released from the TN-40 cask. There is no additional decay of Kr-85 in transit from the spent fuel storage cask to the receptor and no credit is taken for personnel protection due to any structure or system.

The maximum individual is assumed to be located at the site boundary where the least amount of atmospheric dispersion takes place (largest χ/Q value). The dose results for this location are conservative for any individual (maximum) and may be reported as dose to an individual at the nearest site boundary.

The site boundary χ/Q values were calculated as described in Section 2.3.4 and shown on Table 2.3-1.

As shown on Table 7.2-7, there are 9.67E4 Curies of Kr-85 activity in the TN-40 spent fuel cask following 10 years of fuel decay. The gamma (whole body) dose conversion factor (K) for Kr-85 is 1.9 rem/hr per Ci/cu. meter. The atmospheric dispersion factors for ground level release were calculated as described in Section 2.3.4.

Dose evaluations were performed based on Regulatory Guide 1.3 (Reference 5) methodology and equations.

The offsite radiological consequences of a postulated loss of spent fuel cask confinement barrier for a cask located at the ISFSI are provided in Table 8.2-1 and plotted in Figure 8.2-33.

The nearest site boundary or maximum individual whole body dose for the loss of spent fuel cask confinement barrier is determined to be 0.15 rem. This dose is well within the 5 rem criteria given in 10CFR72.106(b)

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8.3 SITE CHARACTERISTICS AFFECTING SAFETY ANALYSIS

Site characteristics have been considered in the formation of the bases for these safety analyses. Conservative assumptions concerning meteorology were used in the determination of χ/Q. The characteristics of extreme winds and their contribution to maximum flood level were considered. Regional and site seismology and geology were used to help define the design earthquake acceleration value and analyze for liquefaction potential. Population distribution and other demographic data were used to determine radiation doses.

Other site characteristics affecting safety analyses include the proximity to the Mississippi River to the ISFSI assumptions concerning barge traffic was used to develop the bases for the explosion (Section 8.2.4).

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8.4 REFERENCES

1. American Nuclear Society, ANSI/ANS-57.9, Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type), 1984.

2. American National Standards Institute, ANSI N14.6, Special Lifting Devices for Shipping Containers Weighing 10,000 pounds or More, 1986.

3. Electric Power Research Institute, Report No. NP-4830, The Effects of Target Hardness on the Structural Design of Concrete Storage Pads for Spent-Fuel Casks, October 1986.

4. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, 1989.

5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.3, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors, Revision 2, June 1974.

6. Chun, R., Witte, M. and Schwartz, Dynamic Impact Effects on Spent Fuel Assemblies, Lawrence Livermore National Laboratory Report UCID-21246, Oct. 20, 1987.

7. American Society of Mechanical Engineers, ASME B96.1, Welded Aluminum Alloy Storage Tanks, 1989.

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TABLE 8.2-1

RADIOLOGICAL CONSEQUENCES – LOSS OF CONFINEMENT BARRIER

LOCATION	DISTANCE (MILES)	χ/Q (SEC/CU. METER)	DOSE FACTOR (REM-SEC PER CU. METER)	DOSE (REM)
Max. Individual*	0.068	6.63E-03	51.04	3.38E-01
Max. Individual**	0.11	2.95E-03	51.04	1.50E-01
1	0.5	2.23E-04	51.04	1.14E-02
2	1.5	7.45E-05	51.04	3.80E-03
3	2.5	4.54E-05	51.04	2.32E-03
4	3.5	3.24E-05	51.04	1.65E-03
5	4.5	2.50E-05	51.04	1.28E-03
6	7.5	1.46E-05	51.04	7.45E-04
7	15	6.84E-06	51.04	3.49E-04
8	25	4.35E-06	51.04	2.22E-04
9	35	3.94E-06	51.04	2.01E-04
10	45	3.67E-06	51.04	1.87E-04

*Located at nearest site boundary distance of 110 meters from the western edge of the ISFSI concrete pads.

**Located at nearest site boundary distance of 180 meters from a central point between the two ISFSI concrete pads.

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FIGURE 8.2-1, REV. 1

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LOCATION		MERIDIONAL STRESS (PSI) OUTER SIGBS	MERIDIONAL STRESS (PSI) INNER SIGTS	HOOP STRESS (PSI) OUTER SIGBTH	HOOP STRESS (PSI) INNER SIGTTH	STRESS INTENSITY SI (PSI)
CONTAINMENT VESSEL	1		1543		1016	2526
	2	-504		23		527
	3		-1910		-4	1906
	4	2954		1110		2954
	5		-1509		-42	1509
	6	-1508		-42		1508
	7		-442		-12	442
	8	-983		-174		983

		STRESS COMPONENTS (PSI) Sx	Sy	Sz	Sxy	STRESS INTENSITY SI (PSI)
CONTMNT FLANGE & LID	9	-105	-96	-181.	-99	198
	10	-62	-334	-76	102	340
	11	1266	-17	1264	-34	1285
	12	-3106	-136	-3119	-180	2995
NON CONTAINMENT GAMMA SHIELDING	13	145	-1049	95	-28	1197
	14	105	-1069	98	3	1174
	15	-1	-1321	5	-1	1327
	16	≈0	-1328	3	-1	1331
NON CONTAINMENT TRUNNION	17	1	-657	48	7	706
NON CONTAINMENT WELDS	18	-14	-3445	-467	9	3431
	19	22	209	94	18	190
	20	3661	533	1202	1185	3925

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Figure 8.2-1A

HYPOTHETICAL ACCIDENT - 50G BOTTOM DROP ON CONCRETE PAD (50G'S)

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

HYPOTHETICAL ACCIDENT - 50G BOTTOM DROP ON CONCRETE PAD (50G'S)

NORTHERN STATES POWER COMPANY
Xcel Energy
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA

DRAWN BY: VLS	REVISION: 3	FIG8.2-1A_REV_3
PAGE. NO. NA	DATE: 04-27-06	

LOCATION			STRESS INTENSITY S.I. (PSI)				ΣS.I. (a) PSI	TOTAL FOR BOLT PRELOAD = 51k ΣS.I. (b) PSI
			BOLT PRELOAD(a)	INTERNAL PRESSURE	BOTTOM DROP			
CONTAINMENT VESSEL		1	137	433	10868		11438	11580
		2	52	311	1476		1839	1893
		3	2148	4248	42864		49260	51494
		4	1428	4385	52299		58112	59597
		5	343	736	3778		4857	5214
		6	342	590	3779		4711	5067
		7	2444	405	2391		5240	7782
		8	1857	915	8131		10903	12834
CON'MNT FLANGE & LID		9	2347	593	1271		4211	6652
		10	2369	464	2700		5533	7997
		11	1284	839	1375		3498	4833
		12	2091	2034	2839		6964	9139
NON CONTAINMENT	GAMMA SHIELDING	13	246	1216	20715		22177	22433
		14	265	1413	18970		20648	20924
		15	59	462	1091		1612	1673
		16	59	310	1070		1439	1500
	TRUNNION	17	45	370	316		731	778
	WELDS	18	568	2031	24132		26731	27322
		19	2987	1200	3402		7589	10695
		20	5754	2549	3924		12227	18211
CONTAINMENT ALLOWABLE STRESS			HYPOTHETICAL ACCIDENT LOAD					
			COMBINATION			$P_m < 2.4$ Sm =	42.000	
						$(P_l + P_b) < 3.6$ Sm =	63,000	

(a) CORRESPOND TO 25,000 psi BOLT PRELOAD STRESS

(b) $$\text{TOTAL} = \Sigma\text{S.I.} + \text{BOLT PRELOAD} \times \left(\frac{51}{25} - 1\right)$$

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Figure 8.2–1B

HYPOTHETICAL ACCIDENT – LOAD COMBINATION (1)

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

HYPOTHETICAL ACCIDENT - LOAD COMBINATION (1)

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 5	FIG8.2-1B_REV_5
	PAGE. NO. NA	DATE: 04-27-06	

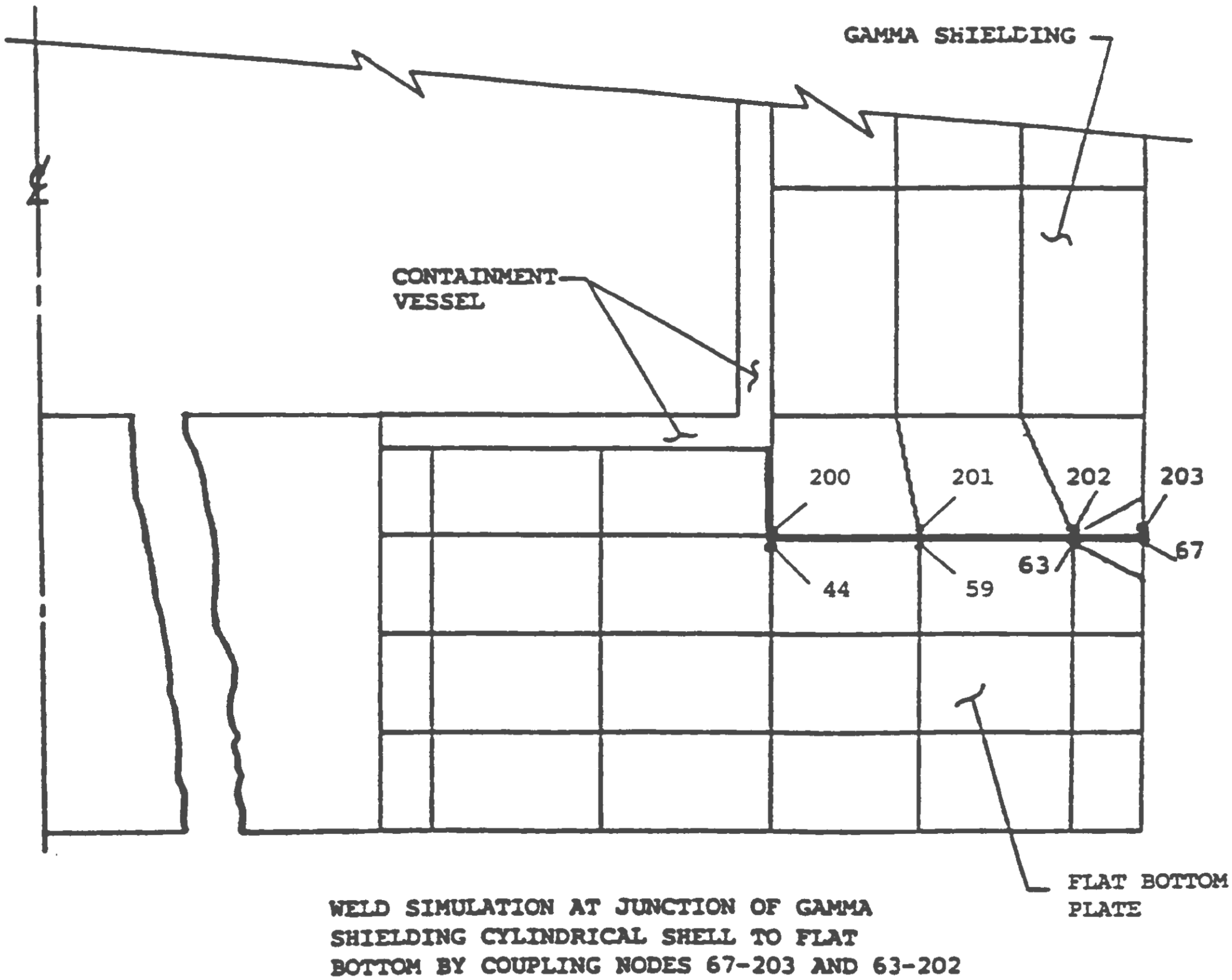

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Figure 8.2-1C

TN-40 CASK BODY BOTTOM CORNER 50G BOTTOM DROP

PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

TN-40 CASK BODY BOTTOM CORNER 50G BOTTOM DROP

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG8.2-1C_REV_3
	PAGE. NO. NA	DATE: 04-27-06	

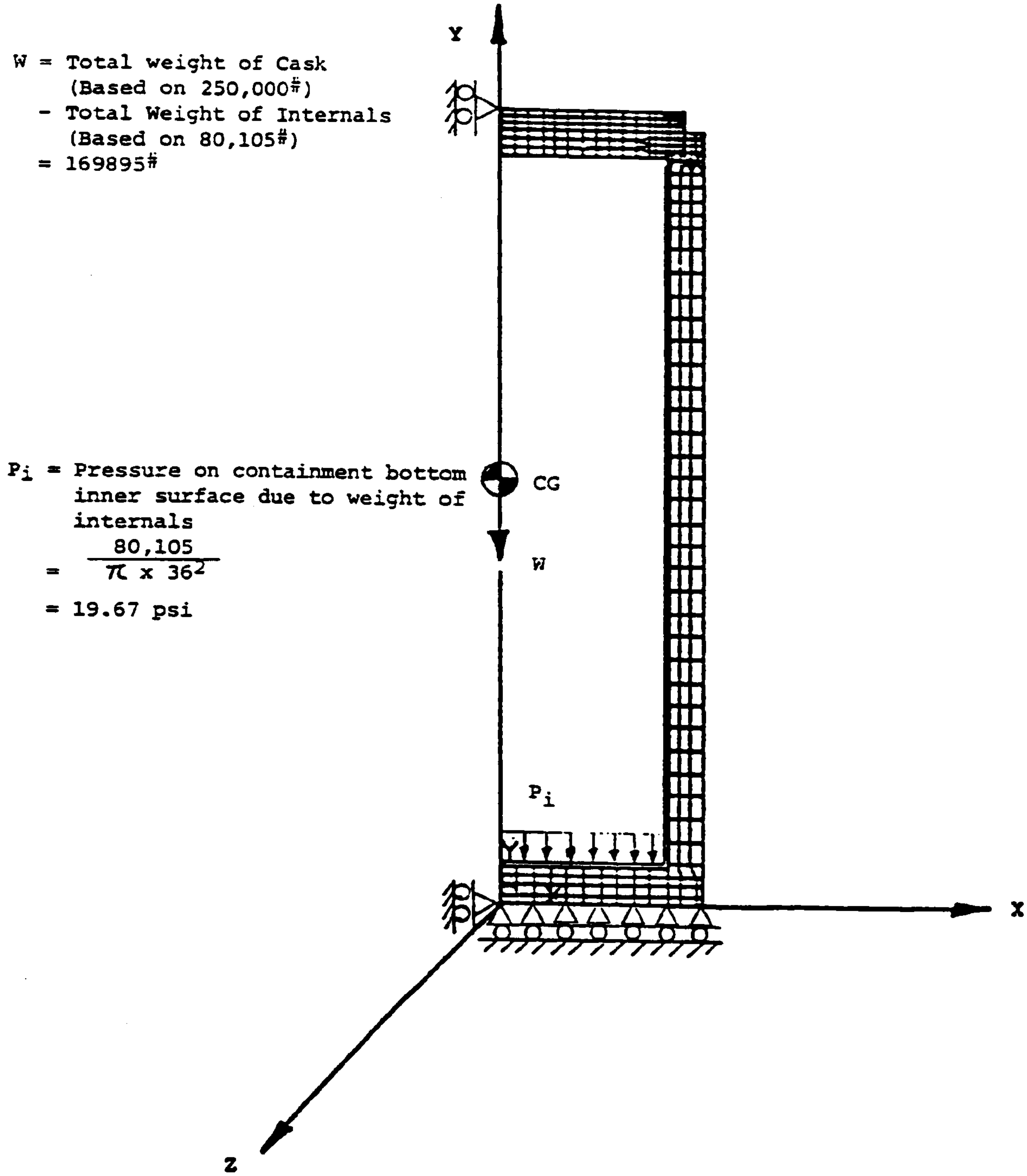

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Figure 8.2-1D

TN-40 50G DOWN LOADING & BOUNDARY CONDITIONS

PRAIRIE ISLAND ISFSI SAFETY ANALYSIS REPORT

TN-40 50G DOWN LOADING & BOUNDARY CONDITIONS

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 3	FIG8.2-1D_REV_3
	PAGE. NO. NA	DATE: 04-27-06	

FIGURE 8.2-2, REV. 1

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CROSS SECTION		DIRECT STRESS	BOLT PRELOAD(a)	INTERNAL PRESSURE	HYPOTHETICAL ACCIDENT	PRINCIPAL STRESSES $\Sigma_{(a)}$	PRINCIPAL STRESSES Σ (DUE TO BOLT PRELOAD(b)=51k)
3-4	BOTTOM DROP COMB.	MERIDONAL S DIR S	-360	-119	4349	3870	3496
		HOOP S DIR TH	-174	-119	4380	4087	3906

SI = 4087 psi < 2.4 Sm = 42,000 psi

(a) CORRESPOND TO 25,000 psi PRELOAD STRESS

(b) TOTAL = PRINCIPAL STRESS Σ + BOLT PRELOAD X $\left(\frac{51}{25} - 1\right)$

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Figure 8.2-5

MEMBRANE STRESS INTENSITY AT SECTION 3-4 (TN-40 CASK)

PRAIRIE ISLAND ISFSI SAFETY ANALYSIS REPORT

MEMBRANE STRESS INTENSITY AT SECTION 3-4 (TN-40 CASK)

NORTHERN STATES POWER COMPANY
Xcel Energy
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING, MINNESOTA

DRAWN BY: VLS	REVISION: 8	FIG8.2-5_REV_8
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FIGURE 8.2-20, REV. 1

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FIGURE 8.2-21, REV. 1

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FIGURE 8.2-22, REV. 1

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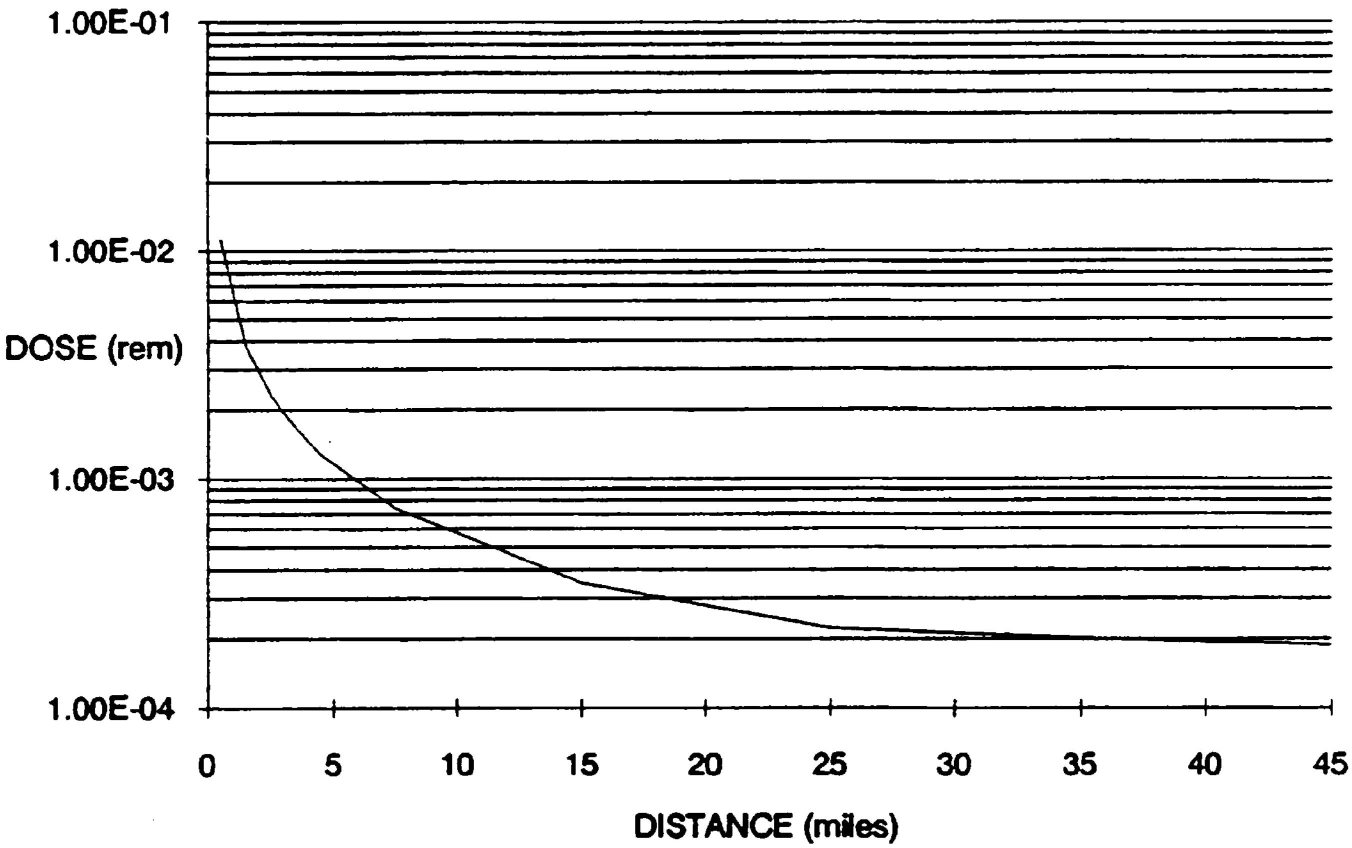

NOTE:

1. **DOSES CALCULATED BASED UPON LOSS OF CONFINEMENT BARRIER OF TN-40 CASK WITH 40 FUEL ASSEMBLIES AS SPECIFIED IN SECTION 3.1.1**

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Figure 8.2-33
DOSE VS. DISTANCE AT POPULATION SECTOR MIDPOINT DISTANCES BETWEEN 0 AND 50 MILES
PRAIRIE ISLAND ISFSI
SAFETY ANALYSIS REPORT

DOSE VS. DISTANCE AT POPULATION SECTOR MIDPOINT DISTANCES BETWEEN 0 AND 50 MILES

NORTHERN STATES POWER COMPANY Xcel Energy PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING, MINNESOTA	DRAWN BY: VLS	REVISION: 0	FIG8.2-33_REV_0
	PAGE. NO. NA	DATE: 04-27-06	

SECTION 9

CONDUCT OF OPERATIONS

9.1 ORGANIZATIONAL STRUCTURE

9.1.1 CORPORATE ORGANIZATION

9.1.1.1 CORPORATE FUNCTIONS, RESPONSIBILITIES AND AUTHORITIES

Northern States Power Company was incorporated in Minnesota as a wholly owned subsidiary of Xcel Energy, Inc. Figure 9.1-1 illustrates the NSPM corporate organization. The NSPM corporate organization is fully described in Chapter 13 of the Prairie Island USAR (Reference 1). All activities at Prairie Island Nuclear Generating Plant are the responsibility of the NSPM Chief Nuclear Officer (CNO). 01153239

The Plant Manager is responsible for operational activities at the Prairie Island Nuclear Generating Plant. He is also responsible for operation of the Prairie Island ISFSI.

9.1.1.2 ISFSI PROJECT ORGANIZATION

The project team responsible for engineering and design, procurement, construction, quality assurance, and testing of the Prairie Island ISFSI is shown in Figure 9.1-3.

The Project Manager - Dry Cask Storage at Prairie Island is responsible for activities relating to engineering and design, procurement, construction, quality assurance, and testing of the Prairie Island ISFSI.

Quality Assurance is provided as described in the Northern States Power Company Operational Quality Assurance Plan. The Project Manager - Dry Cask Quality Assurance for Prairie Island is responsible for quality assurance activities during design and construction.

9.1.1.3 RELATIONSHIP WITH CONTRACTORS AND SUPPLIERS

The Project Manager is the primary interface with the cask supplier, architect-engineer, and other equipment vendors. The Construction Superintendent is the primary interface with construction subcontractors.

9.1.1.4 TECHNICAL STAFF

The engineering technical staff, under the direction of the Project Manager, is responsible for development of design criteria associated with procurement of the storage casks and engineering services. The technical staff is also responsible for review of cask, transporter, and other ISFSI design documentation provided by the cask vendor, Transnuclear, and the architect-engineer, Stone & Webster.

9.1.2 OPERATING ORGANIZATION, MANAGEMENT AND ADMINISTRATIVE CONTROL SYSTEM

9.1.2.1 ONSITE ORGANIZATION

Figure 9.1-2 illustrates the onsite organization at Prairie Island Nuclear Generating Plant. Figure 9.1-4 illustrates the organization responsible for cask operation. The Plant Manager will be responsible for operations at the ISFSI. The various superintendents and staff members will have functional responsibility for operations, maintenance, and radiation protection. In addition, an engineer in the Nuclear Engineering Group has been assigned the responsibility of interfacing with the Project Manager during design, construction, and operation of the ISFSI.

9.1.2.2 PERSONNEL FUNCTIONS, RESPONSIBILITIES AND AUTHORITIES

The Prairie Island USAR, Section 13.2 provides a general description of personnel functions, responsibilities, and authorities of the onsite organization. Due to the passive nature of the ISFSI, it is expected that the existing organization can accommodate the additional responsibilities associated with ISFSI operation without the need for additional staff.

9.1.3 PERSONNEL QUALIFICATION REQUIREMENTS

9.1.3.1 MINIMUM QUALIFICATION REQUIREMENTS

Each member of the plant staff is required to meet or exceed the minimum qualifications of ANSI 18.1-1971 as noted in the applicable section of the Technical Specifications. Training and retraining programs are conducted to maintain a qualified staff of technical and operations personnel. The training program is under the direction of the Plant Manager.

9.1.3.2 QUALIFICATIONS OF PERSONNEL

Qualifications of personnel assigned to managerial and technical positions are set forth in the Prairie Island USAR, Section 13.2.

9.1.4 LIAISON WITH OUTSIDE ORGANIZATIONS

Stone & Webster, as architect-engineer, provided engineering expertise required to design the ISFSI structures and foundations and developed specifications for use in procurement of the cask transporter and other equipment. Transnuclear is providing technical expertise required in the design, fabrication, and delivery of the casks.

The Project Manager is responsible for directing the activities of these contractors and for procuring equipment and services of other contractors.

9.2 STARTUP TESTING AND OPERATION

9.2.1 ADMINISTRATIVE PROCEDURES FOR CONDUCTING TEST PROGRAM

The administrative procedures for the ISFSI will be the same as those used for the Prairie Island Nuclear Generating Plant. Any changes to, or deviations from, these procedures and instructions will be reviewed and approved by the Station Operations Committee as appropriate.

9.2.2 TEST PROGRAM DESCRIPTION

9.2.2.1 PHYSICAL FACILITIES

Before startup and during the lifetime of the ISFSI, the cask monitoring instrumentation, the electrical system, the communications system, and the storage casks will be tested to ensure their proper functioning.

The cask monitoring instrumentation alarms will be tested to ensure that individual alarm signals annunciate at the local annunciator enclosure at the ISFSI location.

The electrical system will be tested to ensure that power is available for the cask monitoring instrumentation and the local annunciator. The lighting and service receptacles are also tested for proper operation.

The communications system will be tested to ensure that the telephone at the local annunciator is properly connected into the plant phone system.

The storage casks will be tested with a dummy fuel assembly prior to fuel loading to ensure that the assemblies will fit properly.

The seals of the casks will be inspected prior to and tested following fuel loading.

9.2.2.2 OPERATIONS

Testing and calibration of instruments and components in use at the ISFSI will be done in accordance with procedures established for similar equipment in use at the Prairie Island Nuclear Generating Plant. Acceptance criteria and corrective actions for test margins and response times will be specified by the equipment vendors.

9.2.3 TEST DISCUSSION

The preoperational test purposes, responses, acceptance criteria, margins, and corrective actions are discussed in Section 9.2.2. Instrumentation, electrical, and communications equipment will be functionally tested to confirm operability.

9.2.4 COMPLETION OF PRE-OPERATIONAL TEST PROGRAM

Preoperational testing of the Prairie Island ISFSI was completed on April 15, 1995. Pursuant to the requirements of 10CFR72.82(e), a report of the preoperational test acceptance criteria and test results was submitted to the NRC by Reference 3.

9.3 TRAINING PROGRAM

The training program has the objective of providing and maintaining a well qualified work force for safe and efficient operation of the ISFSI. The existing Prairie Island Nuclear Generating Plant training program will be used to provide this training and indoctrination. Additional sections to this program will be added as needed to include information pertinent to the ISFSI. Training courses will be prepared by training center personnel in cooperation with engineering personnel qualified in the particular topical or functional area. All personnel working in the fuel storage area will receive radiation and safety training and those actually performing cask and fuel handling functions will be given additional training in specific areas as required by the Radiation Protection program in effect at the Prairie Island Nuclear Generating Plant. The retraining schedule will be consistent with retraining requirements in effect for personnel involved in fuel handling operations.

Training records will be maintained for 5 years. Such records will include dates and hours of training and other documentation on training subjects, information on physical requirements, job performance statements, copies of written examinations, information pertaining to walk-through examinations, and retesting particulars.

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9.4 NORMAL OPERATIONS

9.4.1 PROCEDURES

Detailed written procedures for all normal operating, maintenance, and testing procedures will be prepared and will be in effect prior to operation of the ISFSI. These procedures are briefly described in Section 9.4.1.1 through 9.4.1.8.

These procedures will be reviewed and approved by the cognizant superintendent and the Plant's Operations Committee. Any later revisions necessitated by operational experience, changes to systems and components, new requirements, clerical errors, etc., will be reviewed, approved, and documented by the same personnel.

9.4.1.1 ADMINISTRATIVE PROCEDURES

Administrative procedures will provide rules and instructions to all ISFSI personnel to provide a clear understanding of operating philosophy and management policies. These procedures include instructions pertaining to personnel conduct and control, including consideration of job-related factors, e.g., work hours, entering and exiting the ISFSI, organization, and responsibility, etc.

9.4.1.2 ANNUNCIATOR RESPONSE GUIDES

Annunciator response guides will provide information relative to each alarm annunciator which monitors cask and fuel parameters. The procedures will provide alarm set points and appropriate corrective action.

9.4.1.3 RADIATION PROTECTION PROCEDURES

Radiation protection procedures are used to implement a radiation control program. The radiation control program will involve the acquisition of data and provision of equipment to perform necessary radiation surveys, measurements, and evaluations for the assessment and control of radiation hazards associated with the operation of the ISFSI. Procedures will be developed and implemented for: monitoring exposures of employees, utilizing accepted techniques, radiation surveys of work areas, radiation monitoring of maintenance activities, and for records maintenance demonstrating the adequacy of measures taken to control radiation exposures of employees and others within prescribed limits and as low as practicable.

Entrance to the ISFSI and all work performed inside will require a radiation work permit and will be controlled by radiation protection personnel. Existing radiation protection procedures will be revised, where applicable, prior to operation of the ISFSI. The revised procedures will ensure safety of personnel performing surveillance and maintenance at the ISFSI.

Regularly scheduled surveillance will be required, but calculations have shown that doses received by personnel from all anticipated activities are within the guidelines set forth in 10CFR20. There are no credible events during normal storage that could lead to a high radiation condition (as defined by PINGP Operations manual, F2, Reference 2) at the ISFSI. Accident analyses in Chapter 8 also show no credible event leading to the high radiation release during transport, placement, or storage. In the unlikely event that maintenance on the cask involving cask integrity is to be performed, the cask would be returned to the Prairie Island Nuclear Generating Plant and the work performed inside the Auxiliary Building.

9.4.1.4 MAINTENANCE PROCEDURES

Maintenance procedures will be established for performing preventative and corrective maintenance on ISFSI equipment. Preventative maintenance will be performed on a periodic basis to preclude the degradation of ISFSI systems, equipment, and components. Corrective maintenance is that performed to rectify any unexpected system, equipment, or component malfunction, and is initiated as the need arises.

9.4.1.5 OPERATING PROCEDURES

The operating procedures will provide instructions for handling, loading, sealing, transporting, and storing the ISFSI casks.

9.4.1.6 TEST PROCEDURES

Periodic test procedures will be formulated to ensure that ISFSI systems, equipment, and components are observed on a routine basis to verify operability.

9.4.1.7 PREOPERATIONAL TEST PROCEDURES

Preoperational test procedures will be established to ensure that ISFSI structures, systems, and components satisfactorily perform their required functions. These test procedures will further ensure that the ISFSI has been properly designed and constructed and is ready to operate in a manner that will not endanger the health and safety of the public.

9.4.1.8 QUALITY ASSURANCE PROCEDURES

Quality assurance procedures will be established to ensure that the operation and maintenance of the ISFSI is performed in accordance with the QA program described in Chapter 11.

9.4.2 RECORDS

Records will be maintained on file as described in the Prairie Island USAR, Section 13.5.

9.5 EMERGENCY PLANNING

The Prairie Island Emergency Plan describes the organization, assessment actions, conditions for activation of the emergency organization, notification procedures, emergency facilities and equipment, training, provisions for maintaining emergency preparedness, and recovery criteria used at the Prairie Island Nuclear Generating Plant. This emergency plan will also be used for any radiological emergencies that may arise at the ISFSI.

Portions of the Emergency Plan and applicable implementing procedures reflect the actions to be taken during off normal and accident conditions. Damage to a loaded cask confinement boundary will result in declaration of a Notification of Unusual Event. 01029651

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9.6 PHYSICAL SECURITY PLAN

The purpose of the security program for the ISFSI is to establish and maintain a physical security program that has the capabilities for the protection of spent fuel stored in the cask system.

Additional information regarding the security program for the ISFSI is contained in a separate enclosure that is withheld from public disclosure in accordance with 10CFR2.790(d) and 10CFR73.21. This enclosure addresses the Physical Security Plan, Safeguards Contingency Plan, and Training and Qualification Plan.

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9.7 REFERENCES

1. Northern States Power Company, Prairie Island Nuclear Generating Plant Updated Safety Analysis Report, Revision 18, Docket Nos. 50-282 (Unit 1) and 50-306 (Unit 2).

2. Prairie Island Nuclear Generating Plant, Operations Manual, Section F2, Radiation Safety.

3. Letter, R.O. Anderson (NSP) to U S Nuclear Regulatory Commission, dated April 20, 1995, "Prairie Island Independent Spent Fuel Storage Installation Preoperational Test Acceptance Criteria and Test Results".

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Xcel Corporate Organization

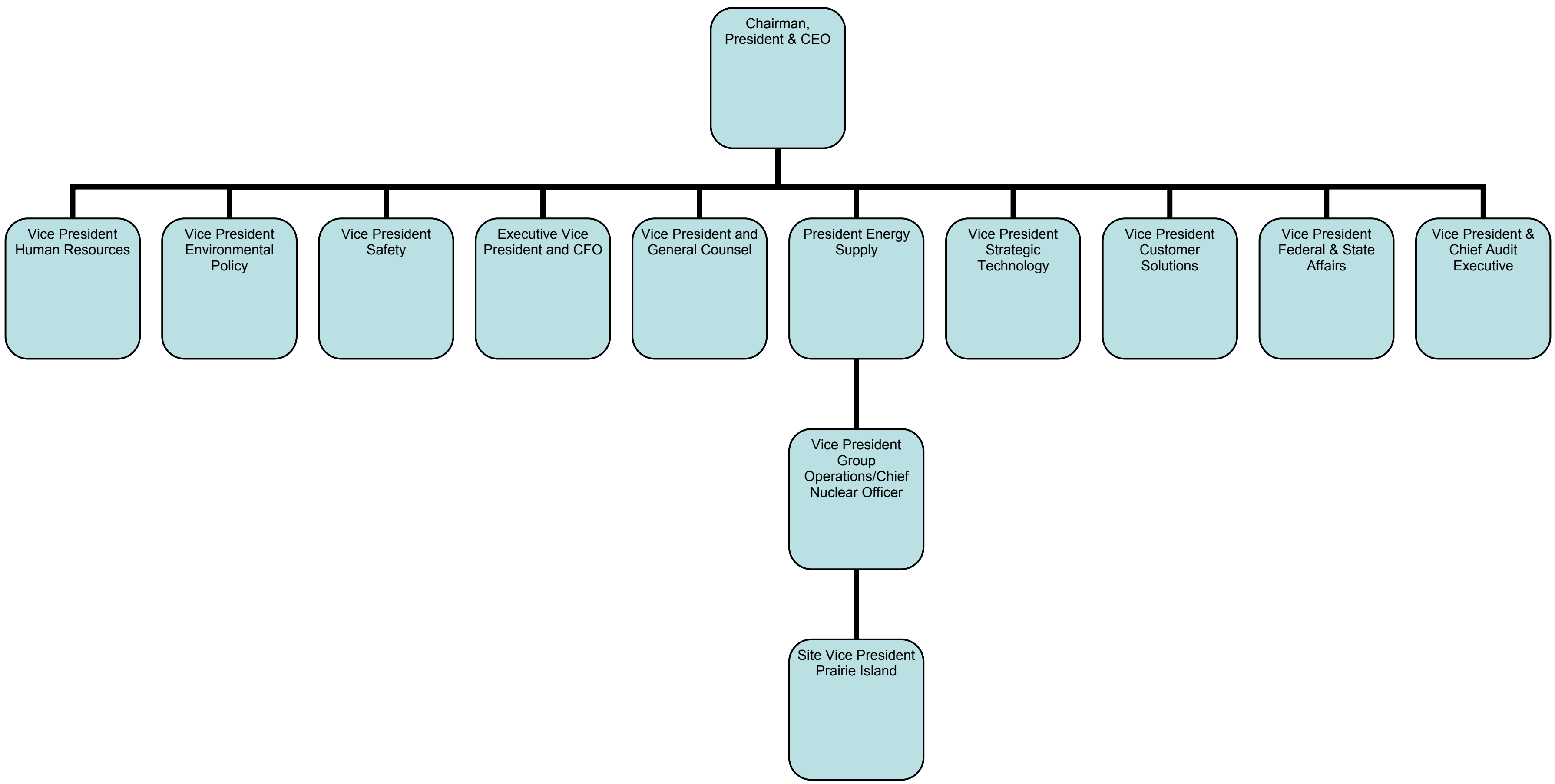

FIGURE 9.1-1, REV 12

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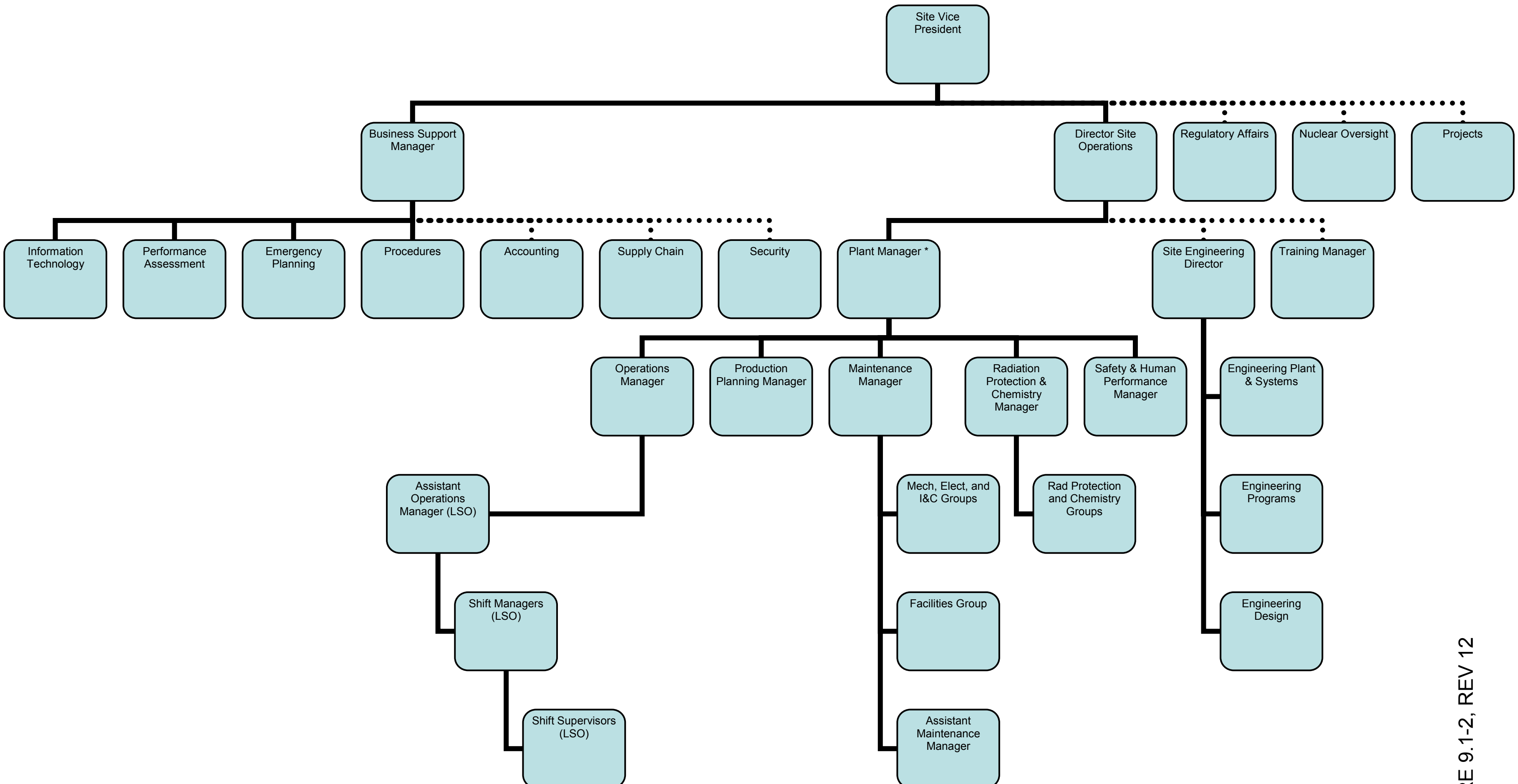
Prairie Island Plant Organization



* = The Plant Manager chairs the PORC (Plant Operations Review Committee)
LSO = Licensed Senior Operator

FIGURE 9.1-2, REV 12

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ISFSI Cask Fabrication Organization

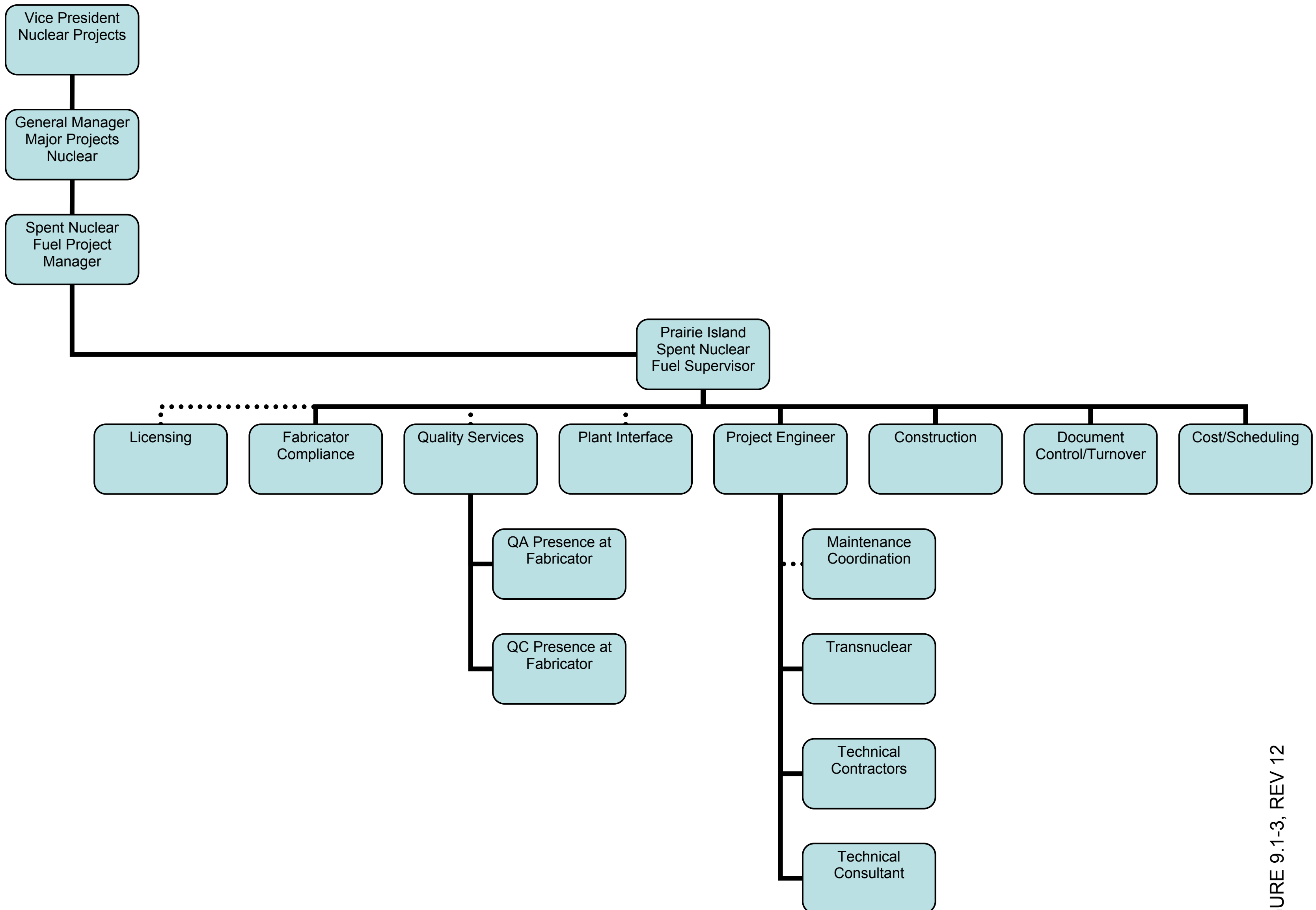

FIGURE 9.1-3, REV 12

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ISFSI Operating Organization

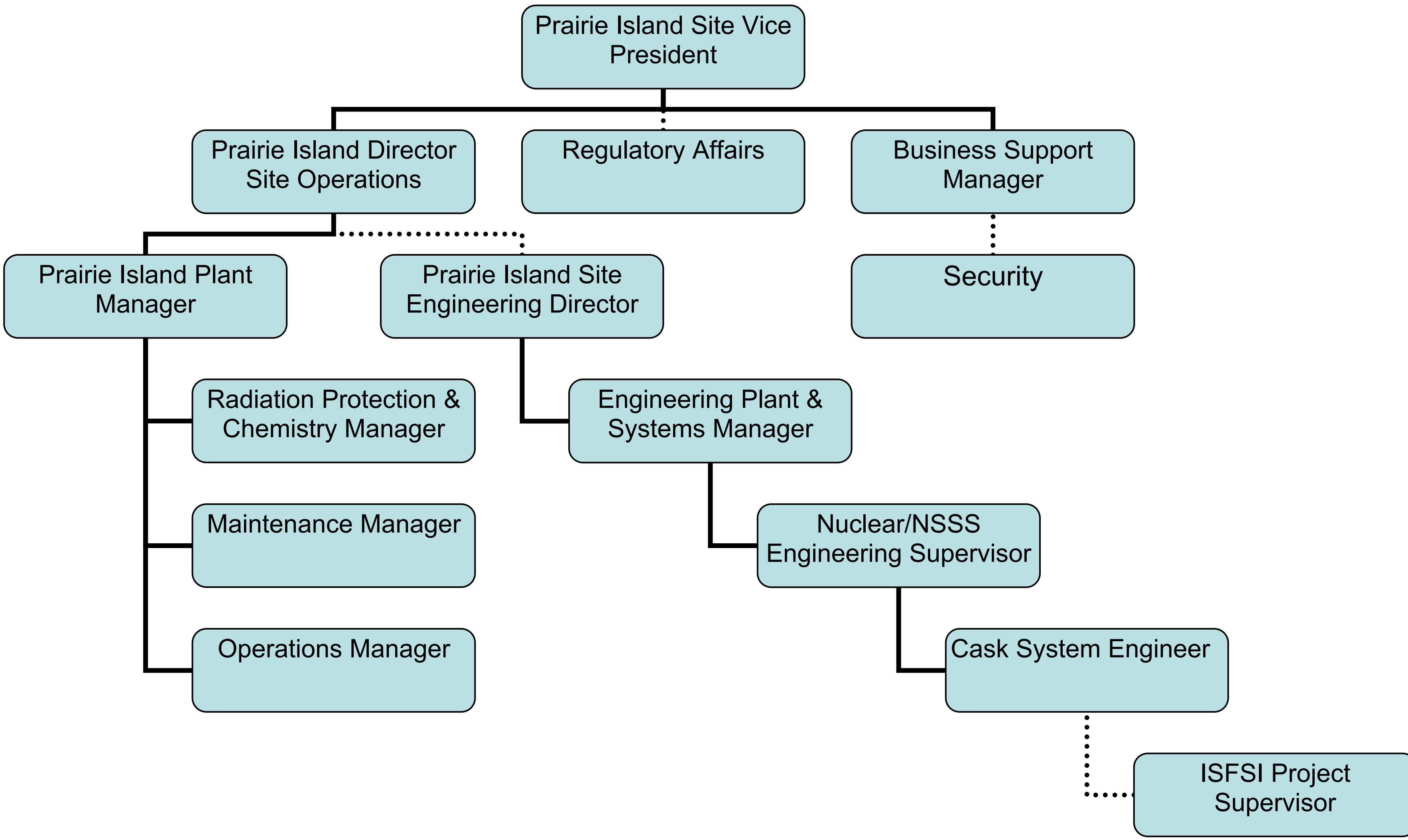

FIGURE 9.1-4, REV 12

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SECTION 10

OPERATING CONTROLS AND LIMITS

10.1 FUNCTIONAL AND OPERATING LIMITS, MONITORING INSTRUMENTS AND LIMITING CONTROL SETTINGS

10.1.1 FUEL

1. Specification:

   The spent nuclear fuel to be stored at the ISFSI shall meet the following requirements:

   A. Only fuel irradiated at the Prairie Island Nuclear Generating Plant may be used.

   B. Maximum initial enrichment shall not exceed 3.85 weight percent U-235.

   C. Maximum assembly average burnup shall not exceed 45,000 megawatt-days per metric ton uranium.

   D. Fuel shall have cooled a minimum of 10 years after reactor discharge and prior to storage in the ISFSI.

   E. Fuel shall be intact unconsolidated fuel. Partial fuel assemblies, that is, fuel assemblies from which fuel pins are missing, must not be stored unless dummy fuel pins are used to displace an amount of water equal to that of the displaced original pins.

   F. Fuel assemblies known or suspected to have structural defects sufficiently severe as to adversely affect fuel handling shall not be loaded into a cask for storage, unless canned.

2. Applicability: The specification is applicable to all fuel to be stored in the TN-40 casks at the Prairie Island ISFSI.

3. Objective: The specification was derived to ensure that the peak fuel rod temperature, surface doses, and nuclear subcriticality are below design values.

4. Action: If this specification is not met, additional analysis and/or data must be presented demonstrating that the nonconformance does not exceed safe operating limits before the spent fuel can be placed in the cask for storage.

5. Surveillance: Surveillance shall be performed as specified in Sections 10.3.1 and 10.3.2.

6. Basis: The design criteria and subsequent safety analyses of the ISFSI and storage casks assumed certain characteristics and limitations for the fuels that are to be received and stored. This specification assures that these bases remain valid by defining the source of the spent fuel, and limits on maximum initial enrichment, irradiation history, and minimum post irradiation cooling time. The objective of these limits is to protect the integrity of the spent fuel by ensuring that the thermal and criticality analyses are valid for fuel stored at the ISFSI.

10.1.2 CASKS

1. Specification: The spent fuel storage casks used at the ISFSI shall meet the following requirements:

    A. Cask surface temperature shall be less than 250°F.

    B. The cask surface dose rate (on the pad) shall be less than 200 mRem/hr.

    C. Removable surface contamination levels on the cask shall be less than 1000 dis/min/100cm^2 from beta and gamma emitting sources and 20 dis/min/100cm^2 from alpha emitting sources.

    D. Maximum lifting height of a cask by a non-single failure proof lifting device shall be less than 18 inches.

2. Applicability: The specification is applicable to the TN-40 casks.

3. Objective: The objective is to ensure that the casks have been loaded and handled in accordance with design basis criteria.

4. Action: If temperature, surface dose rates, or contamination levels exceed limits, the cask shall not be transported to the ISFSI. If maximum lift height is exceeded, the transport activities shall be stopped and the cask lowered to within the acceptable limit.

5. Surveillance: Surveillances shall be performed as specified in Sections 10.3.3, 10.3.4, and 10.3.5.

6. Basis: The design criteria and subsequent safety analysis of the Transnuclear TN-40 cask assumed certain characteristics and operating limits for the use of the casks. This specification assures that those design criteria are not exceeded.

Confirmation that the cask surface temperature is within the prescribed limit will ensure that the cladding temperature of the fuel assemblies is less than the maximum design basis temperature of 340°C. This will protect the integrity of the spent fuel stored in the ISFSI by ensuring that the thermal analyses are valid for the fuel stored in the ISFSI.

Confirmation that cask surface dose and surface contamination levels are below prescribed limits will protect employees against occupational exposures by ensuring compliance with occupational dose limits and ALARA principles.

Confirmation that cask lifting heights are within the prescribed limit will protect the cask integrity and guard against uncontrolled release of radioactive material by ensuring the thermal, criticality, and radiological analyses remain valid following an accidental cask drop.

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10.2 LIMITING CONDITIONS FOR OPERATION

Limiting conditions are the lowest functional capability or performance levels of equipment required for safe operation.

10.2.1 CASK INTERNAL HELIUM PRESSURE

1. Specification: The cask shall be backfilled with a helium cover gas to a pressure of 20 $\pm$ 1 psia (5.3 $\pm$ 1 psig).

2. Applicability: The specification is applicable to the TN-40 casks.

3. Objective: The objective is to ensure that the cask is backfilled with helium in accordance with design basis criteria.

4. Action: If internal pressure is not within specified limits, the cask shall not be transported to the ISFSI.

5. Surveillance Requirements: Surveillances shall be performed as specified in Section 10.3.6.

6. Basis: The thermal and pressure analyses performed for the cask assume the use of a cover gas. Compliance with this limiting condition will ensure long term maintenance of fuel clad integrity. Periodic testing is not required due to the reliability of the redundant monitoring system.

10.2.2 CASK LEAKAGE

1. Specification: The cask leakage shall be less than 10^{-5} atm-cc/sec.

2. Applicability: The specification is applicable to the TN-40 cask.

3. Objective: The objective is to ensure that cask leakage is within limits assumed in the radiological dose calculations.

4. Action: If leakage is above the specified limit, the cask shall not be transported to the ISFSI.

5. Surveillance Requirements: Surveillances shall be performed as specified in Section 10.3.7.Basis:

6. Basis: Compliance with this limiting condition will ensure long-term maintenance of cask integrity. Periodic testing is not required due to the reliability of the redundant monitoring system.

10.3 SURVEILLANCE REQUIREMENTS

10.3.1 FUEL PARAMETERS

Prior to cask loading, the fuel selected to be loaded shall have been reviewed to ensure that it is within the cask-specific functional and operating limits. This information shall be documented for each assembly to be loaded into the cask.

10.3.2 CASK LOADING

Prior to cask closure, the actual cask loading will be verified to be correct.

10.3.3 CASK TEMPERATURE

A minimum of 24 hours after cask loading and prior to moving the cask to the storage pad, the surface temperature of the cask shall be measured to ensure that it is within the functional and operating limit.

10.3.4 CASK SURFACE DOSE RATE

Prior to moving a loaded cask to the storage pad, gamma and neutron measurements shall be taken on the outside surface of the cask surface. These dose rates shall be less than the surface dose rate limit.

10.3.5 CASK CONTAMINATION

Prior to moving a loaded cask to the storage pad, the cask removable surface contamination levels shall be measured to ensure they are less than the contamination limit.

10.3.6 CASK INTERNAL HELIUM PRESSURE

Prior to moving a loaded cask to the storage pad, the helium pressure shall be measured to ensure it is within the pressure limit.

10.3.7 CASK LEAKAGE

Prior to moving the cask to the storage pad, the cask seal shall be tested using a helium leak detector to ensure that the seal leak tightness is within the leakage limit

10.3.8 ISFSI SAFETY STATUS

A visual surveillance of the ISFSI shall be performed on a quarterly basis to determine that no significant damage or deterioration of the exterior of the emplaced casks has occurred. Surveillance shall also include observation to determine that no significant accumulation of debris on cask surfaces has occurred.

10.3.9 ISFSI AREA DOSE RATE

TLDs located on the ISFSI site fence shall be read quarterly.

10.4 DESIGN FEATURES

The ISFSI cask storage pads are constructed of reinforced concrete, with nominal dimensions of 36 ft. x 216 ft. x 3 ft. thick. The top of the concrete pad elevation is set at a minimum elevation of 693'-0" to preclude immersion of the cask seals during the maximum probable flood. The actual pad elevation is 694'-6".

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10.5 ADMINISTRATIVE CONTROLS

The ISFSI is located on the Prairie Island Nuclear Generating Plant site and is managed and operated by the Station staff. The administrative controls shall be in accordance with the requirements of the Station Facility Operating License and associated Technical Specifications.

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10.6 REFERENCES

None.

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SECTION 11

QUALITY ASSURANCE

11.1 QUALITY ASSURANCE PROGRAM DESCRIPTION

10CFR72.140 requires that a quality assurance program be established and
implemented. The previously approved NSPM QA Program which satisfies applicable 01153239
criteria of 10CFR50, Appendix B, will be applied to activities, structures, systems, and
components of the ISFSI commensurate with their importance to safety.

Since NSPM is currently licensed under 10CFR50 to operate nuclear power facilities, a
quality assurance (QA) program meeting the requirements of 10CFR50, Appendix B, is 01153239
already in place. The governing document for this program is "Northern States Power
Company-Minnesota, Quality Assurance Topical Report," (QATR) (Reference 1) which 01201386
has been reviewed and approved by the NRC. This program is implemented through
directives, instructions and procedures. The objective of the company QATR is to
comply with the criteria as expressed in 10CFR50, Appendix B, as amended, and with
the quality assurance program requirements for nuclear power plants as referenced in
the Regulatory Guides and industry standards. This program will be applied to those
activities associated with the ISFSI. No changes to this program are required for the
ISFSI activities.

As indicated in previous chapters, the storage casks and the concrete storage pads are the only components with safety related components. Those components of the storage casks and concrete storage pads which are safety related are listed in Table 4.5-1. As such, the QATR delineates the requirements for the engineering, procurement, fabrication, and inspection of this equipment.

The procurement documents (specifications, requisitions, etc.) of the casks will be
reviewed technically prior to use to ensure that the proper criteria have been specified.
During the cask design phase, vendor information (drawings, specifications,
procedures, etc.) will be reviewed to ensure compliance with technical requirements.
During cask fabrication, Shop Inspectors will visit the vendor's shop to ensure
compliance with requirements and to witness parts of the cask fabrication and testing.
Until NSPM is satisfied that the cask meets the technical requirements, the vendor may 01153239
not ship the cask.

The concrete and reinforcing steel for the concrete storage pads was purchased by NSP and installed by the general contractor under NSP's direction to assure that materials conform as specified and that the concrete placement conforms to the drawings and the specification requirements.

Reinforcing steel is required to be furnished with certified material test reports for chemical analysis and physical tests for each heat of steel to verify compliance with the applicable ASTM specification requirements. In addition, tensile properties are required for representative samples of actual reinforcing steel.

Concrete is required to be furnished with materials (cement, aggregate, water, and admixtures) having certified material test reports which verify compliance with the specified ASTM requirements. Concrete materials are required to be stored, handled, measured, mixed and transported per standard industry practice (ACI 304). Inspections to verify compliance will be performed throughout the project.

Batch plant inspections will be performed to ensure proper mixing proportions.

The fresh concrete will be sampled at the site by an independent testing agency to verify key properties (slump, air content, temperature and unit weight) meet the specification requirements. Compressive strength will be tested for compliance on the 28th day following placement.

Each of the 18 criteria of 10CFR50, Appendix B and their applicability to the storage casks, concrete storage pads and associated activities are described in Sections 11.1.1 through 11.1.18.

11.1.1 ORGANIZATION

Sections A.2, A.3 and A.4 of the QATR define and describe the organization responsible for the establishment and execution of the quality assurance program at the Prairie Island Nuclear Generating Plant. This same organization will be responsible for ensuring that the ISFSI meets the appropriate guidelines of the QATR.

11.1.2 QUALITY ASSURANCE PROGRAM

Sections A.1, A.5 and B.2 of the QATR describe the quality assurance program for the Prairie Island Nuclear Generating Plant. This program will be applied to the ISFSI in accordance with the guidelines for applicability contained in the referenced section.

11.1.3 DESIGN CONTROL

Sections B.2 and B.3 of the QATR ensure that applicable specified design requirements, quality standards, interfaces, checks, and changes are applied, coordinated, and controlled. These same design controls will be applied to the ISFSI in accordance with the applicability guidelines of the referenced section.

11.1.4 PROCUREMENT DOCUMENT CONTROL

Section B.4 of the QATR provides guidelines for ensuring that applicable regulatory requirements, design bases, and other requirements necessary to ensure adequatequality are included or referenced in the documents for procurement of items or services. These same controls will be applied to the storage casks and concrete storage pads.

11.1.5 INSTRUCTIONS, PROCEDURES AND DRAWINGS

Section A.1 of the QATR establishes guidelines for preparing instructions, procedures, and drawings for activities affecting quality. These same guidelines will be applied to the procedures for the handling and maintenance of the storage casks and concrete storage pads.

11.1.6 DOCUMENT CONTROL

Section B.14 of the QATR provides general requirements and guidance for the establishment and execution of document control systems for the Prairie Island Nuclear Generating Plant. The documents associated with the storage casks and concrete storage pads and the records committed to in Chapter 10 will come under the control of this system.

11.1.7 CONTROL OF PURCHASED MATERIALS, EQUIPMENT AND SERVICES

Section B.5 of the QATR establishes procedures which ensure that purchased material, equipment, and services conform to the procurement documents. These same procedures will be applied to the storage casks and concrete storage pads.

11.1.8 IDENTIFICATION AND CONTROL OF MATERIALS, PARTS AND COMPONENTS

Section B.6 of the QATR provides methods and conditions for the identification and control of materials, parts, and components. These same methods and conditions will be applied to the storage casks and concrete storage pads.

11.1.9 CONTROL OF SPECIAL PROCESSES

Section B.11 of the QATR establishes procedures which ensure that special processes (e.g., welding, heat treatment, etc.) are controlled and accomplished by qualified personnel, using qualified procedures, in accordance with applicable requirements. The same procedures will be applied to the storage casks and concrete storage pads.

11.1.10 INSPECTION

Section B.12 of the QATR establishes a program for the inspection of activities affecting quality to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. The same program will be applied to the storage casks and concrete storage pads.

11.1.11 TEST CONTROL

Section B.8 of the QATR establishes a program to ensure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily is performed in accordance with appropriate test procedures. This same program will be applied to the storage casks and concrete storage pads.

11.1.12 CONTROL OF MEASURING AND TEST EQUIPMENT

Section B.9 of the QATR sets forth the requirements of a calibration program to control and verify the accuracy of measuring and test equipment used in activities affecting quality. This same program will be applied to the storage casks and concrete storage pads.

11.1.13 HANDLING, STORAGE AND SHIPPING

Section B.7 of the QATR establishes measures for the packaging, shipping, storage, and handling of Category I items. These same measures will be applied to the storage casks and concrete storage pads.

11.1.14 INSPECTION, TEST AND OPERATING STATUS

Section B.10 of the QATR establishes and defines the measures to indicate the status of inspections and tests performed upon individual items of the Prairie Island Nuclear Generating Plant. These same measures will be applied to the storage casks and concrete storage pads.

11.1.15 NON-CONFORMING MATERIALS, PARTS OR COMPONENTS

Section B.13 of the QATR establishes guidelines and inspections for reporting any deviation from or violation of an authorized code, standard, engineering document, or procedurally established quality requirement. These same guidelines and instructions will be applied to the activities described in Sections 11.1.1 through 11.1.14.

11.1.16 CORRECTIVE ACTION

Sections A.6 and B.13 of the QATR provide procedures for identifying, documenting, reporting, determining the cause of, and correcting defects and conditions adverse to quality. These same procedures will be applied to the activities described in Sections 11.1.1 through 11.1.14.

11.1.17 QUALITY ASSURANCE RECORDS

Section B.15 of the QATR provides general requirements and guidance for the establishment and execution of the records control system at the Prairie Island Nuclear Generating Plant. The records associated with the storage casks and concrete storage pads will be controlled by this system.

11.1.18 AUDITS

Section C.3 of the QATR establishes a comprehensive system of planned and periodic audits to be carried out in order to verify compliance with all aspects of the quality assurance program and to determine the effects of the program. This same system will be applied to the activities described in Sections 11.1.1 through 11.1.17.

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11.2 QUALITY ASSURANCE PROGRAM – CONTRACTORS

11.2.1 ARCHITECT-ENGINEER

As described in the QATR, NSPM has the ultimate responsibility to ensure that the design and engineering of the ISFSI is done in accordance with the plan. In accordance with the plan, the contractor NSP hired to perform the design and engineering of the ISFSI, Stone & Webster Engineering Corporation, performed its work in accordance with an approved QA program (Reference 2). 01153239

11.2.2 CASK SUPPLIER

As described in the QATR, NSPM has the ultimate responsibility for ensuring that the manufacture of safety-related components is done in accordance with the plan. In accordance with the plan, the cask manufacturer must do work under the approved NSPM QA Program. 01153239 01201386

11.2.3 CONCRETE STORAGE PAD CONTRACTOR

As described in its “Operational Quality Assurance Plan,” NSP had the ultimate responsibility for ensuring that the construction of the safety-related concrete storage pads is done in accordance with the plan. In accordance with the plan, the general contractor worked under NSP's approved QA plan.

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11.3 REFERENCES

1. Northern States Power Company-Minnesota, Quality Assurance Topical Report (NSPM-1), most recent revision. 01153239 01201386

2. Stone & Webster Engineering Corporation, Standard Nuclear Quality Assurance Program, Docket No. 99900509, Revision E.

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