Xcel Energy®

Prairie Island Nuclear Generating Plant
Northern States Power Company, Minnesota

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10 CFR 72.48
10 CFR 72.70

U S Nuclear Regulatory Commission
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Prairie Island Independent Spent Fuel Storage Installation (ISFSI)
Docket 72-10
License No. SNM-2506

2009 Biennial Report of Changes, Tests, and Experiments for the Prairie Island ISFSI and Prairie Island ISFSI Safety Analysis Report (SAR)

Per 10 CFR 72.48(d)(2), the interval for the report containing a brief description of any changes, tests, and experiments is not to exceed 24 months. Per 10 CFR 72.70(c)(6), ISFSI SAR updates shall be filed every 24 months from the date of issuance of the license. The Prairie Island ISFSI license was issued October 19, 1993.

There is one new 10 CFR 72.48 Evaluation to report at this time. Revision No. 12 to the ISFSI SAR with updating instructions is provided in Enclosure 1. This revision is submitted pursuant to the requirements of 10 CFR 72.70, and brings the information in the ISFSI SAR up-to-date through October 7, 2009. Revision No. 12 includes a number of editorial changes. One change required a 10 CFR 72.48 Evaluation, one change was determined to be acceptable via a 10 CFR 72.48 Screening, and one change was made with the support of a license amendment. Information regarding these changes and a summary of the 72.48 evaluation is provided in Enclosure 2.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

I certify that the information contained herein accurately presents changes made since the previous submittal.

M. Schimmel

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Enclosures (2)

cc: Director, Spent Fuel Project Office, NRC
NMSS Project Manager, NRC
Regional Administrator - Region III, NRC
Senior Resident Inspector, NRC
NRR Project Manager, NRC

ENCLOSURE 1

PRAIRIE ISLAND
INDEPENDENT SPENT FUEL STORAGE INSTALLATION
SAFETY ANALYSIS REPORT
REVISION 12
Revised Pages (with update instructions)

Instructions:

1. Remove individual ISFSI SAR pages and replace with the new Revision 12 pages provided in this attachment. Removed pages should be discarded.

2. When page removal/replacement is complete, review the ISFSI SAR Listing of Effective Pages to ensure the copy of the ISFSI SAR is current and complete. Contact Prairie Island Site Licensing at 651-388-1121, Extension 7384 if you require additional assistance.

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Prepared By

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or Designee

Approved By
Site Vice President
or Designee

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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. 01153239

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 is rated at 1,650 MWt, which is the equivalent of 560 MWe. A complete description of the power plants is contained in the Prairie Island USAR.

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.

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* Northern States Power Company was incorporated in Minnesota as a wholly owned subsidiary of Xcel Energy, Inc. effective August 18, 2008. 01153239

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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 procurement, development of security plans, construction management and construction services using specialty subcontractors, as required. 01153239

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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.

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 $E = 30 \times 10^6$ 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 $\div$ 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.

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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:

$$\frac{\textit{Applied Comprehensive Load}}{\textit{Allowable Comprehensive Load}} + \frac{\textit{Applied Bending Moment}}{\textit{Allowable Bending Moment}} \leq 1.0$$

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 01136263
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.

4.2.4 INSTRUMENTATION SYSTEM DESCRIPTION

No safety related instrumentation is required due to the passive nature of the ISFSI design.

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

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.

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 4.2-15

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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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.

<u>Loading</u>

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_o	Outside radius of attachment	*6.0 in.*
$B = \frac{.875 r_o}{R_m}$	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:

$$\tau_{shielding} = \frac{Force}{shield\ thickness \times plank\ perimeter} = \frac{442{,}080\,lb.}{8in. \times 32in.}$$
$$= 1{,}727\ psi$$

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:

$$M_{\max} = \frac{3}{2} WR^2 \text{ (maximum at section where F is applied)}$$

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:

$$\tau_{lid} = \frac{Force}{lid\ thickness \times plank\ perimeter} = \frac{442{,}080 lb.}{4.5 in. \times 32 in.} = 3{,}070\ psi$$

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:

$$P_{b_{lid}} = \frac{3F}{2\pi m t^2}\left[m + (m+1)ln\frac{a}{r_o} - (m-1)\frac{r_o^2}{4a^2}\right]$$

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:

$$F_{pressure} = 100 \frac{\pi}{4} (75.9)^2 = 452{,}453 \ lb.$$

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*

$$J = \frac{\pi r^4}{2} = 0.354^4 \text{ in torsional moment of inertia of threaded bolt}$$

$$\tau_{torsion} = \frac{6_R294 \times 0.689}{0.354} = 12{,}250 \text{ psi torsional shear}$$

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

$\tau_p = P/A = 2712/1.492 = 1821$ *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

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TABLE 4A.7-1

STRESS IN OUTER SHELL AND CLOSURE PLATES (TN-40 CASK)

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

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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 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. 01153239

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).

Protection of the public 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.

Protection of the Facility 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.

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.

Xcel Corporate Organization

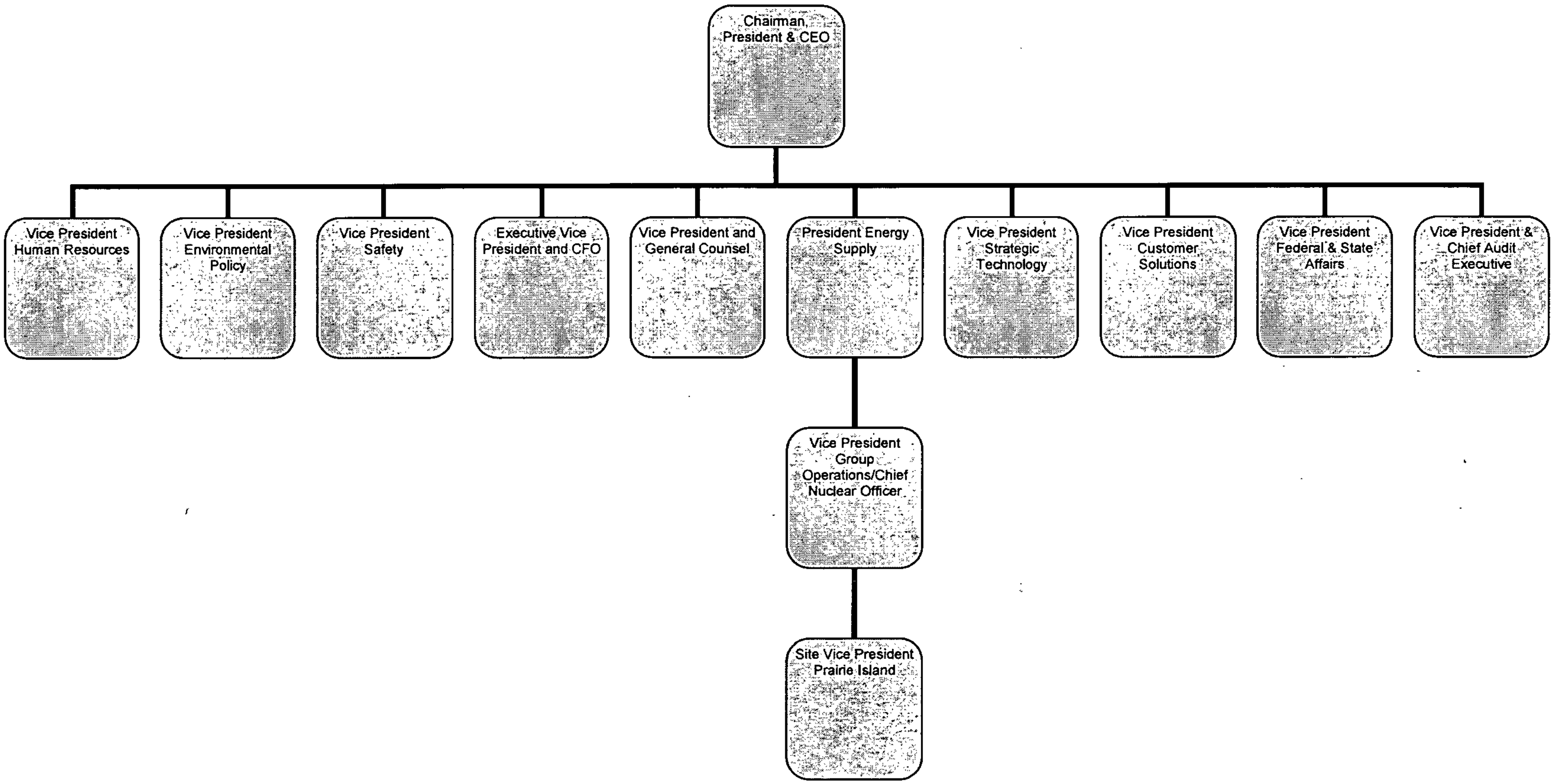

FIGURE 9.1-1, REV 12

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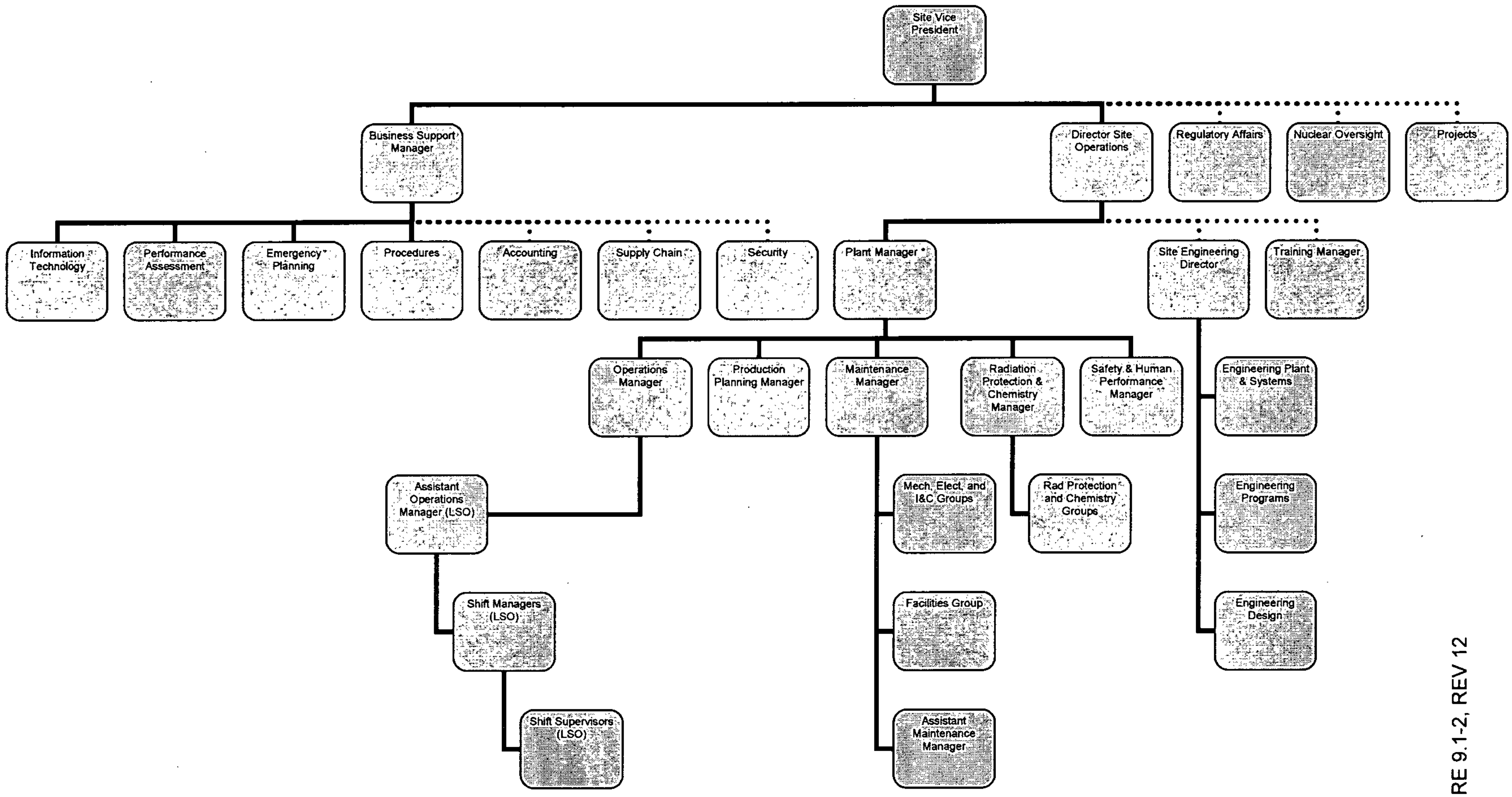
Prairie Island Plant Organization



* = 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

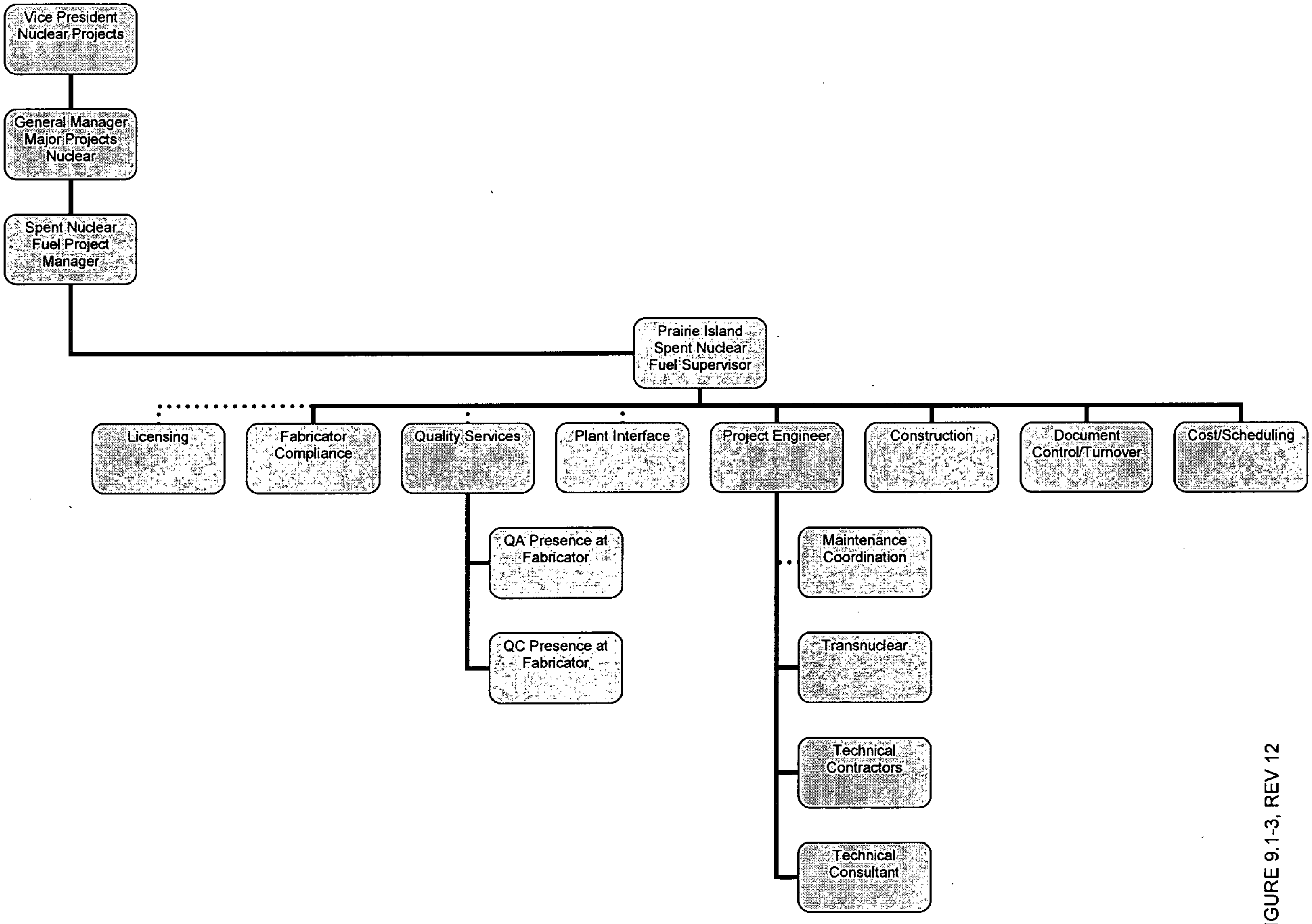

FIGURE 9.1-3, REV 12

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ISFSI Operating Organization

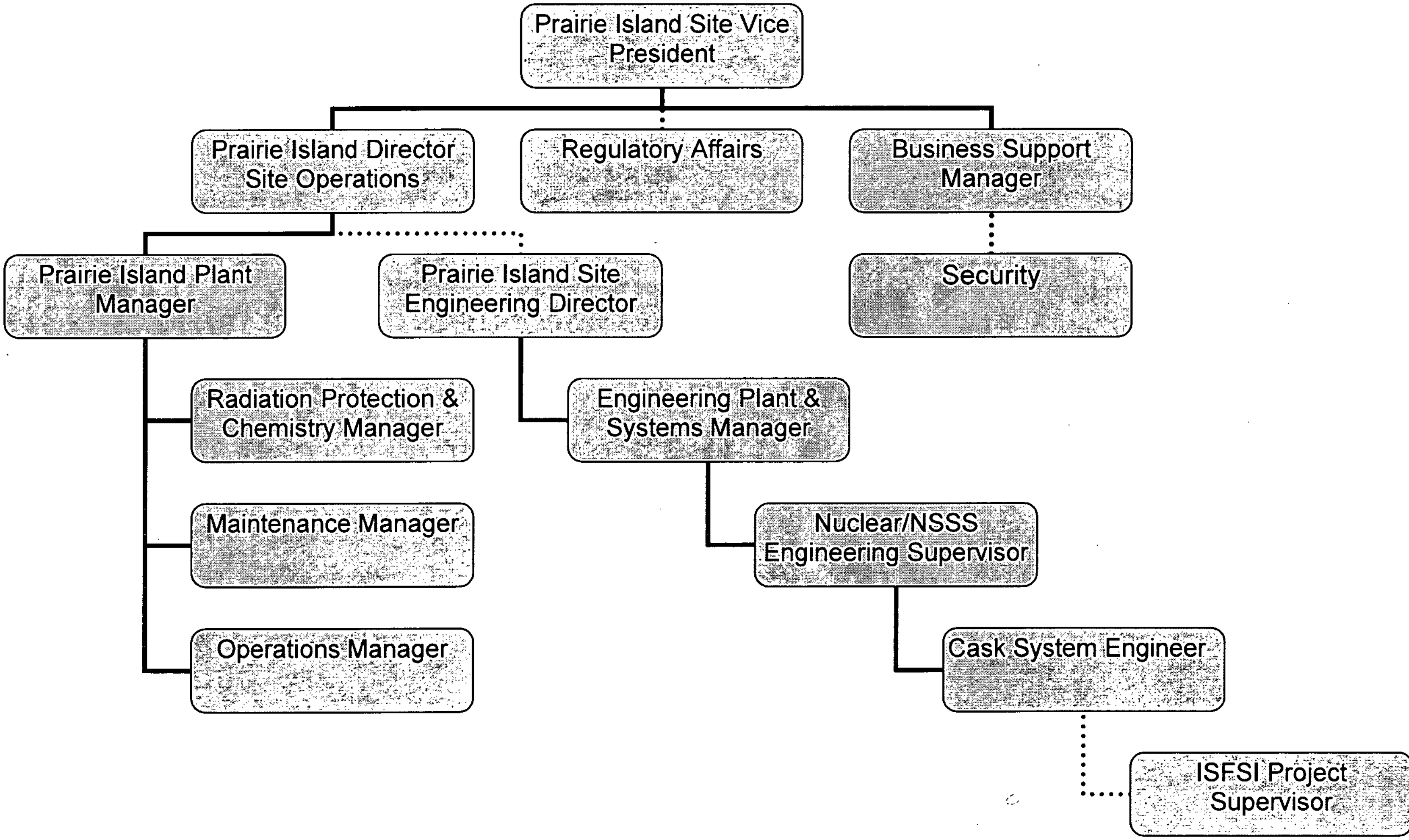

FIGURE 9.1-4, REV 12

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SECTION 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 criteria of 10CFR50, Appendix B, will be applied to activities, structures, systems, and components of the ISFSI commensurate with their importance to safety. 01153239

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 already in place. The governing document for this program is "Northern States Power Company-Minnesota, Quality Assurance Topical Report," (QATR) (Reference 1) which 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. 01153239 01201386

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 not ship the cask. 01153239

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. 01201386 01153239

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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ENCLOSURE 2

INFORMATION REGARDING CHANGES TO THE PRAIRIE ISLAND ISFSI SAR

Below is a brief description of each of the changes made to the ISFSI SAR. A summary of the safety evaluation is included for the change made without prior NRC approval pursuant to the requirements of 10 CFR 72.48. The following represent those changes listed by input item number (with which side-barred changes are denoted).

(01136263) 72.48 Evaluation No. 2952 – Transnuclear (TN) -40 Storage Cask Neutron Shield Outer Shell Structural Evaluation

Description of Change

The outer shell of the neutron shield in the TN-40 storage cask consists of a cylindrical shell section and closure plates at each end which connect the cylinder to the cask body. The change allows fabrication of the neutron shield outer shell closure plates via four segments joined with partial penetration welds of one 0.38" weld on the top plate and two 0.35" welds on the bottom plate. The structural analysis has been updated for the neutron outer shell including the segmented closure plates and two welding options for the attachment of the closure plates to the cask body partial penetration welds.

Summary of 72.48 Evaluation

The neutron shield outer shell, closure plates and attachment welds do not impact accident initiators and thus there is no impact on the frequency of occurrence of an accident. The structural analysis of the components shows that the stress limit has been satisfied, thus the proposed change does not affect the likelihood of occurrence of a malfunction, or create an accident of a different type. The components do not impact nor are involved in the Loss of Confinement Barrier accident and thus the change does not impact the consequence of an accident or malfunction. The SAR contains an evaluation of the dose rates resulting from a complete loss of the neutron shield and outer shell thus bounding any malfunction of these components. The change does not involve a design basis limit for a fission product barrier. The structural analysis utilized methods that have been previously approved by the NRC or are conservative to methods previously approved, thus there is not a departure from a method of evaluation described in the SAR.

(01145327) Screening No. 3029 - Revision of Factor of Safety for Cask Overturning and Sliding

This change to the last paragraph of Section 4.2.1.2 is to revise the Factor of Safety for Overturning and Sliding. An error in the calculation used to determine these factors has been corrected. These revisions are for the Factor of Safety for cask overturning to

change from 1.97 to 1.35 and the Factor of Safety for sliding to change from 1.2 to 1.14. These new values remain greater than the minimum values also listed in section 4.2.1.2. Also, the change does not revise the method of evaluation for these factors. Thus, design functions and method of evaluation are not adversely affected. This determination is supported by 10 CFR 72.48 Screening No. 3029.

(01153239 & 01201386) Operational Authority Transfer per License Amendment 6

A transfer of operating authority to Northern States Power Company, a Minnesota Corporation (NSPM) from Nuclear Management Company (NMC) was approved by NRC Order dated September 15, 2008. Throughout the ISFSI SAR, references to NMC were replaced with references to NSPM.