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Prairie Island Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

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10 CFR 72.48
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U S Nuclear Regulatory Commission
ATTN: Document Control Desk
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Prairie Island Independent Spent Fuel Storage Installation (ISFSI)
Docket 72-10
License No. SNM-2506

2007 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 are no new 10 CFR 72.48 Evaluations to report at this time. Revision No. 11 to the ISFSI SAR is enclosed. 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 September 28, 2007. Revision No. 11 includes a number of editorial changes. None of these changes required a 10 CFR 72.48 Evaluation.

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.

Michael D. Wadley

Michael D. Wadley
Site Vice President, Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosure

cc: Director, Spent Fuel Project Office, NRC
NMSS Project Manager, NRC
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ENCLOSURE 1

PRAIRIE ISLAND
INDEPENDENT SPENT FUEL STORAGE INSTALLATION
SAFETY ANALYSIS REPORT
REVISION 11
Revised Pages (with update instructions)

86 Pages Follow

Prairie Island Independent Spent Fuel Storage Installation
Revision 11 to the Safety Analysis Report

Instructions:

1. Remove individual ISFSI SAR pages and replace with the new Revision 11 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 4120 if you require additional assistance.

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6/11/09

Lynn R. Johnson

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4.4 OPERATING SYSTEMS

4.4.1 LOADING AND UNLOADING SYSTEMS

4.4.1.1 FUNCTION

The functions of the loading and unloading systems are to transfer the spent fuel from the spent fuel pool to the storage cask and move the storage cask from the spent fuel pool to the transport vehicle in the Auxiliary Building rail bay.

The performance objectives are to load the fuel into the casks in such a manner as to preclude damage to the fuel or criticality and to move the loaded cask in a manner which will preclude damage to the cask body and its internals and to any other safety related system or component in the load path.

4.4.1.2 MAJOR COMPONENTS AND OPERATING CHARACTERISTICS

The fuel storage area at Prairie Island Nuclear Generating Plant is located between the two reactor buildings, and consists of a new fuel pit, two pools for storing spent fuel, and a canal for transfer of fuel elements between the reactors and the pool. The two spent fuel storage pools are designated as Pool No. 1 and Pool No. 2. Pool No. 1, the smaller of the two pools, has inside plan dimensions of 18 ft.-11 in. x 18 ft.-3 in. Pool No. 2 has inside plan dimensions of 18 ft.-11 in. x 43 ft.-5 in. Normal water depth for both pools is about 40 ft. The southeast corner of Pool No. 1 is designated as the cask loading and unloading area. Five racks in Pool No. 1 could be used for fuel storage during loading of the spent fuel cask.

The spent fuel pool area is surrounded by a reinforced concrete enclosure. Access into the enclosure for the spent fuel cask is provided by a door and a narrow slot in the ceiling for attaching the spent fuel cask to the overhead Auxiliary Building crane which physically restricts cask movement to the north-south path over the cask set down area of Pool No. 1. Figure 4.4-1 is a layout drawing showing the cask load path between the spent fuel pool enclosure and the Auxiliary Building rail bay. Figure 4.4-2 and 4.4-3 are section views showing the cask load path.

4.4.1.3 SAFETY CONSIDERATION AND CONTROLS

Fuel handling activities in the spent fuel pool are subject to limiting conditions for operation as set forth in Section 3.7 of the Prairie Island Technical Specifications. A single-failure-proof crane was installed for cask handling in the spent fuel pool enclosure and Auxiliary Building rail bay. This crane and the fuel loading and cask handling is subject to conditions as set forth in the Prairie Island Technical Specifications and Operating License.

4.4.2 DECONTAMINATION SYSTEM

Standard decontamination methods will be used to remove surface contamination from the casks resulting from their submersion in the spent fuel pool during fuel loading. Decontamination of the casks will be performed in the cask decontamination area, located in the rail bay of the Auxiliary Building. Decontamination will be done manually, using water detergents and wiping cloths.

4.4.3 STORAGE CASK REPAIR AND MAINTENANCE

Maintenance on the casks can be performed as described in Section 5.1.3.3.

4.4.4 UTILITY SUPPLIES AND SYSTEMS

The storage casks are passive devices. No utility services are needed for operation of the casks.

4.4.5 OTHER SYSTEMS

4.4.5.1 ELECTRICAL SYSTEMS

Non-safety related electrical power is provided to the ISFSI for lighting, general utility, and pressure monitoring instrumentation purposes. Electrical power is provided from a new pole mounted transformer and an existing overhead distribution line.

4.4.5.2 ALARM SYSTEM

Cask interseal pressure will be monitored. The pressure monitoring devices will provide an analog input signal to actuate alarm indication at a monitoring panel outside the ISFSI gate.

4.4.5.3 FIRE PROTECTION SYSTEM

No fires other than small electrical or tow vehicle fuel fires are considered credible at the ISFSI. Accordingly, only portable fire extinguishers are provided. Smoke detectors are installed in the ISFSI buildings to alert operators if a fire is started. The fire fighting equipment at the Prairie Island Nuclear Generating Plant is available, if needed.

4.4.5.4 VACUUM SYSTEMS

A Vacuum Drying System will be used to remove residual water left in the cask after it has been removed from the fuel pool and drained. The system applies vacuum at the vent port vaporizing any water present and sweeping the water out of the cask. The drying time is approximately 12 to 16 hours.

A Vacuum Backfill System will be used to replace air in the cask with helium. The system applies vacuum at the vent port and evacuates the cask cavity to 10 millibar. Once evacuated, the system backfills the cavity with dry helium gas. 01106798

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4.5 CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

The structures, components and systems of the TN-40 Dry Storage Cask and storage facility are classified as Safety Related, Augmented Quality, or Non QA Related. A tabulation of the structures, components, and systems by their classification is shown in Table 4.5-1. The criteria for selecting the classification for particular structures, components, or systems, are based on the following definitions given in 5AWI 7.1.1 (Reference 18).

Safety Related Item

Any structure, system or component that prevents or mitigates the consequences of postulated nuclear accidents that could cause undue risk to the health and safety of the public. Any component whose failure would produce radiation levels at the site boundary in excess of 10CFR100 limits is classified as safety related.

Augmented Quality

Augmented quality is a procurement classification for items or services which do not perform a safety related function, but are subject to special utility requirements or NRC imposed regulatory requirements. This includes items and services classified as non-safety QA related, fire protection related, 10CFR71 related, security related and other applicable site specific items and services. The difference between safety related and augmented quality items and services is that 10CFR50 Appendix B and 10CFR21 requirements do not apply to items and services purchased as augmented quality.

Non QA Related (Standard Quality)

This is a procurement quality classification for items or services which do not have a safety related function and are not subject to special utility requirements or NRC imposed regulatory requirements.

When purchasing safety related or augmented quality items and services, the procurement documents must contain the technical requirements, as well as the source of those requirements. These requirements must be verified by means such as vendor supplied documentation, receipt inspection, or testing. This process is similar for both safety and augmented quality items.

All items related to the storage cask, regardless of their classification, are designed in accordance with the requirements of the TN-40 Design Criteria which ensure that the General Design Criteria of 10CFR72(F), are satisfied. Those items related to the storage cask which are classified as safety related are designed, fabricated, inspected and tested, to the maximum practicable extent, in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB. The lifting trunnions are designed to ANSI N14.6 as described in Appendix 4A. Those items related to the concrete storage pads which are classified as safety related are designed, installed, inspected and tested in accordance with the specific requirements of the American Concrete Institute (ACI) and American Society for Testing and Materials (ASTM).

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The quality assurance requirements of 10CFR50 Appendix B to are applied.

Those items which are classified as augmented quality or non QA related are designed in accordance with design rules which are indicated in the structural analysis of those items in Section 4.2.

4.5.1 CONTAINMENT VESSEL

The containment vessel and trunnions are classified as safety related since they serve as the primary confinement structure for the fuel assemblies and are designed to remain intact under all accident conditions analyzed in Chapter 8. The basket is classified as safety related because it provides criticality control, as well as serving as the structural support for the fuel and is designed to remain intact during all of the accidents described in Chapter 8.

4.5.2 PENETRATION GASKETS

Cask interseal pressure will be monitored. The pressure monitoring devices will provide an analog input signal to actuate alarm indication at a monitoring panel outside the ISFSI gate.

4.5.3 SHIELDING

The neutron shield body shield and lid shield are classified as augmented quality items. The basis for this classification is that they perform no function required by the accident analysis in Chapter 8 but they do provide radiation protection for personnel and are related to ISFSI Technical Specification requirements.

4.5.4 PROTECTIVE COVER AND OVERPRESSURE SYSTEM

The protective cover and overpressure system serve no safety function and are therefore classified as non QA related items.

4.5.5 CONCRETE STORAGE PADS

The concrete and reinforcing steel in the concrete storage pads are classified as safety related and are Seismic Category 1. The concrete storage pads provide structural support for the storage casks and are designed to prevent the failure of the casks due to an accident described in Chapter 8 while being moved or stored at the ISFSI.

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TEMPERATURE DEPENDENT MATERIAL PROPERTIES
COEFFICIENTS OF THERMAL EXPANSION (NOTE 1)

	TEMPERATURE, °F										
MATERIAL	**100**	**150**	**200**	**250**	**300**	**350**	**400**	**450**	**500**	**550**	**600**
SA-350,SA-320,SA-203 (Note 2)	6.27	6.41	6.54	6.65	6.78	6.88	6.98	7.07	7.16	7.24	7.32
SA-105 & SA-516, Gr 55 (Note 2)	5.73	5.91	6.09	6.27	6.43	6.57	6.74	6.89	7.06	7.18	7.28
SA-516, Gr 70 (Note 3) (Note 4)	5.53	5.71	5.89	6.09	6.26	6.43	6.61	6.77	6.91	7.06	7.17

NOTES

1. Values listed are the mean coefficients of thermal expansion x 10^{-6} (in./in. °F) from 70° F to the indicated temperature.

2. Source of data is ASME Section III, Appendix I, Table I-5.0, p. 123-128.

3. Optional plate material used as a substitute for cask shell, bottom and shield plate SA-105 forged material.

4. Source of data is ASME Section II, Part D, Subpart 2, Table TE-1, p. 638, 1992.

TABLE 4A.2-2
SHEET 2 OF 3

TEMPERATURE DEPENDENT MATERIAL PROPERTIES
MODULI OF ELASTICITY, E (NOTE 1)

	TEMPERATURE, °F					
MATERIAL	**70**	**200**	**300**	**400**	**500**	**600**
SA-203, SA-320, SA-350 (Note 2)	27.8	27.1	26.7	26.1	25.7	25.2
SA-516 Grade 55, SA-105 (Note2)	29.5	28.8	28.3	27.7	27.3	26.7
SA-516 Grade 70 (Note 3) (Note 4)	29.3	28.6	28.1	27.5	27.1	26.5

NOTES

1. Values listed are the moduli of elasticity x 10^6 psi for the indicated temperature.
2. Source of data is ASME Section III, Appendix I, Table I- 6.0, p. 129-138.
3. Optional plate material used as a substitute for cask shell, bottom and shield plate SA-105 forged material.
4. Source of data is ASME Section II, Part D, Subpart 2, Table TM-1, p.664, 1992

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TABLE 4A.2-2
SHEET 3 OF 3

TEMPERATURE DEPENDENT MATERIAL PROPERTIES DESIGN STRESS PARAMETER

MATERIAL	STRESS PARAMETER (NOTE 1)	TEMPERATURE, °F 100	200	300	400	500	600	DATA SOURCE (NOTE 2)
SA-350, Grade LF3, SA-203, Grade E	S	17.5	17.5	17.5	17.5	17.5	17.5	Table I-7.1 p. 140
SA-203, Grade A and D	S	16.2	16.2	16.2	16.2	16.2	16.2	Table I-7.1 p. 140
SA-320, Grade L43	S_y	105.0	99.0*	95.7*	91.8*	88.5*	84.3*	Table I-1.3 p. 42. *(Note 3)
SA-105	S_m	23.3	21.9	21.3	Not Required			Table I-1.1 p. 9
	S_y	36.0	32.8	31.9				Table I-2.1 p. 52
	S_u	70.0	70.0	70.0				Table I-3.1 p.85
SA-516, Grade 55	S_y	30.0	27.3	26.6	25.7			Table I-2.1 p. 50
	S_m	13.7	13.7	13.7	13.7			Table I-7.1 p, 132
SA-516, Grade 70 (Note 4)	S_m	23.3	23.1	22.5	Not Required			Table 2A p. 300
	S_y	38.0	34.6	33.7				Table y-1 p. 516
	S_u	70.0	70.0	70.0				Table U p. 481 (Note 5)

NOTES

1. Values listed are the stress parameters which form the basis for structural analysis acceptance criteria.
   S refers to the ASME allowable stress for Class 2 or Class 3 components,
   Sm refers to the ASME design stress intensity for Class ! components, and
   Sy refers to minimum yield strength.
2. Data are taken from ASME Section III, Appendix I as noted.
3. For bolting materials, Sy≥ 3 Sm. Optional plate material used as a substitute for cask shell, bottom and shield plate SA-105 forged material.
4. Optional plate material used as a substitute for cask shell, bottom and shield plate SA-105 forged material.
5. Data is taken from ASME Section II, Part D, Subpart 1, 1992 as noted.

SECTION 5

STORAGE SYSTEM OPERATIONS

5.1 OPERATION DESCRIPTION

5.1.1 NARRATIVE DESCRIPTION

Fuel handling and cask loading operations in the Auxiliary Building will be done in accordance with requirements of the Prairie Island Nuclear Generating Plant Operating License. Cask transport and storage at the ISFSI will be subject to requirements of the ISFSI Operating License. Operating activities in the Auxiliary Building and during transport and storage are described below.

The fuel storage operation will commence with the cask being brought into the Auxiliary Building rail bay through the west roll-up door. The cask weather cover and the cask lid will be removed and the cask inspected. A lifting yoke will be attached to the cask and 01106798
connected to the Auxiliary Building crane hook. The cask will be laterally transferred and lifted by the crane from the basemat, elevation 695 feet, through a large opening in the floor slab at the 755 ft. elevation, laterally transferred and aligned with the access door to the fuel pool area, moved directly north to above the loading and unloading area of Pool No. 1 and lowered to the surface of the pool. The load path for the cask is illustrated in Figures 4.4-1 through 4.4-3. The narrow slot in the ceiling of the pool enclosure prohibits any movement of the cask except in the north-south direction.

The cask containment lid will be removed and the cask will be lowered into the spent fuel pool. Fuel assemblies will be loaded into the cask using a long handled tool suspended from the spent fuel pool bridge crane hoist and manipulated by an operator standing on the movable bridge over the pool.

After the cask is loaded with spent fuel and the lid is placed on the cask, the cask will be lifted to the pool surface and the lid bolts will be installed. The internal fuel cavity will be drained by displacing the water with air or with a suitable drain pump.

The cask will be returned to the Auxiliary Building rail bay by retracing the load path described above. The cask will be decontaminated in the Auxiliary Building rail bay/cask decontamination area and will be dried by using a vacuum system. The cavity 01106798
will be filled with helium to design pressure and the cask lid seal will be leak tested. The top neutron shield will be installed on the lid. The overpressure monitoring system will be installed, and the interspaces between the double metallic seals pressurized to equilibrium pressure. Prior to transfer from the Auxiliary Building to the ISFSI, the cask will be monitored for contamination, temperature, radiation dose rates and the proper functioning of the seal tightness monitoring system. The protective cover will be installed and the pressure transducer connections fed through the external fitting. 01106798

The transport vehicle will be pulled to the ISFSI site by a tow vehicle as described in Section 5.4. The cask will be set in its storage position. The cask will be connected to the cask seal monitoring system and a functional check of the monitoring system will be performed.

5.1.2 FLOW SHEETS

The sequence of operations performed in loading fuel into the TN-40 storage cask and placing the cask into storage at the ISFSI is outlined in Table 5.1-1 and is shown in simplified flowsheet form in Figure 5.1-1.

Details of the number of personnel and the time required for the various operations are given in Table 5.1-2 for use in the radiation exposure determinations developed in Chapter 7. The data are based on Transnuclear's experience with transport cask operations.

5.1.3 IDENTIFICATION OF SUBJECTS FOR SAFETY AND RELIABILITY ANALYSIS

5.1.3.1 CRITICALITY PREVENTION

As discussed in Section 3.3.4, criticality is controlled by utilizing poison materials in the fuel basket and in the spent fuel pool water. These features are only necessary underwater during cask loading in the spent fuel pool. During storage, with the cavity dry and sealed from the environment, no further criticality control measures within the installation are necessary because of the low reactivity of the dry cask and the assurance that no unborated water can enter the cask during storage.

5.1.3.2 INSTRUMENTATION

Due to the totally passive and inherently safe nature of the storage casks, there is no need for any instrumentation to perform safety functions. However, transmitters are utilized to monitor the cask seals for leakage and are described in Section 3.3.3. The transmitters monitor the pressure in an interspace between the double metallic seals to provide an indication of seal failure before any release occurs.

An initial function check is performed at the manufacturer's plant and another function check of the transmitters is performed in preparation for cask storage.

5.1.3.3 MAINTENANCE TECHNIQUES

Because of their passive nature, the storage casks will require little, if any, maintenance over the lifetime of the ISFSI. Typical maintenance tasks involve occasional replacement and recalibration of monitoring instrumentation and recoating of some casks with corrosion-inhibiting coatings. No special maintenance techniques are necessary.

TABLE 5.1-1

SEQUENCE OF OPERATIONS

A. Receiving

1. Unload empty cask and separately packaged seals at plant site.
2. Inspect the following for shipping damage: exterior surfaces, sealing surfaces, trunnions, seals, accessible interior surfaces and basket assembly, bolts, bolt holes and threads, neutron shield vents.
3. Remove weather shield and install plug in neutron shield vent hole (threaded hole in the top of the steel shell surrounding the resin which contains a pressure relief valve during storage).
4. Remove lid bolts and lid.
5. Install protective plate over cask body sealing area.
6. Obtain lid and lid seal from storage.
7. Attach lid seal to lid by means of six retaining screws.
8. Move to spent fuel pool area.

B. Spent Fuel Pool Area

1. Lower cask into cask loading pool.
2. Load preselected spent fuel assemblies into the 40 basket compartments.
3. Verify identity of the fuel assemblies loaded into the cask.
4. Remove protective plate from cask body flange.
5. Lower lid and place on cask body flange over the two alignment pins.
6. Lift cask to surface of pool and install some lid bolts.
   Note: Steps B.7 through B.11 may use a drain pump with a quick-disconnect coupling or suction lance to drain water as an alternate method.
7. Connect drain line to quick-disconnect coupling in the drain port.
8. Bolt special adapter, with quick-disconnect coupling, to vent port bolt holes.

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TABLE 5.1-1

PAGE 2 OF 3

SEQUENCE OF OPERATIONS

9. Connect plant compressed air line to special adapter quick-disconnect coupling.

10. Pressurize cavity to force water from cavity through drain port to the spent fuel pool.

11. Disconnect plant compressed air line and drain line from their quick-disconnect couplings.

12. Move cask to the decontamination area.

C. Decontamination Area (Rail Bay)

1. Decontaminate cask until acceptable surface dose levels are obtained.

2. Install remaining lid bolts and torque lid bolts using the prescribed procedure. 01106798

3. Remove plug from neutron shield vent and install pressure relief valve.

4. Connect Vacuum Drying System (VDS) to vent port.

5. Evacuate cavity to remove remaining moisture using prescribed procedure.

6. Evacuate cavity to 10 millibar and backfill with dry helium gas.

7. Pressurize cavity to about 2 atm with helium.

8. Disconnect VDS at vent port and install vent port cover with seal and bolts.

9. Perform helium leak test of lid seals.

10. Remove overpressure test connector.

11. Load cask on transport vehicle.

12. Check external surface temperatures using infrared camera or equivalent. 01106798

13. Install top neutron shield drum.

14. Torque the bolts using prescribed procedure.

15. Perform leak test on overpressure system.

TABLE 5.1-1

SEQUENCE OF OPERATIONS

16. Pressurize overpressure system (seal interspaces) with helium to a pressure of about 5.5 atm. 01106798

17. Check surface radiation levels.

18. Install protective cover with seal and bolts (could be performed at storage area).

19. Move cask to storage area.

D. Storage Area

1. Position cask in preselected location on storage pad.

2. Unload cask from transport vehicle. 01106798

3. Check for surface defects.

4. Connect pressure instrumentation to cask and to monitoring panel.

5. Check that pressure instrumentation is functioning.

6. Check surface radiation levels.

TABLE 5.1-2

ANTICIPATED TIME AND PERSONNEL REQUIREMENTS FOR CASK HANDLING OPERATIONS

OPERATION	NO. OF PERSONNEL	TIME (MIN)	AVG. DISTANCE (ft) FROM CASK
RECEIVING (Table 5.1-1 [A1 – A8])			
1. Unloading	*	*	*
2. Inspection	*	*	*
3. Transfer to cask	*	*	*
CASK LOADING POOL (Table 5.1-1 [B1 – B12])			
4. Lower cask into pool	*	*	*
5. Load fuel	5	*	*
6. Place lid on cask	5	*	*
7. Lift cask to pool surface	5	30	5
8. Install lid bolts	5	120	3
9. Drain cavity	5	90	6
10. Transfer to decontamination area	3	60	10
DECONTAMINATION AREA (RAIL BAY) (Table 5.1-1 [C1 – C19])			
11. Decontaminate cask	3	120	3
12. Remove vent plugs	2	30	5
13. Drying, evacuating, backfilling	2	480	5

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TABLE 5.1-2

ANTICIPATED TIME AND PERSONNEL REQUIREMENTS FOR CASK HANDLING OPERATIONS

OPERATION	NO. OF PERSONNEL	TIME (MIN)	AVG. DISTANCE (ft) FROM CASK
14. Install top neutron shield	2	15	3
15. Install pressure transducers	2	30	5
16. Pressurize interspace	*	*	*
17. Check leakage	2	30	5
18. Check surface temperature	2	30	5
19. Check surface dose rate	2	30	3
20. Install protective cover	2	30	5
21. Load transport vehicle	3	60	5
22. Transfer to storage area	3	60	10
STORAGE AREA (Table 5.1-1 [D1 – D6])			
23. Unload from vehicle position in location	5	60	5
24. Check surface dose rate	5	30	3
25. Connect pressure instrumentation	5	30	5

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TABLE 5.1-2

PAGE 3 OF 3

ANTICIPATED TIME AND PERSONNEL REQUIREMENTS FOR CASK HANDLING OPERATIONS

OPERATION	NO. OF PERSONNEL	TIME (MIN)	AVG. DISTANCE (ft) FROM CASK
PERIODIC MAINTENANCE			
1. Visual surveillance (NA)	2	15	5
2. Repair surface defects (NA)	2	60	3
3. Instrument testing and calibration	2	180	5
4. Instrument repair (NA)	2	60	3
MAJOR MAINTENANCE (ONCE IN 20 YEARS)			
1. Replace cask lid seals	3	1950 **	8

* No measurable dose associated with this activity. Therefore, the number of personnel, time and distance are not significant.

\# Parenthetical information corresponds to Table 5.1-1 activity numbers.

** Total time to transfer cask to spent fuel pool, replace lid seals, and return cask to ISFSI pad.

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. NMC 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.

The ALARA program ensures that:

1. An effective ALARA program is administered at the Prairie Island Nuclear Generating Plant that appropriately integrates management philosophy and NRC regulatory requirements and guidance.
2. Facility design features, operating procedures and maintenance practices are in accordance with ALARA program guidelines; and that written reviews of the on-site radiation protection program assure that objectives of the ALARA program are attained.
3. Pertinent information concerning radiation exposure of personnel from other utilities and research work are reflected in design and operation.
4. Appropriate experience gained during the operation of nuclear power stations relative to in-plant radiation control is factored into revisions of procedures to assure that the procedures continually meet the objectives of the ALARA program.
5. Necessary assistance is provided to insure that operations, maintenance, and decommissioning activities are planned and accomplished in accordance with ALARA objectives.
6. Trends in station personnel and job exposures are analyzed in order to permit corrective actions to be taken with respect to adverse trends.

Reports of the findings of the radiation protection staff are also effectively conveyed to management.

Specific responsibilities of station personnel are to ensure that:

1. Activities are planned and accomplished in accordance with the objectives of the ALARA program.

2. Procedures and their revisions are implemented in accordance with the objectives of the ALARA program.

3. The general office radiation protection staff is consulted as necessary for assistance in meeting ALARA program objectives.

The primary goal of the radiation protection and ALARA programs is to minimize exposure to radiation such that the total individual and collective exposure to personnel in all phases of design, construction, operation and maintenance are kept As Low As Reasonably Achievable. This is achieved by integrating ALARA concepts into design, construction, and operation of facilities.

Trained personnel adequate to develop and conduct all necessary radiation protection and ALARA programs are provided. These personnel are trained to assure that all procedures are followed to meet company and regulatory requirements. Training programs in the basics of radiation protection and exposure control are provided to all facility personnel whose duties require working in radiation areas.

The administrative organization is responsible for and has appropriate authority for assuring that the three basic objectives of the radiation protection program are achieved. These objectives are to:

1. Protect personnel

2. Protect the public

3. Protect the facility

<u>Protection of Personnel</u> Includes surveillance and control over internal and external radiation exposure and maintaining the exposure of all personnel within permissible limits and as low as reasonably achievable (ALARA).

<u>Protection of the public</u> includes surveillance and control over all conditions and operations that may affect the health and safety of the public. Included are such activities as radioactive gas, liquid, and solid waste disposal, shipment of radioactive materials, an environmental radioactivity monitoring plan and maintaining portions of the station emergency plan.

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. 01106798

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. 01106798

7.1.2 DESIGN CONSIDERATIONS

The equipment design takes into account radiation protection considerations, which ensure that occupational radiation exposures are ALARA. The fuel will be stored dry, inside sealed, heavily-shielded casks. The most significant radiation protection design consideration provides for heavy shielding to minimize personnel exposures. To avoid personnel exposure, the casks will not be opened nor fuel removed from the casks while at the ISFSI. Storage of the fuel in dry sealed casks eliminates the possibility of leakage of contaminated liquids. Gaseous releases are not considered credible. The exterior of the casks will be decontaminated before leaving the Auxiliary Building, thereby minimizing exposure of personnel to surface contamination. The storage casks will contain no active components which require periodic maintenance or surveillance. This method of spent fuel storage minimizes direct radiation exposures and eliminates the potential for personnel contamination.

Both concrete storage pads and the Equipment Storage Building at the ISFSI will be constructed prior to ISFSI operation. This will be done to eliminate occupational radiation exposure which would result from additional construction following placement of storage casks in the ISFSI.

An annunciator panel monitoring cask pressure will be located outside of the ISFSI protected area. This will minimize time required for periodic cask surveillance and reduce personnel exposure.

The ISFSI site is within the exclusion area of the Prairie Island site. The location of the ISFSI is of sufficient distance from frequently occupied areas of the Prairie Island Nuclear Generating Plant such that the increased dose to personnel will not be significant.

Regulatory Position 2 of Regulatory Guide 8.8, is incorporated into design considerations, as described below:

- ALARA objective 2a on access control is met by use of a fence with a locked gate that surrounds the ISFSI and prevents unauthorized access.

- Regulatory Position 2b on radiation shielding is met by the heavy shielding of the casks which minimizes personnel exposures.

- Regulatory Position 2c on process instrumentation and controls is met by designing the instrumentation for a long service life and locating readouts in a low dose rate location.

- Regulatory Position 2d on control of airborne contaminants does not apply because no gaseous releases are expected. No significant surface contamination is expected because the exterior of the casks and racks will be decontaminated before they leave the decontamination area in the Auxiliary Building.

- Regulatory Position 2e on crud control is not applicable to the ISFSI because there are no systems at the ISFSI that could transport crud.

- Regulatory Position 2f on decontamination is met because the exteriors of the casks are designed for decontamination. The casks and racks are decontaminated before they are released from the decontamination area in the Auxiliary Building.

- Regulatory Position 2g on radiation monitoring does not apply because the casks are sealed. There is no need for airborne radioactivity monitoring since no airborne radioactivity is anticipated. Area radiation monitors will not be required because the ISFSI will not normally be occupied; however, TLDs will be installed along the controlled access fence.

- Regulatory Position 2h on resin treatment systems is not applicable to the ISFSI because there will be no radioactive systems containing resins.

- Regulatory Position 2i concerning other miscellaneous ALARA items is not applicable because these items refer to radioactive systems not present at the ISFSI.

7.1.3 OPERATIONAL CONSIDERATIONS

The ALARA procedures for the ISFSI will be the same as those used in the radiation protection program for Prairie Island Nuclear Generating Plant. Section 7.1.1 describes the policy and procedures that ensure that ALARA occupational exposures and contamination levels are achieved. Section 7.1.2 describes how the design considerations are ALARA.

Storage of spent fuel in storage casks is expected to involve lower exposures than other alternative methods or designs for onsite storage. For example, storage in a fuel pool would involve use of radioactive water cooling and cleanup systems and filtered HVAC that would result in higher operator exposures during pump, valve, and motor maintenance of these systems, and filter and resin replacement. This alternative would also lead to additional airborne and liquid releases that will not be present at the ISFSI.

Operational requirements for surveillance are incorporated into the design considerations in Section 7.1.2 in that the casks are stored with adequate spacing to allow ease of on site surveillance. In addition, annunciation will be available outside the ISFSI protected area to minimize surveillance time. The operational requirements are incorporated into the radiation protection design features described in Section 7.3 since the casks are heavily shielded to minimize occupational exposure.

The ISFSI contains no systems that process liquids or gases or contain, collect, store, or transport radioactive liquids or solids other than the stored spent fuel and contaminated spent fuel racks. Therefore, the ISFSI meets ALARA requirements since there are no such systems to be maintained, be repaired, or be a source of leaks.

7.6 RADIATION PROTECTION PROGRAM

The ISFSI is located on the site of the Prairie Island Nuclear Generating Plant within the Owner Controlled Area. The Radiation Protection and Chemistry Manager will have responsibility for radiation protection activities at the ISFSI. 01106798

The Radiation Protection and ALARA programs are discussed in Section 7.1.1.

Radiation protection requirements for all radiological work at the Prairie Island Nuclear Generating Plant are governed by existing station directives, and station Radiation Protection procedures. Radiation protection practices for cask loading, transfer, storage, monitoring, and retrieval will also be based on existing procedures, as well as on current and anticipated conditions when the task is to be performed. These procedures include, but are not limited to, the following:

- Procedure for personnel dosimetry issue.
- Issuance, revision, and termination of radiation work permits and standing radiation work permits.
- Procedure for roping off, barricading, and posting of radiation control zones.
- Decontamination procedure for equipment and areas.
- Smear swab sampling, counting, and calculation.
- Procedure for quantifying airborne radioactivity.

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8.2 ACCIDENTS

Accidents are design events of the third and fourth type (Design Events III and IV) as defined in ANSI/ANS 57.9. Design Event III consists of that set of infrequent events that could reasonably be expected to occur during the lifetime of the ISFSI.

Design Event IV consists of the events that are postulated because their consequences may result in the maximum potential impact on the immediate environs. Their consideration establishes a conservative design basis for certain systems with important confinement features.

8.2.1 EARTHQUAKE

8.2.1.1 CAUSE OF ACCIDENT

The design earthquake (DE) is postulated to occur as a design basis extreme natural phenomenon.

8.2.1.2 ACCIDENT ANALYSIS

Seismic response characteristics of the storage casks are provided in Section 3.2.3 and 4.2.3. Results of these analyses show that cask leak-tight integrity is not compromised and that no damage will be sustained.

8.2.1.3 ACCIDENT DOSE CALCULATIONS

The DE is not capable of damaging the cask. Hence, no radioactivity is released and there is no associated dose.

8.2.2 EXTREME WIND

8.2.2.1 CAUSE OF ACCIDENT

The extreme winds due to passage of the design tornado as defined in Section 3.2.1 are postulated to occur as an extreme natural phenomenon.

8.2.2.2 ACCIDENT ANALYSIS

The effects and consequences of extreme winds on the casks are presented in Section 3.2.1.

8.2.2.3 ACCIDENT DOSE CALCULATIONS

Extreme winds are not capable of overturning these casks nor of damaging their seals. Since no radioactivity is released, no resultant doses will occur.

Local damage to the neutron shield may be caused by tornado missiles. However, Table 7A-4 shows that the dose rate without any shield to be less than the allowable accident dose rate.

8.2.3 FLOOD

8.2.3.1 CAUSE OF ACCIDENT

The probable maximum flood has been calculated to reach a level of 703.6 ft., with wave action to a maximum level of 706.7 ft.

8.2.3.2 ACCIDENT ANALYSIS

The casks are designed to withstand the forces developed by the probable maximum flood without damage to cask integrity or tipping of the casks. The height of the cask seals will be above the level of the probable maximum flood and associated wave action. Accordingly no fuel damage or criticality is postulated to occur as a result of flooding. Analyses are contained in Section 3.2.2.

8.2.3.3 ACCIDENT DOSE CALCULATIONS

The probable maximum flood is not capable of overturning the casks or of damaging their seals. Therefore, no resultant doses are projected.

8.2.4 EXPLOSION

8.2.4.1 CAUSE OF ACCIDENT

A munition barge explosion has been postulated to occur at a location approximately 2600 feet from the ISFSI. This occurrence is described in detail in Section 2.2. The pressure wave of 2.25 psi is estimated to occur at the ISFSI.

8.2.4.2 ACCIDENT ANALYSIS

The cask accident analysis includes the consideration for a 2.25 psi overpressure from the postulated explosion near the ISFSI location, as described in Section 3.2.5.3.4. 01106798

8.2.4.3 ACCIDENT DOSE CALCULATIONS

The cask will not tip as a result of the postulated pressure wave. Accordingly, no cask damage or release of radioactivity is postulated. Since no radioactivity is released, no resultant doses would occur.

8.2.5 FIRE

8.2.5.1 CAUSE OF ACCIDENT

The only combustible materials in the ISFSI are in the form of insulation on instrumentation wiring, and paint on the outside surface of the storage casks. In addition, the tow vehicle will contain a small amount of gasoline or diesel fuel. No other combustible or explosive materials are allowed to be stored on the ISFSI slabs. The ISFSI area is cleared of trees. The entire area surrounding the Equipment Storage Building and concrete pad within the perimeter road is covered with crushed rock. In addition, other equipment in the area are adequately separated from the ISFSI slabs. Therefore, no fires other than small electrical fires are considered credible at the ISFSI.

8.2.5.2 ACCIDENT ANALYSIS

The ability of the cask to withstand postulated fires is addressed in Section 3.2.5.2.9 and 3.3.2.2.2.

8.2.5.3 ACCIDENT DOSE CALCULATIONS

Since no activity is released, no resultant doses would occur.

8.2.6 INADVERTENT LOADING OF A NEWLY DISCHARGED FUEL ASSEMBLY

8.2.6.1 CAUSE OF ACCIDENT

The possibility of a spent fuel assembly, with a heat generation rate greater than 0.675 kw, being erroneously selected for storage in a cask has been considered. The cause of this accident is postulated to be an error during the loading operations, e.g., wrong assembly picked by the fuel handling crane, or a failure in the administrative controls governing the fuel handling operations.

8.2.6.2 ACCIDENT ANALYSIS

The fuel assemblies require several years of storage in the spent fuel pool before the heat generation decays to a rate below 0.675 kw. This accident scenario postulates the inadvertent loading of an assembly not intended for storage in the storage canister, with a heat generation rate in excess of the design basis specified in Section 3.1.1 and 10.1.1.

In order to preclude this accident from going undetected, and to ensure that appropriate rectification actions can take place prior to the sealing of the casks, a final verification of the assemblies loaded into the casks and a comparison with fuel management records will be performed to ensure that the loaded assemblies do not exceed any of the specified limits.

These administrative controls and the records associated with them will be included in the procedures described in Chapter 9 and in the proposed license requirements described in Chapter 10, and will comply with the applicable requirements of the Quality Assurance Program described in Chapter 11.

Therefore, appropriate and sufficient actions will be taken to ensure that an erroneously loaded fuel assembly does not remain undetected. In particular, the storage of a fuel assembly with a heat generation in excess of 0.675 kw is not considered credible in view of the multiple administrative controls.

8.2.6.3 ACCIDENT DOSE CALCULATIONS

The inadvertent loading of a fuel assembly not intended for storage in a storage cask is not considered to be a credible occurrence. Therefore, no doses are postulated.

8.2.7 CASK SEAL LEAKAGE

The storage casks feature redundant seals in conjunction with an extremely rugged body design. Additional barriers to the release of radioactivity are presented by the sintered fuel pellet matrix and the Zircaloy cladding which surrounds the fuel pellets. Furthermore, the interseal gaps are pressurized in excess of the cask cavity. As a result, no credible mechanisms that could result in leakage of radioactive products have been identified. Nevertheless, a complete loss of the storage cask confinement capability is postulated in Section 8.2.9, and the results found to be negligible.

8.2.8 HYPOTHETICAL CASK DROP ACCIDENT

8.2.8.1 CAUSE OF ACCIDENT

The stability of the TN-40 storage cask in the upright position on the ISFSI concrete storage pad is demonstrated in Section 3.2 of this SAR. The effects of tornado wind and missiles, flood water and earthquakes are described in Sections 3.2.1, 3.2.2 and 3.2.3, respectively. It is shown in those sections that the cask will not tip over under the most severe natural phenomena specified in the Prairie Island Updated Safety Analysis Report. Also the cask will not slide on its pad any more than about one inch under any of these loadings.

The cask will be lifted at Prairie Island using a single failure proof crane. The trunnions are designed to the requirements of ANSI N14.6 (Reference 2) for lifting devices for critical loads with increased stress design factors. Safety factors of 6 against the trunnion yield strength and 10 against the ultimate strength are provided. In addition, the cask will be handled by the transport vehicle. The cask will always be in a vertical orientation and never lifted higher than 18 in. Therefore it is extremely unlikely that the cask could be dropped.

However, this section of the SAR considers design events of the third and fourth types (includes accidents) as defined in ANSI/ANS-57.9. The third type of events are those that could reasonably be expected to occur over the lifetime of the ISFSI (does not include dropping of the cask). The fourth type of events include severe natural phenomena (described in Section 8.2.1 through 8.2.5) and man induced low probability events postulated because their consequences could result in the maximum potential impact on the immediate environs. Therefore the cask is examined for a dropping accident, which is a hypothetical impact event that is extremely unlikely to occur.

8.2.8.2 ACCIDENT ANALYSIS

In this section the cask is evaluated under bottom end impact on the ISFSI storage pad after a drop from a height of 18 in. The storage pad is the hardest concrete surface outside of the containment building. The cask is always oriented vertically and is never lifted higher than 18 in. once it leaves the containment building. Therefore this case is an upper bound drop event since impact onto a softer surface would result in lower cask deceleration and a lower impact force.

8.2.8.2.1 DYNAMIC IMPACT LOADS

The impact analysis is based on the methodology of EPRI NP-4830 (Reference 3). This report considers the mass and geometry of the cask but assumes it to be rigid compared to the concrete storage pad. The storage pad properties and the cask geometry are used to determine the pad hardness parameter. The report provides graphs that show the force on the cask as a function of storage pad hardness.

TARGET HARDNESS

The target (or storage pad) hardness parameter, S, for the end drop case is:

$$S_{end} = \frac{2}{w^3} \frac{rAkM_u\sigma_u}{\left(1 - e^{-\beta r} \cos \beta r\right)} = 95{,}680 \quad \text{(formula 4, reference 3)}$$

Where:

$r = cask\ bottom\ radius = 45.5\ in.$

$A = bottom\ area = \pi r^2 = 6{,}504\ in.^2$

$k = \frac{\pi E}{1 - v_s^2} s = 118{,}200\ psi/in.$

$E_s = Soil\ modulus = 30{,}000\ psi$

$v_s = Poisson's\ ratio\ of\ soil = 0.45$

M_u = *is based on pad thickness of 36 in., #14 rebar @ 12 in. spacing (nominal),* S_y *rebar of 60,000 psi, 2 in. cover (nominal),* σ_u *of 4,000 psi max. concrete compressive strength*

$= 4.25 \times 10^6$ *in.lb/ft.*

$\beta = 0.02686$

$W = 240{,}000$ *lb. (slightly low weight gives conservatively high hardness)*

DECELERATIONS

Figure 22 for a 20 in. drop height from EPRI NP-4830 (Reference 3) can be conservatively used to determine the cask deceleration after the 18 in. end drop. The upper bound deceleration is 40 g's for a hardness parameter, S, of 95,680. The maximum impact force is then 40 times the weight of the cask. The TN-40 end impact stress analyses below are conservatively performed for a deceleration of 50 g's.

8.2.8.2.2 CASK BODY ANALYSIS

The analysis results for the hypothetical cask drop accident are reported in this section as the 18 in. bottom end drop onto the storage pad. As explained in Section 8.2.8.1, this accident has a very low probability, but in view of its potential impact on the environs, a detailed analysis was performed.

A conservative 50 g bottom drop onto the concrete pad was analyzed. The ANSYS model in Section 4A.3.2 was used to evaluate the stresses in the cask body due to the 50 g bottom drop with the following modifications:

1. Nodes 44-200 and 59-201 (See Figure 8.2-1C) are also coupled. They are coupled in the axial direction but are allowed to slide over each other in the radial and hoop directions.

2. All nodes on the outside bottom surface of the cask are fixed in the axial direction.

3. The internal loading effects are simulated by distributed pressure acting on the inside bottom surface of the cavity, as opposed to nodal forces.

4. A distributed inertial force of 50 g's was applied on each finite element in the model (See Figure 8.2-1D).

Stress results for this individual load case are reported in Figure 8.2-1A. Figure 4A.3-12 shows the locations on the cask body where stress results are reported. This case is combined with the stress results for bolt preload and internal pressure of 100 psi in Figure 8.2-1B. This case will be evaluated against the cask body criteria for a Level D event below.

8.2.8.2.3 LID BOLT ANALYSIS

The lid bolts are analyzed in this section under the loadings selected to bound those for the hypothetical bottom end drop onto the concrete storage pad.

BOTTOM END DROP

The bottom end drop from a height of 18 in. onto the concrete storage pad is analyzed above in Section 8.2.8.2.1. That section indicates that the cask deceleration may reach 40 g. This analysis conservatively examines the effects (if any) of a 50 g quasistatic loading on the lid bolts.

During a bottom end drop, the rim of the lid is forced against the flange of the cask body. The lid is initially seated against the flange by preloading (torquing) the bolts. The bolt preload will not be affected if compressive yielding of the contact bearing area does not occur.

The contact force on the bearing area, conservatively neglecting internal pressure, is the bolt preload force less the seal compression force plus the 50 g inertial force of the lid system. The preload force, from Section 4A.4.1, is 1,790,400 lb.[1] The seal seating force is 1,037,164 lb. The weight of the lid system (including shield plate and resin disk) is 15,397 lb.

Therefore, during a 50 g deceleration in the axial direction the contact force between lid and cask body is:

$$\begin{aligned} F_{contact} = F_{Bolt\ Preload} - F_{seal\ seating} + 50\ (W_{lid\ system}) &\\ = 1{,}790{,}400 - 1{,}037{,}164 + 50(15{,}397) &\\ = 1{,}523{,}086\ lb. & \end{aligned}$$

Figure 8.2-7A illustrates the bearing interface between lid edge and body flange. The bearing area equals the area within the diameter of the lid raised section (77.25 in.) less that outside of the body chamber (73.29 in.) less the area of the seal groove.

$$A_{bearing} = \frac{\pi}{4}\left(77.25^2 - 76.2^2 + 74.00^2 - 73.29^2\right)$$

$$= 208.7\ in.^2$$

The bearing stress during impact is then equal to:

$$S_{bearing} = \frac{F_{contact}}{A_{bearing}} = \frac{1{,}523{,}086\ lb.}{208.7\ in.^2} = 7{,}229\ psi$$

---

[1]Based on 25,000 psi bolt preload stress (conservative)

This contact stress is well below the 37,500 psi yield stress of the lid and flange material. The bolt preload will not be affected by the bottom drop. Therefore, this hypothetical accident case will not affect the bolts.

8.2.8.2.4 BASKET ANALYSIS

BOTTOM END DROP

The basket analysis is presented in detail in Appendix 4B. The analysis of the basket under vertical loading is found in Section 4B.7. The fuel assemblies react directly against the bottom of the cask in the vertical load case. They do not load the basket as in the side impact case. The fuel assemblies themselves can withstand more than 80 g as indicated in Reference 6.

The Appendix 4B analysis performed for the hypothetical 50g bottom end drop onto the concrete pad does not take credit for the aluminum strength. It is conservatively assumed that all of the load is taken by the 304 stainless steel. Therefore:

$$\sigma = \frac{Total\ Compressive\ Load}{Cross\ Section\ of\ 304SS}$$

On a single wall panel the stress calculated in Section 4B.7 for a 3g load would simply be 50/3 X 304 = 5067 psi. This 5067 psi compressive stress is acceptable since the Section 4B.5 Level D membrane stress intensity limit for 304 stainless steel is 44,900 psi at 400°F (the approximate maximum temperature at the end of the basket). In addition, axial compression tests of unsupported fuel compartment box sections described in Section 4C.2 show that the compressive stress can reach 23,000 psi before failure (even without the stiffening effects of the aluminum plates). Therefore buckling will not occur.

SHEAR STRESS IN ½ IN. FUSION PLUG WELDS

This section describes the analysis of the shear stresses at the basket plug welds due to the combination of differential thermal expansion between aluminum and stainless steel and the bottom end drop.

The thermal analysis of the basket is described in Section 3.3.2.2. The analysis determines the basket temperatures for the condition with maximum solar heating, maximum decay heat from the cask contents, and 100°F ambient air temperature. The basket temperatures were used directly in the ANSYS structural models to calculate the basket panel stresses due to differential thermal expansion. Stresses occur due to differences between the coefficients of thermal expansion of the 304 stainless, aluminum and boral (See Section 4B for detailed ANSYS model descriptions). In order to calculate the maximum shear stresses at the 1/2 in. plug welds, it was conservatively assumed that the 1.38 in. diameter stainless plugs that penetrate the 1.5 in. diameter hole in the aluminum (and boral) plates are not centered. The plugs were assumed to be in contact initially (at 70°F) with the opposing sides of the two holes in the aluminum (the sides toward the center of the panel) so that the maximum interference of aluminum and steel will occur when the panel is heated. In this worst plug misalignment case with the highest temperature of 530°F, seen by any portion of the basket, the weld shear stress could reach a maximum of 25,434 psi as shown on Figure 4B.6-6.

A full length compartment wall (160 in. long) with a span length of 8.05 in. is evaluated for shear stresses at the 1/2 in. plug welds due to a 50 g end drop.

Size of weld = 0.5 dia.

$$Number\ of\ welds = 2 \times 2 \times \frac{160}{8} = 80$$ *(each 8" spacing has two rows of plugs, each plug has two welds, one on each side)*

$$Shear\ area = \pi/4\ (0.5)^2 \times 80 = 15.71\ in^2$$

$$Weight\ of\ aluminum = 2 \times 8.05 \times 0.25 \times 160 \times 0.105 = 67.62\ lbs.$$

$$Weight\ of\ boral = 1 \times 7.50 \times 0.075 \times 160 \times 0.0903 = 8.13\ lbs.$$

Total weight of aluminum and boral = 75.75 lbs.

Assuming the 50 g compressive load is uniformly distributed to all of the 80 welds, the shear stress is

$$\tau = \frac{75.75 \times 50}{15.71} = 241\ psi$$

Appendix F to Section III of the ASME Code (Reference 4) provides a basic 0.42 S_u limit on the average primary shear stress across a section loaded in pure shear for Level D conditions. The combined shear stress in the weld due to differential thermal expansion and the end drop accident is 25,434 + 241 = 25,675 psi which is below the limit of 26,670 psi (0.42 S_u) based on the temperature of 530°F at the worst panel location.

Based on the results of this analysis, it is concluded that the basket is structurally adequate for withstanding the combined loads due to thermal expansion and the 18 inch end drop accident, and will properly support and position the fuel assemblies.

BASKET WELD MATERIAL TOLERANCE

The 18 inch cask end drop is the most limiting postulated event in terms of structural analysis. Achieving a tight tolerance between the aluminum plates and the stainless steel boxes at the bottom edge of the basket was considered. If the aluminum plates are recessed, the stainless steel boxes must support the load due to the end drop. In order to evaluate this possibility, an analysis was performed.

The basket was analyzed first assuming that the entire load is supported by the stainless steel, and then analyzed assuming that the entire load is supported by the aluminum.

The basket is composed of stainless steel boxes, aluminum plates and boral plates held together by plug welds every 8 inches vertically. A cross section of the basket wall is shown in Figure 4B.7-1.

The stainless steel is analyzed as a box section with a thickness of 0.10 inches and inside dimensions of 8.05 inches square. The highest loading occurs at the base of the basket, which is analyzed at a temperature of 300°F. The basket segments below the bottom welds will be stressed the most, since they are taking the entire weight of the basket. The yield stress of Type 304 stainless steel at 300°F is 22,500 psi.

The area of the box section is:

$$A = 8.25^2 - 8.05^2 = 3.26 \text{ in.}^2$$

The radius of gyration is:

$$r = 0.408d = 0.408(8.15) = 3.33 \text{ in.}$$

Assuming hinged ends, the slenderness ratio is:

$$\frac{kl}{r} = \frac{(1)(8.0)}{(3.33)} = 2.4$$

From Section NF-3322.1, equation 6a and Table NF-3523(b)1 of Reference 4, the critical stress is:

$$F_a = 1.5S_y\left[0.47 - (kl/r)/444\right]$$

$$= 1.5(22,500)\left(0.47 - \frac{2.4}{444}\right)$$

$$= 15,680 \text{ psi } \textit{for level C service conditions.}$$

Level C service conditions are more restrictive than level D accident conditions but are used because Appendix F for level D conditions gives no guidance for austenitic stainless steel.

The critical load applied to the box section is:

$$Pc = F_a \times A = 15{,}680 \times 3.26 = 51{,}116 \text{ lbs.}$$

The weight of the basket per box section is approximately 320 lbs. Therefore the maximum g load sustainable by the stainless steel is 51,116/320 = 160 g's.

This result is conservative as shown by the tests on the bare boxes presented in Section 4C. The test showed that a stress of 23,000 psi was sustainable at room temperature. This is roughly equivalent to 17,000 psi at 300°F, which results in a sustainable g load of 176 g's.

The condition where the aluminum panels are supporting the load was evaluated by assuming a stable cross section with each member bending about a common neutral axis as shown in Figure 4B.5-6. The allowable stresses were taken from Reference 7.

$$A = 0.5 \text{ in}^2/\text{in}$$

$$r = 0.1443 \text{ in.}$$

Assuming hinged ends,

$$l = 8 \text{ in.}$$

$$k = 1.0$$

$$\frac{kl}{r} = 55.44$$

If T = 300°F, the allowable axial compression stress is:

$F_c = 15.8 - 0.096\ kl/r = 10.48$ *ksi for cross sections farther than 1.0 inch from any weld, and*

$F_c = 9.5$ *ksi for cross sections within 1.0 inch of a weld.*

The critical load is $9{,}500 \text{ psi} \times 0.5 = 4{,}750 \text{ lb/in.}$

The critical load per panel is $4{,}750 \times 8.05 = 38{,}237 \text{ lbs.}$

This is equivalent to a g load of $\frac{38{,}237}{153} = 250\ g's$

Based on this analysis, the stainless steel members can withstand a 160 g impact load of the entire basket during the end drop. The aluminum members can withstand a 250 g impact load due to the entire weight of the basket.

Based on the analysis in Section 8.2.8.2.2 showing that the maximum g load during the end drop is less than 50 g's, the stainless steel and the aluminum panels can each withstand the end drop loading separately. Therefore the tolerance on the lengths of the steel boxes and aluminum panels is not critical.

8.2.8.3 ACCIDENT DOSE CALCULATIONS

Cask drop will not breach the cask confinement barrier. No radioactivity will be released and no resultant doses will occur. Table 7A-4 shows the calculated dose rate assuming the neutron shield and outer shell are removed.

8.2.9 LOSS OF CONFINEMENT BARRIER

8.2.9.1 CAUSE OF ACCIDENT

The following postulated accident scenario is not considered to be credible. It is hypothesized solely to demonstrate the inherent safety of the ISFSI by subjecting it to a set of simultaneous multiple failures, any one of which is far beyond the capability of natural phenomena or man-made hazards to produce. A simultaneous failure of all protective layers of confinement is postulated to occur by unspecified nonmechanistic means in the cask.

8.2.9.2 ACCIDENT ANALYSIS

In this accident, the confinement function is nonmechanistically removed. Heat removal and radiation shielding functions operate in the normal passive manner.

This is equivalent to breaking the cask seal barriers (no release), removing the closure lids (no release), failing all the cladding in all the loaded fuel assemblies (gap activity release), and finally, failing the fuel pellets themselves such that the remaining Kr-85 is released from the fuel matrix.

8.2.9.3 ACCIDENT DOSE CALCULATIONS

Table 7.2-7 lists the nuclides present in a cask containing 40 design basis fuel assemblies. The only nuclide listed in Table 7.2-7 which naturally occurs in the gaseous state, which could escape from the cask following a postulated breach of cask confinement barrier and which would be a significant dose contributor, is Kr-85.

All of the Kr-85 gas is conservatively assumed to be instantaneously released from the TN-40 cask. There is no additional decay of Kr-85 in transit from the spent fuel storage cask to the receptor and no credit is taken for personnel protection due to any structure or system.

The maximum individual is assumed to be located at the site boundary where the least amount of atmospheric dispersion takes place (largest χ/Q value). The dose results for this location are conservative for any individual (maximum) and may be reported as dose to an individual at the nearest site boundary.

The site boundary χ/Q values were calculated as described in Section 2.3.4 and shown on Table 2.3-1.

As shown on Table 7.2-7, there are 9.67E4 Curies of Kr-85 activity in the TN-40 spent fuel cask following 10 years of fuel decay. The gamma (whole body) dose conversion factor (K) for Kr-85 is 1.9 rem/hr per Ci/cu. meter. The atmospheric dispersion factors for ground level release were calculated as described in Section 2.3.4.

Dose evaluations were performed based on Regulatory Guide 1.3 (Reference 5) methodology and equations.

The offsite radiological consequences of a postulated loss of spent fuel cask confinement barrier for a cask located at the ISFSI are provided in Table 8.2-1 and plotted in Figure 8.2-33.

The nearest site boundary or maximum individual whole body dose for the loss of spent fuel cask confinement barrier is determined to be 0.15 rem. This dose is well within the 5 rem criteria given in 10CFR72.106(b)

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9.2 STARTUP TESTING AND OPERATION

9.2.1 ADMINISTRATIVE PROCEDURES FOR CONDUCTING TEST PROGRAM

The administrative procedures for the ISFSI will be the same as those used for the Prairie Island Nuclear Generating Plant. Any changes to, or deviations from, these procedures and instructions will be reviewed and approved by the Station Operations Committee as appropriate.

9.2.2 TEST PROGRAM DESCRIPTION

9.2.2.1 PHYSICAL FACILITIES

Before startup and during the lifetime of the ISFSI, the cask monitoring instrumentation, the electrical system, the communications system, and the storage casks will be tested to ensure their proper functioning.

The cask monitoring instrumentation alarms will be tested to ensure that individual alarm signals annunciate at the local annunciator enclosure at the ISFSI location.

The electrical system will be tested to ensure that power is available for the cask monitoring instrumentation and the local annunciator. The lighting and service receptacles are also tested for proper operation.

The communications system will be tested to ensure that the telephone at the local annunciator is properly connected into the plant phone system.

The storage casks will be tested with a dummy fuel assembly prior to fuel loading to ensure that the assemblies will fit properly.

The seals of the casks will be inspected prior to and tested following fuel loading.

9.2.2.2 OPERATIONS

Testing and calibration of instruments and components in use at the ISFSI will be done in accordance with procedures established for similar equipment in use at the Prairie Island Nuclear Generating Plant. Acceptance criteria and corrective actions for test margins and response times will be specified by the equipment vendors.

9.2.3 TEST DISCUSSION

The preoperational test purposes, responses, acceptance criteria, margins, and corrective actions are discussed in Section 9.2.2. Instrumentation, electrical, and communications equipment will be functionally tested to confirm operability.

9.2.4 COMPLETION OF PRE-OPERATIONAL TEST PROGRAM

Preoperational testing of the Prairie Island ISFSI was completed on April 15, 1995. Pursuant to the requirements of 10CFR72.82(e), a report of the preoperational test acceptance criteria and test results was submitted to the NRC by Reference 3.