SECTION 13

PLANT OPERATIONS

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SECTION 13 PLANT OPERATIONS

13.1 SUMMARY DESCRIPTION

Plant personnel are qualified and experienced to perform plant operations and plant maintenance that are necessary for safe operation of the plant.

Training programs are scheduled to maintain sufficient licensed operators and a competent supporting technical staff. Plant activities are conducted in accordance with Quality Assurance, Emergency, and Security Plans and written procedures implemented in response to regulatory requirements. Inspection and testing are conducted in accordance with a program which meets regulatory requirements.

13.2 ORGANIZATION, RESPONSIBILITIES, AND QUALIFICATIONS

13.2.1 Organization

The Prairie Island Nuclear Generating Plant site management and organization, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in the Technical Specifications are described in NSPM-1, “Quality Assurance Topical Report”.

By license amendment, dated September 22, 2008, the NRC made NSPM the licensee authorized to use and operate Prairie Island Nuclear Generating Plant Units 1 and 2.

In support of the individual responsibilities of plant personnel an onsite Plant Operating Review Committee provides multi-discipline review of various plant activities.

The onsite organization includes the technically trained personnel necessary to support all aspects of plant operations.

13.2.2 Duties and Responsibilities of the Operating Staff Personnel

The responsibilities and duties of key headquarter and site personnel are described in NSPM-1 as is allowed by Technical Specification 5.2.1.

13.2.3 Qualification of Plant Personnel

The minimum qualifications of plant technical and operating personnel are specified in Technical Specifications Section 5.3, Plant Staff Qualifications.

13.3 PERSONNEL EXPERIENCE AND TRAINING

13.3.1 General

The minimum qualifications of plant technical and operating personnel are specified in Technical Specifications Section 5.3, Plant Staff Qualifications.

13.3.2 Training and Retraining Program

NSPM is a member of the National Academy for Nuclear Training. Training and retraining programs, based on a systems approach to training, have been accredited by the Institute of Nuclear Power Operations (INPO) (Reference 5).

By letter dated April 8, 1988, NSP committed to conduct STA training in accordance with the National Academy for Nuclear Training accredited STA training program. Item I.A.2.1, titled “Immediate Upgrading of Reactor Operator and Senior Reactor Operator Training and Qualifications”, and item II.B.4, titled “Training for Mitigating Core Damage”, were resolved with the NRC in correspondence dated 12/30/80, 8/21/81, 4/16/82 and 12/14/82.

The Plant Manager is responsible for the overall conduct and administration of the plant training program. Implementation responsibilities are delegated as follows. The Training Process Manager is responsible for ensuring that training processes follow the systematic approach to training as described in the INPO accreditation criteria. General Superintendents of the major disciplines are responsible for the development and implementation of the training programs. Included in these programs is a simulator certification program, which meets 10CFR55 and Regulatory Guide 1.149; and operation, maintenance and modification of a site specific simulator.

13.3.3 Promotion of Personnel

NSPM management is responsible for training and maintaining a qualified staff of technical and operations personnel. Every effort is made to promote plant personnel from within the plant organization consistent with the training and experience requirements of the assigned position.

13.3.4 Personnel Behavior

The “Fitness for Duty Program” applies to all nuclear generation personnel, including all badged contract workers and craft union personnel hired by NSPM or its contractors or its agents.

It recognizes that fatigue, stress, illness and temporary physical impairments, as well as drug and alcohol abuse, can have a negative effect on a worker’s fitness and jeopardize safe operations.

All personnel badged for unescorted access to the plant are subject to random drug and alcohol testing and are trained to be observant of co-worker or visitor behavior that may indicate a fitness for duty concern. Supervisors are trained to be observant of employee behavior that might indicate excessive fatigue or unhealthy behavior patterns and to bar employees from working if they appear unfit for duty.

As of January 3, 1991 Northern States Power Company certified implementation of the fitness for duty program in accordance with the requirements of 10 CFR Part 26 (Reference 7).

13.4 OPERATIONAL PROCEDURES

13.4.1 General

A preoperational test program was conducted to assure that all systems and equipment function properly. The initial preoperational and startup test programs are described in Appendix J. Westinghouse and Pioneer Service and Engineering (PS&E) provided written procedures and technical direction for these programs. The plant operating staff participated in the preparation and execution of these tests.

Detailed written procedures, including the applicable check-off sheets and instructions have been prepared in accordance with the Technical Specifications and ANSI N18.7-1976. Subsequent to implementation of NSPM-1, Quality Assurance Topical Report, such procedures are prepared in accordance with the Technical Specification and NSPM-1. Plant operations are conducted in accordance with these procedures.

13.4.2 Procedure Development

The original operations procedures were written by members of the plant staff with the technical assistance of Westinghouse and were reviewed by the Operations Committee.

Procedures are periodically updated to reflect plant modifications and improvements in methods of operation as operating experience accumulates.

Detailed written plant operating procedures and special written one-of-a-kind plant operating procedures covering areas listed in accordance with Technical Specifications Plant Operating Procedures are periodically necessary. These are prepared by qualified personnel and are reviewed by the Plant Operating Review Committee.

Maintenance and test procedures, checklists, and other necessary records to satisfy routine inspections, preventive maintenance programs, and license requirements, have been and will continue to be developed by qualified personnel.

13.4.3 Emergency Plan

The Emergency Plan for the plant consists of a document referred to as the Prairie Island Generating Plant Emergency Plan. This Plan was submitted according to 10CFR50 emergency planning regulations. Subsequent revisions to the plan are issued and reported to the NRC in accordance with 10CFR Part 50.54(q).

The NRC has concluded that onsite and offsite emergency preparedness is adequate and that the emergency plans have been upgraded in accordance with NUREG-0737 Item III.A.2.1 (Reference 2).

The Emergency Plan identifies the location of primary and backup Emergency Operations Facilities (EOF). The location of the EOFs was found acceptable by the NRC in a letter dated October 27, 1983.

The Emergency Plan is dependent upon the Emergency Plan Implementing Procedures for implementation. Revisions to procedures are issued and reported to the NRC in accordance with 10CFR50 Appendix E, Section V.

13.4.4 Security Plans

The security plans for the plant consists of documents referred to as the Prairie Island Nuclear Generating Plant Security Plan and Prairie Island Nuclear Generating Plant Cyber Security Plan as approved by the NRC.

The security plans are periodically revised to meet changing requirements and the current revision is maintained on file on site. Revisions to the security plans, not requiring prior NRC approval, are issued and reported to the NRC in accordance with 10CFR50.54(p).

13.4.5 Quality Assurance Plan

NSPM nuclear plant operational and support activities are conducted under the NSPM Quality Assurance Topical Report (QATR). QATR NSPM-1 superseded QATR NMC-1 on 3/31/09. NSPM-1 is the top-level policy document that establishes the manner in which quality is to be achieved and presents NSPM's overall philosophy regarding achievement and assurance of quality. NSPM-1 responds to and satisfied the requirements of Appendix B of 10 CFR Part 50. NSPM-1 is periodically revised to meet changing requirements. Revisions are submitted for NRC review according to 10 CFR 50.54(a).

NMC-1, Revision 0 was submitted for NRC review on October 31, 2003 and received NRC approval via Safety Evaluation Reports dated January 13, 2005 and March 24, 2005.

13.4.6 Inservice Inspection and Testing Program

The Prairie Island Inservice Inspection and Testing Program (for system pressure retaining components, pumps and valves) are established in accordance with the ASME OM Code in compliance with 10CFR50.55a. Where it is not practical or possible to meet the requirements of the Code, relief requests are submitted for NRC review.

13.5 OPERATIONAL RECORDS AND REPORTING REQUIREMENTS

13.5.1 Records of Initial Tests

All preoperational procedures, test data, and reports are kept on file at the plant site.

Complete records of the plant startup tests (submitted to the NRC for Units1 and 2 on 10/31/74 and 5/15/75 respectively) are kept at the plant site in the test file. These records include:

a. Startup test procedures. This is the final, as run, test procedure, including approvals and data sheets.

b. Pertinent recorder charts and log sheets.

c. Test reports - This includes any reports prepared by NSP, Westinghouse or PS&E.

13.5.2 Routine Operation

Operating, maintenance and testing records and logs are kept on file in accordance with the Federal Regulations and NSPM policy.

13.5.3 Abnormal Operation

In the event of any unusual, unexplained, or potentially unsafe occurrence, appropriate members of the plant staff are assigned to conduct an investigation and prepare a report. Instructions for conducting investigation and the report format are outlined in plant directives. A complete file of investigation reports is maintained.

13.5.4 Reporting Requirements

Reports are submitted to the Commission to satisfy the requirements of Title 10, Code of Federal Regulation, and the Prairie Island Technical Specifications.

13.6 OPERATIONAL REVIEW AND AUDITS

Review and audit of facility operations are performed according to the NSPM Quality Assurance Topical Report, NSPM-1. NSPM-1 is the top-level policy document that establishes the manner in which quality is to be achieved and presents NSPM's overall philosophy regarding achievement and assurance of quality.

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13.7 EMERGENCY PROCEDURES

After the Three Mile Island accident, Westinghouse and the Owners Group developed Emergency Response Guidelines to improve plant specific Emergency Procedures. These guidelines were submitted to, reviewed by and approved by the NRC Staff (References 3 and 4). Completion of the NRC Staff review of the Procedure Generation Package (PGP) for the Prairie Island Emergency Operating Procedures (EOP) was documented in a NRC Safety Evaluation transmitted by Reference 6.

13.8 MAINTENANCE PROCEDURES

Maintenance work requests and their associated procedures shall be reviewed as required by the Quality Assurance Topical Report, NSPM-1.

13.9 TECHNICAL REQUIREMENTS MANUAL

The Technical Requirements Manual (TRM) is a licensee-controlled document that provides a location for items removed from the Technical Specifications that do not meet the criteria of 10CFR50.36. (Reference 11)

For purposes of making changes to the TRM: (1) changes to the TRM will be controlled by the provisions of 10CFR50.59, (2) summaries of 50.59 evaluations will be submitted to the NRC, and (3) the TRM is a general reference in the USAR and, as such, changed pages will not be submitted to the NRC.

13.10 RISK INFORMED CATEGORIZATION AND TREATMENT

13.10.1 Introduction

On November 22, 2004, the NRC issued 10 CFR 50.69 that presented nuclear management with an opportunity to further enhance equipment reliability and plant safety by focusing on those critical structures, systems, and components (SSCs) with the highest safety significance. The rule broadly adjusts the scope of safety related components that are subject to the existing NRC regulations. The implementation of this new rule is strictly voluntary on the part of each licensee.

For those safety related components that are categorized as Low Safety Significant, 10 CFR 50.69(b)(1) allows compliance with alternative requirements in lieu of the following special treatment requirements.

(i) 10 CFR Part 21

(ii) The portion of 10 CFR 50.46a(b) that imposes requirements to conform to Appendix B to 10 CFR Part 50.

(iii) 10 CFR 50.49.

(iv) 10 CFR 50.55(e).

(v) The in-service testing requirements in 10 CFR 50.55a(f): the in-service inspection and repair, and replacement (with the exception of fracture toughness), requirements for ASME Class 2 and Class 3 SSCs in 10 CFR 50.55a(g); and the electrical component quality and qualification requirements in Section 4.3 and 4.4 of IEEE 279, and Sections 5.3 and 5.4 of IEEE 603-1991, as incorporated by reference in 10 CFR 50.55a(h).

(vi) 10 CFR 50.65, except for paragraph (a)(4).

(vii) 10 CFR 50.72.

(viii) 10 CFR 50.73.

(ix) Appendix B to 10 CFR Part 50.

(x) The Type B and Type C leakage testing requirements in both Options A and B of Appendix J to 10 CFR Part 50, for penetrations and valves meeting the following criteria:

(A) Containment penetrations that are either 1-in. nominal size or less, or continuously pressurized.

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(B) Containment isolation valves that meet one or more of the following criteria:

(1) The valve is required to be open under accident conditions to prevent or mitigate core damage events;

(2) The valve is normally closed and in a physically closed, water-filled system;

(3) The valve is in a physically closed system whose piping pressure rating exceeds the containment design pressure rating and is not connected to the reactor coolant pressure boundary; or

(4) The valve is 1-in. nominal size or less.

(xi) Appendix A to Part 100, Sections Vl(a)(1) and Vl(a)(2), to the extent that these regulations require qualification testing and specific engineering methods to demonstrate that SSCs are designed to withstand the Safe Shutdown Earthquake and Operating Basis Earthquake.

It should be noted that 10 CFR 50.69 does not replace the existing “safety related” and “non-safety related” classification. Instead, 10 CFR 50.69 divides these classifications into two subcategories based on high or low safety significance, such that there are four categories of risk-informed safety class (RISC), as shown below:

- RISC-1: safety related SSCs that perform (high) safety significant functions.
- RISC-2: non-safety related SSCs that perform (high) safety significant functions.
- RISC-3: safety related SSCs that perform low safety-significant functions.
- RISC-4: non-safety related SSCs that perform low safety-significant functions.

When applying alternative treatment, 10 CFR 59.69(d) requires that the licensee “shall ensure, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety related functions under design basis conditions, including seismic conditions and environmental conditions and effects throughout their service life.” Periodic inspection and testing activities will be conducted to ensure RISC-3 SSCs remain capable of performing their safety related functions. The corrective action program will be used to document and correct in a timely manner any conditions that would prevent a RISC-3 SSC from performing its safety related functions.

Prairie Island received approval from the NRC to implement 10 CFR 50.69 as outlined in Reference 1.

13.10.2 SSC Categorization

As outlined in the Safety Evaluation Report (SER) for License Amendment 230/218 to Renewed Facility Operating License No. DPR-42 and DPR-60, Prairie Island will use the methodology outlined in NEI 00-04, Reference 8.

10 CFR 50.69(f)(2) requires updating the UFSAR to reflect which systems have been categorized. The following table is revised as part of the periodic UFSAR update to reflect systems that have been categorized.

System	Description
CL	Cooling water
CS	Containment Spray
MS	Main Steam
RC	Reactor Coolant
RH	Residual Heat Removal
SI	Safety Injection
VC	Chemical & Volume Control
ZC	Containment Vent
ZS	Shield Building Vent

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13.10.3 SSC Treatment

13.10.3.1 Treatment of Component Categories

The programs or processes that implement the special treatment requirements are revised to recognize that the special treatments no longer apply to RISC-3 SSCs. The programs or processes either allow continued application of the special treatments or acceptable alternative treatments, as applicable, to provide reasonable confidence that these SSCs would perform their safety related function under design basis conditions.

The following information provides the general approach for applying treatment for the component categories:

A. <u>RISC-1 Components</u>
RISC-1 SSC should continue to satisfy all of the existing regulatory requirements that are applicable, including those insights that were considered during the SSC's categorization.

B. RISC-2 Components
The purpose of treatment applied to RISC-2 SSCs is to maintain their ability to perform risk-significant functions consistent with the categorization process. These components will continue to receive any existing special treatment required by NRC regulations. Additionally, the risk significant functions of these components will receive consideration for enhanced treatment. This consideration is described in paragraph 13.10.3.2.

C. RISC-3 Components
These components may receive alternative treatments, in lieu of special treatments, as described in 13.10.1

D. RISC-4 Components
The treatment of these components is not subject to regulatory control.

E. Uncategorized Components
Until a component is categorized, it continues to receive the special treatment required by NRC regulations and associated Prairie Island implementing programs, as applicable.

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13.10.3.2 Enhanced Treatment of RISC-2 SSCs

The 10 CFR 50.69 procedures and 10 CFR 50.69(d)(1) require that RISC-2 SSCs perform their functions consistent with the categorization process assumptions by evaluating treatment being applied to these SSCs to ensure that it supports the key assumptions in the categorization process that relate to their assumed performance.

Non-safety related HSS components may perform risk-significant functions that are not addressed by the special treatment requirements in NRC regulations or current Prairie Island programs.

When a non-safety related component is categorized as HSS, determine whether enhanced treatment is warranted to enhance the reliability and availability of the component in support of its HSS function(s). In particular, evaluate the treatment applied to the component to ensure that the existing controls are sufficient to maintain the reliability and availability of the component in a manner that is consistent with its categorization. This process evaluates the reliability of the component, the adequacy of the existing controls, and the need for any changes. If changes are needed, additional controls are applied to the component. In addition, the component is placed under the Maintenance Rule monitoring program, if not already scoped in the program. Components under these controls will remain non-safety related, but the enhanced treatments will be appropriately applied to give additional confidence that the component will be able to perform its HSS function(s) when demanded.

These identified processes provide reasonable confidence that HSS components will be able to perform their risk significant functions.

13.11 REFERENCES

1. Safety Evaluation Report by the Office of Nuclear Reactor Regulation related to Amendment No. 230/218 to Renewed Facility Operating License No. DPR-42 and DPR-60, Prairie Island. 60400000631

2. Letter, R A Clark (NRC) to D M Musolf (NSP), “NUREG-0737 Item III.A.2.1 Emergency Plan Upgrade to Meet Rule”, May 13, 1983. (18349/1377)

3. Letter, D G Eisenhut (NRC) to D M Musolf (NSP), “Safety Evaluation of Emergency Response Guidelines (Generic Letter 83-22)”, June 3, 1983. (18349/1626)

4. Letter, T M Novak (NRC) to D B Butterfield (WOG), “Supplemental Safety Evaluation Report by the Office of Nuclear Reactor Regulation in the Matter of Westinghouse Owners Group Emergency Response Guidelines”, December 26, 1985.

5. Letter, D M Musolf (NSP) to the Director of NRR (NRC), “NRC Licensed Operator Training Program and Licensed Operator Requalification Program”, March 21, 1988. (30405/0752)

6. Letter, D C DiIanni (NRC) to T M Parker (NSP), “Safety Evaluation for the Prairie Island Nuclear Generating Plant Units 1 and 2 Procedure Generating Package”, February 14, 1990. (30814/0234)

7. Letter, C E Larson (NSP) to the Director of NRR (NRC), “Certification of Compliance to 10 CFR 26 Fitness for Duty Program”, January 3, 1990. (30814/0007)

8. NEI 00-04, 10 CFR 59.69 SSC Categorization Guidance, Rev. 0. 60400000631

9. Letter, C F Lyon (NRC) to M D Wadley (NSP), “Order Approving the Transfer of Operating Authority Under Facility Operating Licenses for Prairie Island Nuclear Generating Plant...”, May 15, 2000. (3726/2758)

10. Letter, T J Kim (NRC) to M B Sellman (NMC), “Prairie Island Nuclear Generating Plant, Units 1 & 2, and Prairie Island Independent Spent Fuel Storage Installation - Issuance of Conforming Amendments re: Transfer of Operating Authority Under the Facility Operating Licenses and Materials License from Northern States Power Company to Nuclear Management Company, LLC (TAC Nos. MA7275 and MA7276), August 7, 2000. (3752/2124)

11. Letter, T Kim (NRC) to M Nazar (NMC), "Prairie Island Nuclear Generating Plant, Units 1 and 2 -Issuance of Amendments Re: Conversion to Improved Technical Specifications (TAC Nos. MB0695 and MB0696)", July 26, 2002.

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