e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from site to unrestricted areas of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the Process Control Program (PCP) made during the reporting period.

MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORV's or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, ~~and Program Analysis~~, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted as set forth in 6.5 above.

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.11 The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) shall be provided to the Regional Administrator of the Regional Office of Inspection and Enforcement, with a copy to the Director, Nuclear Reactor Regulation, Attention Chief of the Core Performance Branch, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555 for all core planes containing bank "D" control rods and all unrodded core planes at least 60 days prior to cycle initial criticality. In the event that the limit would be submitted at some other time during core life, it shall be submitted 60 days prior to the date the limit would become effective unless otherwise exempted by the Commission.

Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Office of Inspection and Enforcement Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

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PDR ADOCK 05000395
P PDR

RECORDS

6.5.2.10 Records of NSRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NSRC meeting shall be prepared, approved and forwarded to the Vice President, Nuclear Operations within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Vice President, Nuclear Operations within 14 days following completion of the review.

c. Audit summary reports encompassed by Section 6.5.2.8 above, shall be forwarded to the NSRC and to the Vice President, Nuclear Operations. Full audits shall be forwarded to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.3.1 Activities which affect nuclear safety shall be conducted as follows:

a. Procedures required by Technical Specification 6.8 and other procedures which affect plant nuclear safety, and changes thereto, shall be prepared, reviewed and approved. Each such procedure or procedure change shall be reviewed by an individual/group other than the individual/ group which prepared the procedure or procedure change, but who may be from the same organization as the individual/ group which prepared the procedure or procedure change. Procedures other than Administrative Procedures will be approved as delineated in writing by the Director, Nuclear Plant Operations. The Director, Nuclear Plant Operations, will approve administrative procedures, security implementing procedures and emergency plan implementing procedures. Temporary approval to procedures which clearly do not change the intent of the approved procedures can be made by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License. For changes to procedures which may involve a change in intent of the approved procedures, the person authorized above to approve the procedure shall approve the change.

b. Proposed changes or modifications to plant nuclear safety-related structures, systems and components shall be reviewd as designated by the Director, Nuclear Plant Operations. Each such modification shall be designed as authorized by ~~Nuclear Engineering~~ Technical Services and shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modifications. Implementation of modifications to plant nuclear safety-related structures, systems and components shall be concurred in by the Director, Nuclear Plant Operations.

ATTACHMENT 3

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

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DESIGN FEATURES

SECTION | PAGE

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION, REACTOR TRIP, FEEDWATER ISOLATION, CONTROL ROOM ISOLATION, START DIESEL GENERATORS, CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER.					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Reactor Building Pressure-High-1	3	2	2	1, 2, 3	15*
d. Pressurizer Pressure - Low	3	2	2	1, 2, 3#	15*
e. Differential Pressure Between Steam Lines - High	3/steam line	2/steam line twice and 1/3 steam lines	2/steam line	1, 2, 3	15*

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. SAFETY INJECTION, REACTOR TRIP FEEDWATER ISOLATION, CONTROL ROOM ISOLATION START DIESEL GENERATORS, CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Reactor Building Pressure-High-1	S	R	M	N.A.	N.A	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure--Low	S	R	M	N.A	N.A.	N.A.	N.A.	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. Steam Line Pressure Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. REACTOR BUILDING SPRAY								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Reactor Building Pressure-- ~~High-High~~ - 3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

CONTAINMENT SYSTEMS

BASES

3/4.6.1.4 INTERNAL PRESSURE

The limitations on reactor building internal pressure ensure that 1) the reactor building structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of ~~3.0~~ psig and 3.5 2) the reactor building peak pressure does not exceed the design pressure of 57 psig during steam line break conditions.

The maximum peak pressure expected to be obtained from a steam line break event is 47.1 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 47.1 psig which is less than design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on reactor building average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a steam line break accident.

3/4.6.1.6 REACTOR BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 47.1 psig in the event of a steam line break accident. The measurement of containment tendon lift off force, the tensile tests of the tendon wires, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.

The tendon lift off forces are evaluated to ensure that 1) the rate of tendon force loss is within predicted limits, and 2) a minimum required prestress level exists in the containment. In order to assess the rate of force loss, the lift off force for a tendon is compared with the force predicted for the tendon times a reduction factor of 0.95. This resulting force is referred to as the 95% Base Value. The predicted tendon force is equal to the original stressing force minus losses due to elastic shortening of the tendon, stress relaxation of the tendon wires, and creep and shrinkage of the concrete. The 5% reduction on the predicted force is intended to compensate for both uncertainties in the prediction techniques for the losses and for inaccuracies in the lift-off force measurements.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Pool Area (RM-G8)	1	*	≤ 15 mR/hr	10^{-1} - 10^{4} mR/hr	25
b. Reactor Building Manipulator Crane Area (RM-G17A or RM-G17B)	1	6	≤ 1 R/hr	1 - 10^{5} mr/hr	28
c. Reactor Building Area					
i. High Range RM-G7 and High Range RM-G18	2	1, 2, 3 & 4	N/A	10 - 10^{7} R/hr 1 - 10^{7} R/hr	30
2. PROCESS MONITORS					
a. Spent Fuel Pool Exhaust - Ventilation System (RM-A6)					
i. Gaseous Activity	1	**	≤ 1 x 10^{-5} µCi/cc (Kr-85)	10 - 10^{6} cpm	27
ii. Particulate Activity	1	**	N/A	10 - 10^{6} cpm	27
b. Containment					
i. Gaseous Activity - Purge & Exhaust Isolation (RM-A4)	1	6	≤ 2 x background***	10 - 10^{6} cpm	28
ii. Particulate and Gaseous Activity (RM-A2) - RCS Leakage Detection	1	1, 2, 3 & 4	N/A	10 - 10^{6} cpm	26
c. Control Room Isolation (RM-A1)	1	ALL MODES	≤ 2 x background	10 - 10^{6} cpm	29

* With fuel in the storage pool or building

** With irradiated fuel in the storage pool

*** Alarm/trip setpoint will be per the Operational Dose Calculation Manual when purge exhaust operations are in progress

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. AREA MONITORS				
a. Spent Fuel Pool Area (RM-G8)	S	R	M	*
b. Reactor Building Manipulator Crane Area (RM-G17A or RM-G17B)	S	R	M	6
c. Reactor Building Area				
i. High Range (RM-G7)	S	R***	M	1, 2, 3 & 4
ii. High Range (RM-G18)	S	R***	M	1, 2, 3 & 4
2. PROCESS MONITORS				
a. Spent Fuel Pool Exhaust Area - Ventilation System (RM-A6)				
i. Gaseous Activity	S	R	M	**
ii. Particulate Activity	S	R	M	**
b. Containment				
i. Gaseous Activity - Purge & Exhaust Isolation (RM-A4)	S	R	M	6
ii. Particulate and Gaseous Activity - RCS Leakage Detection (RM-A2)	S	R	M	1, 2, 3 & 4
c. Control Room Isolation (RM-A1)	S	R	M	All MODES
d. Noble Gas Effluent Monitors (High Range)				
i. Main Plant Vent (RM-A13)	S	R	M	1, 2, 3 & 4
ii. Main Steam Lines (RM-G19A, B, C)	S	R	M	1, 2, 3 & 4
iii. Reactor Building Purge Supply & Exhaust System (RM-A14)	S	R	M	1, 2, 3 & 4

*With fuel in the storage pool or building

**With irradiated fuel in the storage pool

***Channel Calibration will consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source.