

## TABLE OF CONTENTS

<u>Section</u>		<u>Page</u>
2	<u>SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS</u>	2-1
2.1	<u>Safety Limits, Reactor Core</u>	2-1
2.2	<u>Safety Limits, Reactor System Pressure</u>	2-4
2.3	<u>Limiting Safety System Settings, Protection Instrumentation</u>	2-5
3	<u>LIMITING CONDITIONS FOR OPERATION</u>	3-1
3.0	<u>General Action Requirements</u>	3-1
3.1	<u>Reactor Coolant System</u>	3-1a
3.1.1	Operational Components	3-1a
3.1.2	Pressurization, Heatup and Cooldown Limitations	3-3
3.1.3	Minimum Conditions for Criticality	3-6
3.1.4	Reactor Coolant System Activity	3-8
3.1.5	Chemistry	3-10
3.1.6	Leakage	3-12
3.1.7	Moderator Temperature Coefficient of Reactivity	3-16
3.1.8	Single Loop Restrictions	3-17
3.1.9	Low Power Physics Testing Restrictions	3-18
3.1.10	Control Rod Operation (Deleted)	3-18a
3.1.11	Reactor Internal Vent Valves	3-18c
3.1.12	Pressurizer Power Operated Relief Valve (PORV), Block Valve, and Low Temperature Overpressure Protection (LTOP)	3-18d
3.1.13	Reactor Coolant System Vents	3-18f
3.2	<u>Deleted</u>	3-19
3.3	<u>Emergency Core Cooling, Reactor Building Emergency Cooling and Reactor Building Spray Systems</u>	3-21
3.4	<u>Decay Heat Removal (DHR) Capability</u>	3-25
3.4.1	Reactor Coolant System (RCS) Temperature Greater than 250 Degrees F	3-25
3.4.2	RCS Temperature Less Than or Equal to 250 Degrees F	3-26a
3.5	<u>Instrumentation Systems</u>	3-27
3.5.1	Operational Safety Instrumentation	3-27
3.5.2	Control Rod Group and Power Distribution Limits	3-33
3.5.3	Engineered Safeguards Protection System Actuation Setpoints	3-37
3.5.4	Incore Instrumentation (Deleted)	3-38
3.5.5	Accident Monitoring Instrumentation	3-40a
3.5.6	Deleted	3-40f
3.5.7	Remote Shutdown System	3-40g
3.6	<u>Reactor Building</u>	3-41
3.7	<u>Unit Electrical Power System</u>	3-42
3.8	<u>Fuel Loading and Refueling</u>	3-44
3.9	<u>Deleted</u>	3-46
3.10	<u>Miscellaneous Radioactive Materials Sources</u>	3-46
3.11	<u>Handling of Irradiated Fuel</u>	3-55
3.12	<u>Reactor Building Polar Crane</u>	3-57
3.13	<u>Secondary System Activity</u>	3-58
3.14	<u>Flood</u>	3-59
3.14.1	Periodic Inspection of the Dikes Around TMI	3-59
3.14.2	Flood Condition for Placing the Unit in Hot Standby	3-60
3.15	<u>Air Treatment Systems</u>	3-61
3.15.1	Emergency Control Room Air Treatment System	3-61
3.15.2	Reactor Building Purge Air Treatment System (Deleted)	3-62a
3.15.3	Auxiliary and Fuel Handling Building Air Treatment System	3-62c
3.15.4	Fuel Handling Building ESF Air Treatment System	3-62e

## TABLE OF CONTENTS

<u>Section</u>	<u>Page</u>	
4.8	<u>MAIN STEAM ISOLATION VALVES</u>	4-51
4.9	<u>DECAY HEAT REMOVAL (DHR) CAPABILITY - PERIODIC TESTING</u>	4-52
4.9.1	REACTOR COOLANT SYSTEM (RCS) TEMPERATURE GREATER THAN 250 DEGREES F	4-52
4.9.2	RCS TEMPERATURE LESS THAN OR EQUAL TO 250 DEGREES F	4-52a
4.10	<u>REACTIVITY ANOMALIES</u>	4-53
4.11	<u>REACTOR COOLANT SYSTEM VENTS</u>	4-54
4.12	<u>AIR TREATMENT SYSTEMS</u>	4-55
4.12.1	EMERGENCY CONTROL ROOM AIR TREATMENT SYSTEM	4-55
4.12.2	REACTOR BUILDING PURGE AIR TREATMENT SYSTEM (DELETED)	4-55b
4.12.3	AUXILIARY AND FUEL HANDLING BUILDING AIR TREATMENT SYSTEM	4-55d
4.13	<u>RADIOACTIVE MATERIALS SOURCES SURVEILLANCE</u>	4-56
4.14	<u>DELETED</u>	4-56
4.15	<u>MAIN STEAM SYSTEM INSERVICE INSPECTION</u>	4-58
4.16	<u>REACTOR INTERNALS VENT VALVES SURVEILLANCE</u>	4-59
4.17	<u>SHOCK SUPPRESSORS (SNUBBERS)</u>	4-60
4.18	<u>FIRE PROTECTION SYSTEMS (DELETED)</u>	4-72
4.19	<u>OTSG TUBE INSERVICE INSPECTION</u>	4-77
4.19.1	STEAM GENERATOR SAMPLE SELECTION AND INSPECTION METHODS	4-77
4.19.2	STEAM GENERATOR TUBE SAMPLE SELECTION AND INSPECTION	4-77
4.19.3	INSPECTION FREQUENCIES	4-79
4.19.4	ACCEPTANCE CRITERIA	4-80
4.19.5	REPORTS	4-81
4.20	<u>REACTOR BUILDING AIR TEMPERATURE</u>	4-86
4.21	<u>RADIOACTIVE EFFLUENT INSTRUMENTATION (DELETED)</u>	4-87
4.21.1	RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION (DELETED)	4-87
4.21.2	RADIOACTIVE GASEOUS PROCESS AND EFFLUENT MONITORING INSTRUMENTATION (DELETED)	4-87
4.22	<u>RADIOACTIVE EFFLUENTS (DELETED)</u>	4-87
4.22.1	LIQUID EFFLUENTS (DELETED)	4-87
4.22.2	GASEOUS EFFLUENTS (DELETED)	4-87
4.22.3	SOLID RADIOACTIVE WASTE (DELETED)	4-87
4.22.4	TOTAL DOSE (DELETED)	4-87
4.23.1	MONITORING PROGRAM (DELETED)	4-87
4.23.2	LAND USE CENSUS (DELETED)	4-87
4.23.3	INTERLABORATORY COMPARISON PROGRAM (DELETED)	4-87

- 3.8.8 If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made.
- 3.8.9 The reactor building purge isolation valves, and associated radiation monitors which initiate purge isolation, shall be tested and verified to be operable no more than 7 days prior to initial fuel movement in the reactor building.
- 3.8.10 Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 72 hours.
- 3.8.11 During the handling of irradiated fuel in the Reactor Building at least 23 feet of water shall be maintained above the level of the reactor pressure vessel flange. If the water level is less than 23 feet above the reactor pressure vessel flange, place the fuel assembly(s) being handled into a safe position, then cease fuel handling until the water level has been restored to 23 feet or greater above the reactor pressure vessel flange.

### Bases

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.7 of the UFSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The decay heat removal pump is used to maintain a uniform boron concentration. The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core (Reference 1). The boron concentration will be sufficient to maintain the core  $k_{\text{eff}} \leq 0.99$  if all the control rods were removed from the core, however only a few control rods will be removed at any one time during fuel shuffling and replacement. The  $k_{\text{eff}}$  with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

Per Specification 3.8.6 and 3.8.7, the personnel and emergency air lock doors, and penetrations may be open during movement of irradiated fuel in the containment provided a minimum of one door in each of the air locks, and penetrations are capable of being closed in the event of a fuel handling accident, and the plant is in REFUELING SHUTDOWN or REFUELING OPERATION with at least 23 feet of water above the fuel seated within the reactor pressure vessel. The minimum water level specified is the basis for the accident analysis assumption of a decontamination factor of 200 for the release to the containment atmosphere from the postulated damaged fuel rods located on top of the fuel core seated in the reactor vessel. Should a fuel handling accident occur inside containment, a minimum of one door in each personnel and emergency air lock, and the open penetrations will be closed following an evacuation of containment. Administrative controls will be in place to assure closure of at least one door in each air lock, as well as other open containment penetrations, following a containment evacuation.

Provisions for equivalent isolation methods in Technical Specification 3.8.7 include use of a material (e.g. temporary sealant) that can provide a temporary, atmospheric pressure ventilation barrier for other containment penetrations during fuel movements.

Specification 3.8.9 requires testing of the reactor building purge isolation system. This system consists of the four reactor building purge valves and the associated reactor building purge radiation monitor(s). The test verifies that the purge valves will automatically close when they receive initiation signals from the radiation detectors that monitor reactor building purge exhaust. The test is performed no more than 7 days prior to the start of fuel movement in the reactor building to ensure that the monitors, purge valves, and associated interlocks are functioning prior to operations that could result in a fuel handling accident within the reactor building. For conservatism, the Fuel Handling Accident analysis assumes that the four purge valves remain open.

Specification 3.8.10 is required as the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 72 hours (Reference 2).

REFERENCES

- (1) UFSAR, Section 14.2.2.1 - "Fuel Handling Accident"
- (2) UFSAR, Section 14.2.2.1(2) - "FHA Inside Containment"

**3.15.2 REACTOR BUILDING PURGE AIR TREATMENT SYSTEM**

Deleted

3-62a

THIS PAGE LEFT BLANK INTENTIONALLY

3-62b

Amendment No. ~~55, 108, 157, 226~~, 245

**4.12.2 REACTOR BUILDING PURGE AIR TREATMENT SYSTEM**

**Deleted**

THIS PAGE LEFT BLANK INTENTIONALLY

4-55c

Amendment No. ~~55, 108, 157, 170, 218, 226, 240~~, 245