Attachment I to JPN-93-012

# PROPOSED TECHNICAL SPECIFICATION CHANGES
# MISCELLANEOUS ADMINISTRATIVE CHANGES

(JPTS-92-001)

**New York Power Authority**

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

# TECHNICAL SPECIFICATIONS

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1.1 (cont'd)

B. Core Thermal Power Limit (Reactor Pressure ≤785 psig)

When the reactor pressure is ≤785 psig or core flow is less than or equal to 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

C. Power Transient

To ensure that the Safety Limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

2.1 (cont'd)

b. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

APRM - The APRM flux scram setting shall be ≤15 percent of rated neutron flux with the Reactor Mode Switch in Startup/Hot Standby or Refuel.

c. APRM Flux Scram Trip Settings (Run Mode)

(1) Flow Referenced Neutron Flux Scram Trip Setting

When the Mode Switch is in the RUN position, the APRM flow referenced flux scram trip setting shall be less than or equal to the limit specified in Table 3.1-1. This setting shall be adjusted during single loop operation when required by Specification 3.5.J.

For no combination of recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 117% of rated thermal power.

2.1 BASES (Cont'd)

B. Not Used

C. References

1. (Deleted)

2. "General Electric Standard Application for Reactor Fuel", NEDE 24011-P-A (Approved revision number applicable at time that reload fuel analyses are performed).

3. (Deleted)

4. FitzPatrick Nuclear Power Plant Single-Loop Operation, NEDO-24281, August, 1980.

1.2 and 2.2 BASES

The reactor coolant pressure boundary integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this boundary be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1,325 psig as measured by the vessel steam space pressure indicator is equivalent to 1,375 psig at the lowest elevation of the Reactor Coolant System. The 1,375 psig value is derived from the design pressures of the reactor pressure vessel and reactor coolant system piping. The respective design pressures are 1250 psig at 575°F for the reactor vessel, 1148 psig at 568°F for the recirculation suction piping and 1274 psig at 575° for the discharge piping. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: 1965 ASME Boiler and Pressure Vessel Code, Section III for pressure vessel and 1969 ANSI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10 percent over design pressure (110% x 1,250 = 1,375 psig) and the ANSI Code permits pressure transients up to 20 percent over the design pressure (120% x 1,150 = 1,380 psig). The safety limit pressure of 1,375 psig is referenced to the lowest elevation of the Reactor Coolant System.

The current reload analysis shows that the main steam isolation valve closure transient, with flux scram, is the most severe event resulting directly in a reactor coolant system pressure increase. The reactor vessel pressure code limit of 1,375 psig, given in FSAR Section 4.2, is above the peak pressure produced by the event above. Thus, the pressure safety limit (1,375 psig) is well above the peak pressure that can result from reasonably expected overpressure transients. (See current reload analysis for the curve produced by this analysis.) Reactor pressure is continuously indicated in the control room during operation.

A safety limit is applied to the Residual Heat Removal System (RHRS) when it is operating in the shutdown cooling mode. When operating in the shutdown cooling mode, the RHRS is included in the reactor coolant system.

The numerical distribution of safety/relief valve setpoints shown in 2.2.1.B (2 @ 1090 psi, 2 @ 1105 psi, 7 @ 1140 psi) is justified by analyses described in the General Electric report NEDO-24129-1, Supplement 1, and assures that the structural acceptance criteria set forth in the Mark I Containment Short Term Program are satisfied.

## 3.1 BASES

A. The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.

2. Preserve the integrity of the Reactor Coolant System.

3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The Reactor Protection System is of the dual channel type (Reference subsection 7.2 FSAR). The System is made up of two independent trip systems, each having two subchannels of tripping devices. Each subchannel has an input from at least one instrument channel which monitors a critical parameter.

The outputs of the subchannels are combined in a 1 out of 2 logic; i.e., an input signal on either one or both of the subchannels will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the intent of IEEE-279 (1971) for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the average power range monitor (APRM) channel the intermediate range monitor (IRM) channels, the scram discharge volume, the main steam isolation valve closure and the turbine stop valve closure, each subchannel has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the affected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved.

Three APRM instrument channels are provided for each protection trip system. APRM's A and E operate contacts in one subchannel and APRM's C and E operate contacts in the other

3.2 (cont'd)

E. Drywell Leak Detection

The limiting conditions of operation for the instrumentation that monitors drywell leak detection are given in Table 3.2-5.

F. (Deleted)

G. Recirculation Pump Trip

The limiting conditions for operation for the instrumentation that trip(s) the recirculation pumps as a means of limiting the consequences of a failure to scram during an anticipated transient are given in Table 3.2-7.

H. Accident Monitoring Instrumentation

The limiting conditions for operation of the instrumentation that provides accident monitoring are given in Table 3.2-8.

I. 4kv Emergency Bus Undervoltage Trip

The limiting conditions for operation for the instrumentation that prevents damage to electrical equipment or circuits as a result of either a degraded or loss-of-voltage condition on the emergency electrical buses are given in Table 3.2-2.

4.2 (cont'd)

E. Drywell Leak Detection

Instrumentation shall be calibrated and checked as indicated in Table 4.2-5.

F. (Deleted)

G. Recirculation Pump Trip

Instrumentation shall be functionally tested and calibrated as indicated in Table 4.2-7.

System logic shall be functionally tested as indicated in Table 4.2-7.

H. Accident Monitoring Instrumentation

Instrumentation shall be demonstrated operable by performance of a channel check and channel calibration as indicated in Table 4.2-8.

TABLE 4.2-1

## MINIMUM TEST AND CALIBRATION FREQUENCY FOR PCIS

| | Instrument Channel (8) | Instrument Functional Test | Calibration Frequency | Instrument Check (4) |
|---|---|---|---|---|
| 1) | Reactor High Pressure (Shutdown Cooling Permissive) | (1) | Once/3 months | None |
| 2) | Reactor Low-Low-Low Water Level | (1)(5) | (15) | Once/day |
| 3) | Main Steam High Temp. | (1)(5) | (15) | Once/day |
| 4) | Main Steam High Flow | (1)(5) | (15) | Once/day |
| 5) | Main Steam Low Pressure | (1)(5) | (15) | Once/day |
| 6) | Reactor Water Cleanup High Temp. | (1) | Once/3 months | None |
| 7) | Condenser Low Vacuum | (1)(5) | (15) | Once/day |

| | Logic System Functional Test (7) (9) | Frequency |
|---|---|---|
| 1) | Main Steam Line Isolation Valves<br>Main Steam Line Drain Valves<br>Reactor Water Sample Valves | Once/6 months |
| 2) | RHR - Isolation Valve Control<br>Shutdown Cooling Valves | Once/6 months |
| 3) | Reactor Water Cleanup Isolation | Once/6 months |
| 4) | Drywell Isolation Valves<br>TIP Withdrawal<br>Atmospheric Control Valves | Once/6 months |
| 5) | Standby Gas Treatment System<br>Reactor Building Isolation | Once/6 months |

NOTE: See notes following Table 4.2-5.

Amendment No. 37, 89, 136, 181, 182,

TABLE 4.2-2

## MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS

| | Instrument Channel | Instrument Functional Test | Calibration Frequency | Instrument Check (4) |
|---|---|---|---|---|
| 1) | Reactor Water Level | (1)(5) | (15) | Once/day |
| 2a) | Drywell Pressure (non-ATTS) | (1) | Once/3 months | None |
| 2b) | Drywell Pressure (ATTS) | (1)(5) | (15) | Once/day |
| 3a) | Reactor Pressure (non-ATTS) | (1) | Once/3 months | None |
| 3b) | Reactor Pressure (ATTS) | (1)(5) | (15) | Once/day |
| 4) | Auto Sequencing Timers | None | Once/operating cycle | None |
| 5) | ADS - LPCI or CS Pump Disch. | (1) | Once/3 months | None |
| 6) | Trip System Bus Power Monitors | (1) | None | None |
| 8) | Core Spray Sparger d/p | (1) | Once/3 months | Once/day |
| 9) | Steam Line High Flow (HPCI & RCIC) | (1)(5) | (15) | Once/day |
| 10) | Steam Line/Area High Temp.(HPCI & RCIC) | (1)(5) | (15) | Once/day |
| 12) | HPCI & RCIC Steam Line Low Pressure | (1)(5) | (15) | Once/day |
| 13) | HPCI & RCIC Suction Source Levels | (1) | Once/3 months | None |
| 14) | 4kV Emergency Bus Under-Voltage (Loss-of-Voltage, Degraded Voltage LOCA and non-LOCA) Relays and Timers. | Once/operating cycle | Once/operating cycle | None |
| 15) | HPCI & RCIC Exhaust Diaphragm Pressure High | (1) | Once/3 months | None |
| 17) | LPCI/Cross Connect Valve Position | Once/operating cycle | None | None |

NOTE: See notes following Table 4.2-5.

Amendment No. ~~14~~, ~~43~~, ~~53~~, ~~89~~, ~~106~~, ~~120~~, ~~160~~, ~~181~~,

3.5 (cont'd)

condition, that pump shall be considered inoperable for purposes of satisfying Specifications 3.5.A, 3.5.C, and 3.5.E.

H. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure shall be within limits based on applicable APLHGR limit values which have been approved for the respective fuel and lattice types. These values are specified in the Core Operating Limits Report. If at anytime during reactor power operation greater than 25% of rated power it is determined that the limiting value for APLHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, an orderly reactor power reduction shall be commenced immediately. The reactor power shall be reduced to less than 25% of rated power within the next four hours, or until the APLHGR is returned to within the prescribed limits.

4.5 (cont'd)

2. Following any period where the LPCI subsystems or core spray subsystems have not been maintained in a filled condition; the discharge piping of the affected subsystem shall be vented from the high point of the system and water flow observed.

3. Whenever the HPCI or RCIC System is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI or RCIC shall be vented from the high point of the system, and water flow observed on a monthly basis.

4. The level switches located on the Core Spray and RHR System discharge piping high points which monitor these lines to insure they are full shall be functionally tested each month.

H. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq 25\%$ rated thermal power.

3.6 LIMITING CONDITIONS FOR OPERATION

3.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the operating status of the Reactor Coolant System.

Objective:

To assure the integrity and safe operation of the Reactor Coolant System.

Specification:

A. Pressurization and Thermal Limits

1. Reactor Vessel Head Stud Tensioning

The reactor vessel head bolting studs shall not be under tension unless the temperatures of the reactor vessel flange and the reactor head flange are at least 90°F.

2. In-Service Hydrostatic and Leak Tests

During in-service hydrostatic or leak testing the Reactor Coolant System pressure and temperature shall be on or to the right of curve A shown in Figure 3.6-1 Part 1, 2, or 3 and the maximum temperature change during any one hour period shall be:

4.6 SURVEILLANCE REQUIREMENTS

4.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the periodic examination and testing requirements for the Reactor Coolant System.

Objective:

To determine the condition of the Reactor Coolant System and the operation of the safety devices related to it.

Specification:

A. Pressurization and Thermal Limits

1. Reactor Vessel Head Stud Tensioning

When in the cold condition, the reactor vessel head flange and the reactor vessel flange temperatures shall be recorded:

a. Every 12 hours when the reactor vessel head flange is ≤120°F and the studs are tensioned.

b. Every 30 minutes when the reactor vessel head flange is ≤100°F and the studs are tensioned.

c. Within 30 minutes prior to and every 30 minutes during tensioning of reactor vessel head bolting studs.

2. In-Service Hydrostatic and Leak Tests

During hydrostatic and leak testing the Reactor Coolant System pressure and temperature shall be recorded every 30 minutes until two consecutive temperature readings are within 5°F of each other.

3.6 (cont'd)

B. Deleted

C. Coolant Chemistry

1. The reactor coolant system radioactivity concentration in water shall not exceed the equilibrium value of 3.1 μCi/gm of dose equivalent I-131. This limit may be exceeded, following a power transient, for a maximum of 48 hours. During this iodine activity transient the iodine concentrations shall not exceed the equilibrium limits by more than a factor of 10 whenever the main steamline isolation valves are open. The reactor shall not be operated more than 5 percent of its annual power operation under this exception to the equilibrium limits. If the iodine concentration exceeds the equilibrium limit by more than a factor of 10, the reactor shall be placed in a cold condition within 24 hours.

4.6 (cont'd)

7. Reactor Vessel Flux Monitoring

The reactor vessel Flux Monitoring Surveillance Program complies with the intent of the May, 1983 revision to 10 CFR 50, Appendices G and H. The next flux monitoring surveillance capsule shall be removed after 15 effective full power years (EFPYs) and the test procedures and reporting requirements shall meet the requirements of ASTM E 185-82.

B. Deleted

C. Coolant Chemistry

1. a. A sample of reactor coolant shall be taken at least every 96 hours and analyzed for gross gamma activity.

   b. Isotopic analysis of a sample of reactor coolant shall be made at least once/month.

   c. A sample of reactor coolant shall be taken prior to startup and at 4 hour intervals during startup and analyzed for gross gamma activity.

   d. During plant steady state operation and following an offgas activity increase (at the Steam Jet Air Ejectors) of 10,000 μCi/sec within a 48 hour period or a power level change of ≥ 20 percent of full rated power/hr reactor coolant samples shall be taken and analyzed for gross gamma activity. At least three samples will be taken at 4 hour intervals. These sampling requirements may be omitted whenever the equilibrium I-131 concentration in the reactor coolant is less than 0.007 μCi/ml.

4.6 (cont'd)

e. If the gross activity counts made in accordance with a, c, and d above indicate a total iodine concentration in excess of 0.007 $\mu$Ci/ml, a quantative determination shall be made for I-131 and I-133.

2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 lb/hr except as specified in 3.6.C.3:
   Conductivity 2 $\mu$mho/cm
   Chloride ion 0.1 ppm

3. For reactor startups the maximum value for conductivity shall not exceed 10 $\mu$mho/cm and the maximum value for chloride ion concentration shall not exceed 0.1ppm, for the first 24 hours after placing the reactor in the power operating condition. During reactor shutdowns, specification 3.6.C.4 will apply.

2. During startups and at steaming rates below 100,000 lb/hr, and when the conductivity of the reactor coolant exceeds 2 $\mu$mhos/cm, a sample of reactor coolant shall be taken every 4 hr and analyzed for conductivity and chloride content.

3. a. With steaming rates greater than or equal to 100,000 lb/hr, a reactor coolant sample shall be taken at least every 96 hours and whenever the continuous conductivity monitors indicate abnormal conductivity (other than short-term spikes), and analyzed for conductivity and chloride ion content.

   b. When the continuous conductivity monitor is inoperable, a reactor coolant sample shall be taken at least daily and analyzed for conductivity and chloride ion content.

3.6 (cont'd)

F. Structural Integrity

The structural integrity of the Reactor Coolant System shall be maintained at the level required by the original acceptance standards throughout the life of the Plant.

G. Jet Pumps

Whenever the reactor is in the startup/hot standby or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, the reactor shall be placed in a cold condition within 24 hours.

4.6 (cont'd)

F. Structural Integrity

1. Nondestructive inspections shall be performed on the ASME Boiler and Pressure Vessel Code Class 1, 2 and 3 components and supports in accordance with the requirements of the weld and support inservice inspection program. This inservice inspection program is based on an NRC approved edition of, and addenda to, Section XI of the ASME Boiler and Pressure Vessel Code which is in effect 12 months or less prior to the beginning of the inspection interval.

2. An augmented inservice inspection program is required for those high stressed circumferential piping joints in the main steam and feedwater lines larger than 4 inches in diameter, where no restraint against pipe whip is provided. The augmented in-service inspection program shall consist of 100 percent inspection of these welds per inspection interval.

3. An Inservice Inspection Program for piping identified in the NRC Generic Letter 88-01 shall be implemented in accordance with NRC staff positions on schedules, methods, personnel, and sample expansion included in this Generic Letter, or in acordance with alternate measures approved by the NRC staff.

G. Jet Pumps

Whenever there is recirculation flow with the reactor in the startup/hot standby or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

3.6 and 4.6 BASES (cont'd)

B. Deleted

C. Coolant Chemistry

A radioactivity concentration limit of 20 μCi/ml total iodine can be reached if the gaseous effluents are near the limit as set forth in Radiological Effluent Technical Specification Section 3.2.a if there is a failure or a prolonged shutdown of the cleanup demineralizer.

In the event of a steam line rupture outside the drywell, with this coolant activity level, the resultant radiological dose at the site boundary would be 33 rem to the thyroid, under adverse meteorological conditions assuming no more than 3.1 μCi/gm of dose equivalent I-131. The reactor water sample will be used to assure that the limit of Specification 3.6.C is not exceeded. The total radioactive iodine activity would not be expected to change rapidly over a period of 96 hours. In addition, the trend of the stack offgas release rate, which is continuously monitored, is a good indicator of the trend of the iodine activity in the reactor coolant. Also during reactor startups and large power changes which could affect iodine levels, samples of reactor coolant shall be analyzed to insure iodine concentrations are below allowable levels. Analysis is required whenever the I-131 concentration is within a factor of 100 of its allowable equilibrium value. The necessity for continued sampling following power and offgas transients will be reviewed within 2 years of initial plant startup.

The surveillance requirements 4.6.C.1 may be satisfied by a continuous monitoring system capable of determining the total iodine concentration in the coolant on a real time basis, and annunciating at appropriate concentration levels such that sampling for isotopic analysis can be initiated. The design details of such a system must be submitted for evaluation and accepted by the Commission prior to its implementation and incorporation in these Technical Specifications.

Since the concentration of radioactivity in the reactor coolant is not continuously measured, coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the offgas system and on the main steam lines.

Materials in the Reactor Coolant System are primarily 304 stainless steel and Zircaloy fuel cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on chloride concentration and conductivity. The most important limit is that placed on chloride concentration to prevent stress corrosion cracking of the stainless steel. The attached graph, Fig. 4.6-1, illustrates the results of tests on stressed 304 stainless steel specimens. Failures occurred at concentrations above the curve; no failures occurred at concentrations below the curve. According to the data, allowable chloride concentrations could be set several orders of magnitude above the established limit, at the oxygen concentration (0.2-0.3 ppm) experienced during power operation. Zircaloy does not exhibit similar stress corrosion failures.

However, there are various conditions under which the dissolved oxygen content of the reactor coolant water could be higher than 0.2-0.3 ppm, such as refueling, reactor startup, and hot standby. During these periods with steaming rates less

3.6 and 4.6 BASES (cont'd)

than 100,000 lb/hr, a more restrictive limit of 0.1 ppm has been established to assure the chloride-oxygen combinations of Fig. 4.6-1 are not exceeded. At steaming rates of at least 100,000 lb/hr, boiling occurs causing deaeration of the reactor water, thus maintaining oxygen concentration at low levels.

When conductivity is in its proper normal range, pH and chloride and other impurities affecting conductivity must also be within their normal ranges. When and if conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This is not necessarily the case. Conductivity could be high due to the presence of a neutral salt; e.g., $Na_2SO_4$, which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are, in fact, high due to purposeful addition of additives. In the case of BWR's, however, where no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the condition include operation of the Reactor Cleanup System, reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the Reactor Water Cleanup System to reestablish the purity of the reactor coolant.

During startup periods, which are in the category of less than 100,000 lb/hr, conductivity may exceed 2 μmho/cm because of the initial evolution of gases and the initial evolution of gases and the initial addition of dissolved metals. During this period of time, when the conductivity exceeds 2 μmho/cm (other than short-term spikes), samples will be taken to assure the chloride concentration is less than 0.1 ppm.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses of the reactor coolant required by Specification 4.6.C.1 may be performed by a gamma scan.

D. <u>Coolant Leakage</u>

Allowable leakage rates of coolant from the Reactor Coolant System have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to make up Reactor Coolant System leakage in the event of loss of off-site a-c power. The normally expected background leakage due to equipment design and the detection capability for determining system

3.6 and 4.6 BASES (cont'd)

leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC-sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.D, on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm as specified in 3.6.D, the experimental and analytical data suggest a reasonable margin of safety such that leakage of this magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the Plant should be shut down to allow further investigation and corrective action.

The capacity of the drywell sump pumps is 100 gpm. and the capacity of the drywell equipment drain tank pumps is also 100 gpm. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

The performance of the Reactor Coolant Leakage Detection System will be evaluated during the first 5 years of plant operation, and the conclusions of this evaluation will be reported to the NRC.

It is estimated that the main steam line tunnel leakage detectors are capable of detecting a leak on the order of 3,500 lb/hr. The system performance will be evaluated during the first 5 years of plant operation, and the conclusions of the evaluation will be reported to the NRC.

The reactor coolant leakage detection systems consist of the drywell sump monitoring system and the drywell continuous atmosphere monitoring system. The drywell continuous atmosphere monitoring system utilizes a three-channel monitor to provide information on particulate, iodine and noble gas activities in the drywell atmosphere. Two independent and redundant systems are provided to perform this function. This system supplements the drywell sump monitoring system in detecting abnormal leakage that could occur from the reactor coolant system. In the event that the drywell continuous atmosphere monitoring system is inoperable, grab sample will be taken on a periodic basis to monitor drywell activity.

3.7 LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

Applicability:
Applies to the operating status of the primary and secondary containment systems.

Objective:
To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. The volume and temperature of the water in the torus shall be maintained within the following limits whenever the reactor is critical or whenever the reactor coolant temperature is greater than 212°F and irradiated fuel is in the reactor vessel:

   a. Maximum vent submergence level of 53 inches.

   b. Minimum vent submergence level of 51.5 inches.

   The torus water level may be outside the above limits for a maximum of four (4) hours during required operability testing of HPCI, RCIC, RHR, CS, and the Drywell-Torus Vacuum System.

   c. Maximum water temperature

      (1) During normal power operation maximum water temperature shall be 95°F.

4.7 SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

Applicability:
Applies to the primary and secondary containment integrity.

Objective:
To verify the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. The torus water level and temperature shall be monitored as specified in Table 4.2-8. The accessible interior surfaces of the drywell and above the water line of the torus shall be inspected at each refueling outage for evidence of deterioration. Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the torus shall be conducted before resuming power operation.

4.7 (cont'd)

Type A test shall be performed at each plant shutdown for refueling or approximately every 18 months, whichever occurs first, until two consecutive Type A tests meet the acceptance criteria.

b. Type B tests (Local leak rate testing of containment penetrations)

(1.) All preoperational and periodic Type B tests shall be performed by local pneumatic pressurization of the containment penetrations, either individually or in groups, at a pressure not less than Pa, and the gas flow to maintain Pa shall be measured.

(2.) Acceptance criteria

The combined leakage rate of all penetrations and valves subject to Type B and C tests shall be less than 0.60 La, with the exception of the valves sealed with fluid from a seal system.

4.7 (cont'd)

(5) Type C test.

Type C tests shall be performed during each reactor shutdown for refueling but in no case at intervals greater than two years.

(6) Other leak rate tests specified in Section 4.7d shall be performed during each reactor shutdown for refueling but in no case at intervals greater than two years.

f. Containment modification

Any major modification, replacement of a component which is part of the primary reactor containment boundary, or resealing a seal-welded door, performed after the preoperational leakage rate test shall be followed by either a Type A, Type B, or Type C test, as applicable, for the area affected by the modification. The measured leakage from this test shall be included in the test report. The acceptance criteria as appropriate, shall be met. Minor modifications, replacements, or resealing of seal-welded doors, performed directly prior to the conduct of a scheduled Type A test do not require a separate test.

## 3.7 BASES

### A. Primary Containment

The integrity of the primary containment and operation of the Emergency Core Cooling Systems in combination limit the offsite doses to values less than those specified in 10 CFR 100 in the event of a break in the Reactor Coolant System piping. Thus, containment integrity is required whenever the potential for violation of the Reactor Coolant System integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception to the requirement to maintain primary containment integrity is allowed during core loading and during low power physics testing when ready access to the reactor vessel is required. There will be no pressure on the system at this time, which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period, however, restrictive operating procedures and operation of the RWM in accordance with Specification 3.3.B.3 minimize the probability of an accident occurring. Procedures in conjunction with the Rod Worth Minimizer Technical Specifications limit individual control worth such that the drop of any in-sequence control rod would not result in a peak fuel enthalpy greater than 280 calories/gm. In the unlikely event that an excursion did occur, the reactor building and Standby Gas Treatment System, which shall be operational during this time, offers a sufficient barrier to keep offsite doses well within 10 CFR 100.

The pressure suppression pool water provides the heat sink for the Reactor Coolant System energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1,020 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 56 psig, the suppression chamber design pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber (updated FSAR Section 5.2).

3.7 BASES (cont'd)

complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

The Standby Gas Treatment System is designed to filter and exhaust the reactor building atmosphere to the main stack during secondary containment isolation conditions with a minimum release of radioactive materials from the reactor building to the environs. Both standby gas treatment fans are designed to automatically start upon containment isolation; however, only one fan is required to maintain the reactor building pressure at approximately a negative 1/4 in. water gage pressure; all leakage should be in-leakage. Each of the two fans has 100 percent capacity. If one Standby Gas Treatment System circuit is inoperable, the other circuit must be verified operable daily. This substantiates the availability of the operable circuit and results in no added risk; thus, reactor operation or refueling operation can continue. If neither circuit is operable, the Plant is brought to a condition where the system is not required.

While only a small amount of particulates is released form the Pressure Suppression Chamber System as a result of the loss-of-coolant accident, high-efficiency particulate filters are specified to minimize potential particulate release to the environment and to prevent clogging of the iodine filter. The high-efficiency filters have an efficiency greater than 99 percent for particulate matter larger than 0.3 micron. The minimum iodine removal efficiency is 99 percent. Filter banks will be replaced whenever significant changes in filter efficiency occur. Tests (11) of impregnated charcoal identical to that used in the filters indicated that shelf life up to 5 yr leads to only minor decreases in methyl iodine removal efficiency.

The 99 percent efficiency of the charcoal and particulate filters is sufficient to prevent exceeding 10CFR100 guidelines for the accidents analyzed. The analysis of the loss-of-coolant accident assumed a charcoal filter efficiency of 90 percent, and TID 14844 fission product source term. Hence, requiring 99 percent efficiency for both the charcoal and particulate filters provides adequate margin. A heater maintains relative humidity below 70 percent in order to assure the efficient removal of methyl iodine on the impregnated charcoal filters.

The operability of the Standby Gas Treatment System (SGTS) must be assured if a design basis loss of coolant accident (LOCA) occurs while the containment is being purged or vented through the SGTS. Flow from containment to the SGTS is via 6 inch Valve Number 27MOV-121. Since the maximum flow through the 6 inch line following a design basis LOCA is within the design capabilities of the SGTS, use of the 6 inch line assures the operability of the SGTS.

D. Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space

3.9 (cont'd)

3. From and after the time that one of the Emergency Diesel Generator Systems is made or found to be inoperable, continued reactor operation is permissible for a period not to exceed 7 days provided that the two incoming power sources are available and that the remaining Diesel Generator System is operable. At the end of the 7 day period, the reactor shall be placed in a cold condition within 24 hours, unless the affected diesel generator system is made operable sooner.

4. When both Emergency Diesel Generator Systems are made or found to be inoperable, a reactor shutdown shall be initiated within two hours and the reactor placed in a cold condition within 24 hours after initiation of shutdown.

4.9 (cont'd)

3. The emergency diesel generator system instrumentation shall be checked during the monthly generator test.

4. Once each operating cycle, the conditions under which the Emergency Diesel Generator System is required will be simulated to demonstrate that the pair of diesel generators will start, accelerate, force parallel, and accept the emergency loads in the prescribed sequence.

5. Once within one hour and at least once per twenty-four hours thereafter while the reactor is being operated in accordance with Specifications 3.9.B.1, 3.9.B.2, or 3.9.B.3 the availability of the operable Emergency Diesel Generators shall be demonstrated by manual starting and force paralleling where applicable.

3.9 (cont'd)

F. LPCI MOV Independent Power Supplies

1. Reactor shall not be made critical unless both independent power supplies, including the batteries, inverters and chargers and their associated buses (MCC-155 and MCC-165) are in service, except as specified below.

2. During power operation, if one independent power supply becomes unavailable, repairs shall be made immediately and continued reactor operation is permissible for a period not to exceed 7 days unless the unavailable train is made operable sooner. From and after the date one of the independent power supplies is made or found to be inoperable for any reason, the following would apply:

   a. The other independent power supply including its charger, inverter, battery and associated bus is operable.

   b. Pilot cell voltage, specific gravity and temperature and overall battery voltage are measured immediately and weekly thereafter for the operable independent power supply battery.

   c. The inoperable independent power supply shall be isolated from its associated LPCI MOV bus, and this bus will be manually switched to its alternate power source.

3.9 BASES (cont'd)

C. Diesel Fuel

Minimum on-site fuel oil requirements are based on operation of the emergency diesel generator systems at rated load for 7 days.

Additional diesel fuel can be delivered to the site within 48 hours.

If one of the Emergency Diesel Generator Systems is not operable, the plant shall be permitted to run for 7 days provided both sources of reserve power are operational. This is based on the following:

1. The operable Emergency Diesel Generator System is capable of carrying sufficient engineered safeguards and emergency core cooling system equipment to cover all loss-of-coolant accidents.

2. The reserve (offsite) power is highly reliable.

D. Not Used

E. Battery System

125 v DC power is supplied from two plant batteries each sized to supply the required equipment at design power following a loss-of-coolant accident with a concurrent loss of normal and reserve power. Each battery is provided with a charger sized to maintain the battery in a fully charged state while supplying normal operating loads.

F. LPCI MOV Independent Power Supplies

There are two LPCI MOV Independent Power Supplies each consisting of a charger, rectifier, inverter and battery. Each independent power supply charger-rectifier is normally fed from the emergency A-C power supply system to maintain the battery in a fully charged state. In the event of a LOCA each independent power supply is automatically isolated from the Emergency A-C power system. The battery and inverter have sufficient capacity to power the MOV's essential to the operation of the LPCI System. An alternate power source is provided for each LPCI MOV bus whereby in the event its independent power supply is out of service, the LPCI MOV bus may be energized directly from the Emergency A-C Power System.

3.9 BASES (cont'd)

G. Reactor Protection System Power Supplies

Each of two RPS divisions may be supplied power from it's respective RPS MG set or from an alternate source which derives power from the same electrical division. The MG sets and alternate sources for both divisions are provided with redundant, seismic qualified, class 1E electrical protection assemblies between the power source and the RPS bus. Any abnormal output type failure in either of the MG sets or alternate sources (if in service) would result in a trip of one or both of the electrical protection assemblies producing a half scram on that RPS division and retaining full scram capability in the other RPS division.

Limiting operating conditions in Section 3.9.G provide a high degree of assurance that RPS buses are protected as described above.

4.9 BASES (cont'd)

D. Not Used

E. Battery System

Measurements and electrical tests are conducted at specified intervals to provide indication of cell condition and to determine the discharge capability of the batteries. Performance and service tests are conducted in accordance with the recommendations of IEEE 450-1987.

F. LPCI MOV Independent Power Supply

Measurement and electrical tests are conducted at specified intervals to provide indication of cell condition, to determine the discharge capability of the battery. Performance and service tests are conducted in accordance with the recommendations of IEEE 450-1987.

G. Reactor Protection Power Supplies

Functional tests of the electrical protection assemblies are conducted once each six (6) months utilizing a built-in test device and once per operating cycle by performing an instrument calibration which verifies operation within the limits of Section 4.9.G.

A. High Pressure Water Fire Protection System (Cont'd)

3. If 1. above cannot be fulfilled, place the reactor in Hot Standby within six (6) hours and in Cold Shutdown within the following thirty (30) hours.

A. High Pressure Water Fire Protection System (Cont'd)

| Item | Frequency |
|---|---|
| h. Fire pump diesel engine by verifying the fuel storage tank contains at least 172 gallons of fuel. | Once/Month |
| i. Diesel fuel from each tank obtained in accordance with ASTM-D270-65 is within the acceptable limits for quality as per the following: | Once/Quarter |
| Flash Point - °F | 125°F min. |
| Pour Point - °F | 10°F max. |
| Water & Sediment | 0.05% max. |
| Ash | 0.01% max. |
| Distillation 90% Point | 540 min. |
| Viscosity (SSU) @ 100°F | 40 max. |
| Sulfur | 1% max. |
| Copper Strip Corrosion | No. 3 max. |
| Cetane # | 35 min. |
| j. Fire pump diesel engine by inspection during shut down in accordance with procedures prepared in conjunction with manufacturers recommendations and verifying the diesel, starts from ambient conditions on the auto start signal and operates for ≥20 minutes while loaded with the fire pump. | Once/18 months |

2. If the fire protection systems smoke and/or heat detectors in Tables 3.12.1 and 3.12.2 cannot be restored to an operable status within 14 days, a written report to the Commission outlining the action taken, the cause of inoperability and plans and schedule for restoring the detectors to an operable status shall be prepared and submitted within 30 days.

F. Fire Barrier Penetration Seals

1. All fire barrier penetrations, including cable penetration barriers, fire doors and fire dampers, in fire zone boundaries protecting safety related areas shall be functional.

2. With one or more of the required fire barrier penetrations non-functional, within one hour establish a continuous fire watch on at least one side of the affected penetration or verify the operability of fire detectors on at least one side of the non-functional fire barrier and establish an hourly fire watch patrol. Restore the non-functional fire barrier penetration(s) to functional status within 7 days or, in lieu of any other report required by Specification 6.9.A, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.B within 30 days outlining the action taken, the cause of the non-functional penetration and plans and schedule for restoring the fire barrier penetration(s) to functional status.

F. Fire Barrier Penetration Seals

1. All fire barrier penetration seals for each protected area shall be visually inspected once/1.5 years to verify functional integrity. For those fire barrier-penetrations that are not in the as-designed condition, an evaluation shall be performed to show that the modification has not degraded the fire rating of the fire barrier penetration.

2. Any repair of fire barrier penetration seals shall be followed by a visual inspection.

3.12 and 4.12 BASES

The Fire Protection System specifications provide pre-established minimum levels of operability to assure adequate fire protection during any operating condition including a design basis accident or safe shutdown earthquake.

A. The high pressure water fire protection system is supplied by redundant vertical turbine pumps, one diesel driven and one electric motor driven, each design rated 2500 gpm at 125 psig discharge pressure. Both pumps take suction from the plant intake cooling water structures from Lake Ontario. The high pressure water fire protection header is normally maintained at greater than 115 psig by a pressure maintenance subsystem. If pressure decreases, the fire pumps are automatically started by their initiation logic to maintain the fire protection system header pressure. Each pump, together with its manual and automatic initiation logic combined makes up a redundant high pressure water fire pump.

   A third fire pump, diesel-driven, has been installed and is set to automatically actuate upon decreasing pressure after the actuation of the first two fire pumps. No credit is taken for this pump in any analyses and the requirements of Technical Specifications 3.12 and 4.12 do not apply.

   Pressure Maintenance subsystem checks, valve position checks, system flushes and comprehensive pump and system flow and/or performance tests including logic and starting subsystem tests provide for the early detection and correction of component failures thus ensuring high levels of operability.

B. Safety related equipment areas protected by water spray or sprinklers are listed in Table 3.12.1. Whenever any of the protected areas, spray or sprinklers are inoperable continuous fire detection and backup fire protection equipment is available in the area where the water spray and/or sprinkler protection was lost.

   Performance of the tests and inspections listed in Table 4.12.1 will prevent and detect nozzle blockage or breakage and verify header integrity to ensure operability.

C. The carbon dioxide systems provide total flood protection for eight different safety related areas of the plant from either a 3 ton or 10 ton storage unit as indicated in Table 3.12.2. Both $CO_2$ storage units are equipped with mechanical refrigeration units to maintain the storage tank content at 0°F with a resultant pressure of 300 psig. Automatic smoke and heat detectors are provided in the $CO_2$ protected areas and initiation is automatic and/or manual as indicated in Table 3.12.2. For any area in which the $CO_2$ protection is made or found to be inoperable, continuous fire detection is available and one or more large wheeled $CO_2$ fire extinguisher is also available for each area in which protection was lost.

   Weekly checks of storage tank pressure and level verify proper operation of the tank refrigeration units and availability of sufficient volume of $CO_2$ to extinguish a fire in any of the protected areas.

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5.5.B Bases

The spent fuel pool and high density fuel storage racks are Class I structures designed to store up to 2,797 fuel bundles. The storage racks are designed to maintain a subcritical configuration having a multiplication factor ($k_{eff}$) less than 0.95 for all possible operational and abnormal conditions. The nuclear criticality analyses for the Spent Fuel Racks (References 1 and 3) conclude that fresh fuel bundles with 3.3 w/o U-235 meet the 0.95 $k_{eff}$ limit. This design basis bundle was reanalyzed to determine its infinite lattice multiplication factor, $k_{\infty}$, when in a reactor core geometry (Reference 2). This $k_{\infty}$ was obtained under conservative calculational assumptions and reduced by 2.33 times the standard deviation in the calculation resulting in the Technical Specification limit of 1.36.

References:


1) Increased Spent Fuel Storage Modification, Stone & Webster Engineering Corporation, Boston, Mass. March 15, 1978.

2) General Electric letter, P. Van Dieman to G. Rorke, FitzPatrick Fuel Storage K-infinity Conversion, Revision 1, dated July 10, 1986.

3) Increased Storage Capacity for FitzPatrick Spent Fuel Pool, Holtec International, Mount Laurel, New Jersey, February, 1989.

2. An SRO or an SRO with a license limited to fuel handling shall directly supervise all Core Alterations. This person shall have no other duties during this time;

3. A fire brigade of five (5) or more members shall be maintained on site at all times. This excludes two (2) members of the minimum shift crew necessary for safe shutdown and any personnel required for other essential functions during a fire emergency;

4. In the event of illness or unexpected absence, up to two (2) hours is allowed to restore the shift crew or fire brigade to the minimum complement.

5. The Operations Manager, Assistant Operations Manager, Shift Supervisor and Assistant Shift Supervisor shall hold a SRO license and the Senior Nuclear Operator and the Nuclear Control Operator shall hold a RO license or an SRO license.

6. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, health physicists, auxiliary operators, and maintenance personnel who are working on safety-related systems.

   Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating.

   However, in the event that unforeseen problems require substantial amounts of overtime to be used or during extended periods of shutdown for refueling, major maintenance or major modifications, on a temporary basis, the following guidelines shall be followed:

   a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

   b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

   c. A break of at least eight hours should be allowed between work periods, including shift turnover time.

   d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

   Any deviation from the above guidelines shall be authorized by the Resident Manager or the General Manager - Operations, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Resident Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

6.3 PLANT STAFF QUALIFICATIONS

6.3.1 The minimum qualifications with regard to educational background and experience for plant staff positions shown in FSAR Figure 13.2-7 shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions; except for the Radiological and Environmental Services Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.3.2 The Shift Technical Advisor (STA) shall meet or exceed the minimum requirements of either Option 1 (Combined SRO/STA Position) or Option 2 (Continued use of STA Position), as defined in the Commission Policy Statement on Engineering Expertise on Shift, published in the October 28, 1985 Federal Register (50 FR 43621). When invoking Option 1, the STA role may be filled by the Shift Supervisor or Assistant Shift Supervisor. (1)

6.3.3 Any deviations will be justified to the NRC prior to an individual's filling of one of these positions.

NOTE:

(1) The 13 individuals who hold SRO licenses, and have completed the FitzPatrick Advanced Technical Training Program prior to the issuance of License Amendment 111, shall be considered qualified as dual-role SRO/STAs.

6.4 RETRAINING AND REPLACEMENT TRAINING

A training program shall be maintained under the direction of the Training Manager to assure overall proficiency of the plant staff organization. It shall consist of both retraining and replacement training and shall meet or exceed the minimum requirements of Section 5.5 of ANSI N18.1-1971.

The retraining program shall not exceed periods two years in length with a curriculum designed to meet or exceed the requalification requirements of 10 CFR 55.59. In addition, fire brigade training shall meet or exceed the requirements of NFPA 27-1975, except for Fire Brigade training sessions which shall be held at least quarterly. The effective date for implementation of fire brigade training is March 17, 1978.

6.5 REVIEW AND AUDIT

Two separate groups for plant operations have been constituted. One of these, the Plant Operating Review Committee (PORC), is an onsite review group. The other is an independent review and audit group, the offsite Safety Review Committee (SRC).

## 7.0 REFERENCES


(1) E. Janssen, "Multi-Rod Burnout at Low Pressure," ASME Paper 62-HT-26, August 1962.

(2) K.M. Backer, "Burnout Conditions for Flow of Boiling Water in Vertical Rod Clusters," AE-74 (Stockholm, Sweden), May 1962.

(3) FSAR Section 11.2.2.

(4) FSAR Section 4.4.3.

(5) I.M. Jacobs, "Reliability of Engineered Safety Features as a Function of Testing Frequency," Nuclear Safety, Vol. 9, No. 4, July-August 1968, pp 310-312.

(6) Benjamin Epstein, Albert Shiff, UCRL-50451, Improving Availability and Readiness of Field Equipment Through Periodic Inspection, July 16, 1968, p. 10, Equation (24), Lawrence Radiation Laboratory.

(7) I.M. Jacobs and P.W. Mariott, APED Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards - April 1969.

(8) Bodega Bay Preliminary Hazards Report, Appendix 1, Docket 50-205, December 28, 1962.

(9) C.H. Robbins, "Tests of a Full Scale 1/48 Segment of the Humbolt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.

(10) "Nuclear Safety Program Annual Progress Report for Period Ending December 31, 1966, Progress Report for Period Ending December 31, 1966, ORNL-4071."

(11) Section 5.2 of the FSAR.

(12) TID 20583, "Leakage Characteristics of Steel Containment Vessel and the Analysis of Leakage Rate Determinations."

(13) Technical Safety Guide, "Reactor Containment Leakage Testing and Surveillance Requirements," USAEC, Division of Safety Standards, Revised Draft, December 15, 1966.

(14) Section 14.6 of the FSAR.

(15) ASME Boiler and Pressure Vessel Code, Nuclear Vessels, Section III. Maximum allowable internal pressure is 62 psig.

(16) 10 CFR 50.54, Appendix J, "Reactor Containment Testing Requirements."

(17) 10 CFR 50, Appendix J, February 13, 1973.

Attachment II to JPN-93-012

**SAFETY EVALUATION FOR**
**PROPOSED TECHNICAL SPECIFICATION CHANGES**
**MISCELLANEOUS ADMINISTRATIVE CHANGES (JPTS-92-001)**

I. DESCRIPTION OF THE PROPOSED CHANGES

The proposed changes to the James A. FitzPatrick Technical Specifications are administrative and are addressed below.

Minor changes in format, such as type font, margins or hyphenation, are not described in this submittal. These changes are typographical in nature and do not affect the content of the Technical Specifications.

1. Page i, TABLE OF CONTENTS

   In Specification 3.2.F, replace the title "Surveillance Information Readouts" with the title "DELETED."

2. Page v, LIST OF TABLES

   In the title of Table 4.2-2, replace the word "System" with the word "Systems."

3. Page 8, Specification 2.1

   In the Specification outline numbering pattern, delete Section numbers "A." and "1."

4. Page 20, Bases 2.1

   a. Add a "B." to indicate Section 2.1.B and insert the text "Not Used."

   b. Add Amendment number "14," to the Amendment list at the bottom left corner of the page.

5. Page 29, Bases 1.2 and 2.2

   In the second paragraph, replace the parentheticals "(110% x 1,250 - 1,375 psig)" and "(120% x 1,150 - 1,380 psig)" with the parentheticals "(110% x 1,250 = 1,375 psig)" and "(120% x 1,150 = 1,380 psig)."

6. Page 32, Bases 3.1

   In the first paragraph, add an "A." to indicate Section 3.1.A.

7. Page 54, Specification 4.2.E

   In the first sentence, add a period to the end of the sentence.

8. Page 78, Table 4.2-1

   In the Logic System Functional Test, replace "Main Steam Line Isolation valves" with "Main Steam Line Isolation Valves."

9. Page 79, Table 4.2-2

   In the first column, remove the note "(8)" from the "Instrument Channel" heading.

10. Page 123, Specification 3.5.H

    In the third sentence replace the phrase "If anytime during" with the phrase "If at anytime during."

11. Page 136, Specification 3.6.A.2

    Move Specification 3.6.A.2 to line up with Specification 4.6.A.2.

12. Page 139, Specification 3.6.C.1 and 4.6.C.1

    a. In Specification 3.6.C.1 for the second and last sentences, replace the abbreviation "hr" with the word "hours" in two locations.

    b. In Specification 4.6.C.1.a replace the abbreviation "hr" with the word "hours."

    c. In Specification 4.6.C.1.c replace the abbreviation "hr" with the word "hour."

    d. In Specification 4.6.C.1.d for the first and second sentences, replace the abbreviation "hr" with the word "hour" in two locations.

13. Page 140, Specification 3.6.C.3 and 4.6.C.3

    a. In Specification 3.6.C.3 replace the abbreviation "hr" with the word "hours."

    b. In Specification 4.6.C.3.a replace the abbreviation "hr" with the word "hours."

14. Page 144, Specification 3.6.G

    Move Specification 3.6.G to line up with Specification 4.6.G.

15. Page 149, Bases 3.6.C and 4.6.C

    In the second paragraph replace the abbreviation "hr" with the word "hours."

16. Page 150, Bases 3.6.C and 4.6.C

    In the fourth paragraph replace the abbreviation "hr" with the word "hours."

17. Page 151, Bases 3.6.D and 4.6.D

    a. In the Section outline numbering pattern replace the phrase "3.6 4.6 BASES (cont'd)" with the phrase "3.6 and 4.6 BASES (cont'd)."

    b. In the third and fourth paragraphs replace the abbreviation "yr" with the word "years."

18. Page 165, Specification 4.7

    In the Objective, delete the "," in the phrase "primary, and secondary."

19. Page 170, Specification 4.7.A.2.a.(10.)

    In the last sentence, delete the " * " and the referenced note which reads:

    "* In accordance with an exemption from 10 CFR 50 Appendix J, a Type A test need not be performed during the 1988 refueling outage."

20. Page 174, Specification 4.7.A.2

    a. In Specification 4.7.A.2.e.(5) replace the word "year" with the word "years."

    b. In Specification 4.7.A.2.f delete the " * " and the referenced note which reads:

       "* In accordance with an exemption from 10 CFR 50 Appendix J, a Type A, B, or C test is not required for: 1. The replacement of the HPCI turbine exhaust line block valve (23-HPI-11) during the 1988 outage; or 2. The repair of the Core Spray test return line weld 10-14-884A during the 1989 maintenance outage."

21. Page 187, Bases 3.7.A

    In the third paragraph, replace the parenthetical "(Section 5.2)" with the parenthetical (updated FSAR Section 5.2)."

22. Page 191, Bases 3.7.B and 3.7.C

    In the second paragraph, replace the phrase "be tested daily" with the phrase "be verified operable daily."

23. Page 217, Specification 3.9.B.3

    In the first paragraph replace the phrase "7-day" with the phrase "7 day."

24. Page 222b, Specification 3.9.F.2.c

    Replace the word "maintenance" with the word "alternate."

25. Pages 224 and 224a, Bases 3.9

    a. Add a "D." to indicate Bases Section 3.9.D and insert the text "Not Used."

    b. In Bases Section 3.9.E, replace the phrase "A maintenance power source" with the phrase "An alternate power source."

    c. Renumber Bases Sections "D." , "E." and "F." as "E." , "F." and "G.", respectively.

26. Page 226, Bases 4.9

    a. Add a "D." to indicate Bases Section 4.9.D and insert the text "Not Used."

    b. Renumber Bases Sections "D.", "E." and "F." to read "E." , "F." and "G." , respectively.

27. Page 244c, Specification 3.12.A.1.d.3

    In the title to Section 3.12.A , replace the word "Waster" with the word "Water."

28. Page 244g, Specification 4.12.F.1

    In the last sentence delete the " * " and the referenced note which reads:

    "* The current surveillance interval for visually inspecting fire

barrier penetration seals is extended until May 15, 1992. This is a onetime extension, effective only for this inspection interval. The next surveillance interval began September 27, 1991."

29. Page 244h, Bases 3.12.A and 4.12.A

    Delete the second sentence which reads:

    "Both pumps take suction from the plant insure."

30. Page 246a, Bases 5.5.B

    In the third sentence, replace the word "analysis" with the word "analyses" and the word "concludes" with the word "conclude."

31. Page 247a, Specification 6.2.2.2

    a. Replace the first sentence:

       "An SRO or SRO with a license limited to fuel handling shall directly supervise all Core Alternations"

       with the sentence:

       "An SRO or an SRO with a license limited to fuel handling shall directly supervise all Core Alterations."

    b. Delete the second sentence, which reads:

       "This person shall directly supervise all Core Alterations."

32. Page 248, Specifications 6.0

    a. In Specification 6.3.1, replace the word "Radioligical" with the word "Radiological."

    b. In Specification 6.4 replace the phrase "10 CFR 55, Appendix A" with the phrase "10 CFR 55.59."

33. Page 285, Specification 7.0

    a. In Reference 16, replace the phrase "10CFR50.54" with the phrase "10 CFR 50.54" and add a quotation mark in front of the word "Reactor."

b. In Reference 17, replace the phrase "10CFR50" with the phrase "10 CFR 50."

## II. PURPOSE OF THE PROPOSED CHANGES

This application makes miscellaneous administrative changes (e.g., correcting editorial errors, typographical errors and specification renumeration corrections) to the James A. FitzPatrick Technical Specifications. The proposed changes will clarify and improve the quality of the Technical Specifications. The purpose of each change identified in Section I (the numbers in Sections I and II correspond) is as follows:

1. The proposed change updates the table of contents to reflect Amendment 181 (References 1 and 2) which deleted Specifications 3.2.F and 4.2.F.

2. The proposed change corrects a typographical error making the title in the list consistent with the actual title.

3. The proposed change removes unnecessary specification outline numbers to be consistent with the pattern used everywhere else in the Technical Specifications.

4. The first proposed change modifies the outline numbering pattern in the Bases Section to be consistent with the associated LCO and Surveillance Requirements. The second proposed change adds Amendment number "14" to the page Amendment number listing. This Amendment number was inadvertently omitted when the page was revised as part of Amendment 49 (References 3 and 4).

5. The proposed changes correct two typographical errors in the ASME Boiler & Pressure Vessel Code and ANSI Code pressure transient allowance equations. The proposed changes replace the "-" symbols in two locations with the appropriate "=" symbols.

6. The proposed change modifies the outline numbering pattern in Bases Section 3.1 to be consistent with the associated specification outline numbering pattern.

7. The proposed change corrects a punctuation error by adding a period to the end of the sentence.

8. The proposed change corrects a typographical error by capitalizing the "v" in valves for internal consistency.

9. The proposed change corrects a typographical error by removing reference to Note "(8)" from Table 4.2-2. Note (8) clarifies that surveillance testing for reactor low water level, high drywell pressure and high radiation main steam line tunnel Instruments are not included on Table 4.2-1 but on Table 4.1-2. This note was inadvertently added to Table 4.2-2 as part of Amendment 160 (References 5 and 6).

10. The proposed change corrects a typographical error by adding the word "at" to the sentence. This word was inadvertently omitted as part of Amendment 134 (References 7 and 8). The change restores the sentence to its original form.

11. The proposed change relocates Specification 3.6.A.2 to line up with Specification 4.6.A.2. The change is made to clarify association between the LCO and the surveillance (the change relocates it next to Surveillance Requirement 4.6.A.2) and to be consistent with Technical Specification format between LCO and Surveillance Requirements.

12. The proposed changes make editorial corrections by replacing the abbreviation "hr" with the word "hour" or "hours" as applicable.

13. The proposed changes make editorial corrections by replacing the abbreviation "hr" with the word "hours."

14. The proposed change relocates Specification 3.6.G to line up with Specification 4.6.G. The change is made to clarify association between the LCO and the surveillance (the change relocates it next to Surveillance Requirement 4.6.G) and to be consistent with Technical Specification format between LCO and Surveillance Requirements.

15. The proposed change makes an editorial correction by replacing the abbreviation "hr" with the word "hours."

16. The proposed change makes an editorial correction by replacing the abbreviation "hr" with the word "hours."

17. The proposed change makes an editorial correction by replacing the abbreviation "yr" with the word "years." The change also revises the section heading to be consistent with other section headings.

18. The proposed change corrects an editorial error by removing an unnecessary comma from Specification 4.7.

19. The proposed change removes a past exemption from 10 CFR 50 Appendix J. The exemption, added by Amendment 125 (References 9 and 10), eliminated

a requirement to conduct Type A primary containment integrity leak rate test during the 1988 refueling outage. Since this exemption is no longer in effect, the exemption is removed.

20. The proposed change makes an editorial correction by revising the word "year" to reflect the correct plural form of "years." The proposed change also removes two past exemptions from 10 CFR 50 Appendix J. The exemptions, added by Amendments 125 and 140 (References 11 and 12) eliminated a requirement to conduct Type A, B, or C leak rate tests for two plant modifications. Since these exemptions are no longer in effect, these exemptions are removed.

21. The proposed change clarifies a reference by indicating that the current design Bases is contained in the James A. FitzPatrick updated Final Safety Analysis Report (FSAR).

22. This change makes the Bases Section consistent with a prior change to the Surveillance Requirement. Amendment 148 (References 13 and 14) replaced the word "demonstrate" with the word "verify" where necessary to eliminate redundant and unnecessary surveillance tests performed to satisfy overlapping requirements. Bases Section 3.7.B and 3.7.C should have been changed at that time along with the Amendment 148 changes. This proposed change will correct this omission.

23. The proposed change makes an editorial correction for consistency within the Technical Specifications.

24. The proposed change revises the name of the Low Pressure Coolant Injection (LPCI) "maintenance power source" to "alternate power source." The name change was made as part of a recent plant modification to the LPCI Motor Operated Valve (MOV) power source circuitry. The modification provided a control scheme enabling the plant operators, from the control room, to isolate the LPCI valve bus independent power supplies and connect a maintenance bypass (renamed alternate feed) from another safety related emergency Motor Control Center (MCC) in the same safety division to the valve bus. The modification gives operators full control over the power sources for the LPCI valve bus in the event the reactor building becomes restricted due to postulated post-accident radiation dose levels. The modification had no effect on the Technical Specifications except for this name change. Due to the nature of the change (i.e., renaming the power source) this change is considered administrative.

25. The proposed changes revises the outline numbering pattern of Bases Section 3.9 to be consistent with the associated LCO Specifications of 3.9 by adding a section and indicating that it is not being used. The proposed change also

renumbers other Bases Sections to be consistent with the associated LCO Section outline numbering pattern. The change which renames the "maintenance power source" to "alternate power source" is made to be consistent with the changes of item 24.

26. The proposed changes revise the outline numbering pattern of Bases Section 4.9 to be consistent with the associated Surveillance Requirements of Section 4.9 in the same manner as the revisions to Section 3.9 in item 25. The proposed changes add a section that is indicated as not being used and renumber the Bases Sections to be consistent with the Surveillance Section outline numbering pattern.

27. The proposed change corrects the spelling of the word "Water."

28. The proposed change removes a past one time extension to the fire barrier penetration seal visual inspection interval. The extension, added by Amendment 177 (References 15 and 16), ended on May 15, 1992. Since this extension is no longer in effect, the extension is removed.

29. The proposed change corrects a typographical error introduced by Amendment 176 (Reference 17 and 18). The error inadvertently duplicated part of a following sentence. This change removes the duplication.

30. The proposed changes make two editorial corrections by revising words to reflect the correct plural and singular forms.

31. The proposed changes make three editorial corrections. In the first sentence the word "an" is added and the spelling of the word "Alterations" is corrected. The second sentence is removed since it is duplicate to the first sentence.

32. The proposed change corrects the spelling of the word "Radiological" and revises a reference to reflect a change in the location of regulations from 10 CFR 55, Appendix A to 10 CFR 55.59, effective May 26, 1987 (Reference 19). The regulation change incorporated the licensed operator requalification requirements into 10 CFR 55.59 and subsequently deleted 10 CFR Part 55, Appendix A. This change corrects the reference to Title 10 Code of Federal Regulations.

33. The proposed changes make editorial corrections by revising reference to the Code of Federal Regulations, by adding proper spacing. The changes also add a missing quotation mark.

## III. SAFETY IMPLICATIONS OF THE PROPOSED CHANGES

The proposed changes to the James A. FitzPatrick Technical Specifications will not affect plant safety or operations. The proposed changes will correct editorial and typographical errors as well as remove past exceptions to Specifications. These changes will clarify and improve the quality of the Technical Specifications. The nature of each change assures that no safety implications are associated with these changes. The proposed changes involve no limiting conditions for operation, surveillance requirements, setpoint or safety limit changes, nor do they affect the environmental monitoring program. The proposed changes do not change any system or subsystem and will not alter the conclusions of either the updated FSAR or the SER.

## IV. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Operation of the FitzPatrick plant in accordance with the proposed Amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated.

   The intent of the proposed changes is to clarify and improve quality of the Technical Specifications. The proposed changes will correct editorial and typographical errors as well as remove past exceptions to Specifications. These changes will clarify and improve the quality of the Technical Specifications. The changes by their nature have no affect on previously evaluated accidents. There are no setpoint changes, safety limit changes, surveillance requirement changes or limiting conditions for operation changes. These changes have no affect on plant safety or operations.

2. create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed changes involve no plant modifications, changes to surveillance test methods or frequencies, changes to operating procedures or relaxation of any LCO. The proposed changes are administrative in nature and involve such changes as editorial corrections, typographical corrections and removal of past exceptions to Specifications. These proposed changes clarify and improve the quality of the Technical Specifications and by their nature cannot create the possibility of a new or different kind of accident.

3. involve a significant reduction in a margin of safety.

   The proposed changes are administrative in nature and will clarify and improve quality in the Technical Specifications. The proposed changes will correct editorial and typographical errors as well as remove past exceptions to Specifications. These changes, by their nature, can have no affect on the margin of safety. These changes do not change any setpoint or safety limit changes regarding isolation or alarms. The proposed changes do not affect the environmental monitoring program. These changes do not affect the plants safety systems.

## V. IMPLEMENTATION OF THE PROPOSED CHANGES

Implementation of the proposed changes will not adversely affect the ALARA or Fire Protection Programs at the FitzPatrick plant, nor will the changes affect the environment. This application for an amendment makes miscellaneous administrative changes and can have no affect on these programs or the environment.

## VI. CONCLUSION

The changes, as proposed, do not constitute an unreviewed safety question as defined in 10 CFR 50.59. That is, they:

1. will not change the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report;

2. will not increase the possibility of an accident or malfunction of a type different from any previously evaluated in the Safety Analysis Report; and

3. will not reduce the margin of safety as defined in the basis for any technical specification;

The changes involve no significant hazards consideration, as defined in 10 CFR 50.92.

## VII. REFERENCES

References relied upon to prepare the Technical Specification change request:

1. NYPA Letter, R.E. Beedle to B.C. McCabe dated May 30, 1992 (JPN-92-042).

Submittal for Amendment 181 to the Technical Specifications.

2. NRC Letter, B.C. McCabe to R.E. Beedle dated May 14, 1992 (JAF-92-139). Transmits Amendment 181 to the Technical Specifications.

3. NYPA Letter, G.T. Berry to T.A. Ippolito dated March 4, 1980 (JPN-80-015). Submittal for Amendment 49 to the Technical Specifications.

4. NRC Letter, T.A. Ippolito to G.T. Berry dated July 11, 1980 (JAF-80-142). Transmits Amendment 49 to the Technical Specifications.

5. NYPA Letter, J.C. Brons to D.E. LaBarge dated January 1, 1990 (JPN-90-003). Submittal for Amendment 160 to the Technical Specifications.

6. NRC Letter, D.E. LaBarge to J.C. Brons dated May 18, 1990 (JAF-90-162). Transmits Amendment 160 to the Technical Specifications.

7. NYPA Letter, J.C. Brons to D.E. LaBarge dated May 24, 1989 (JPN-89-030). Submittal for Amendment 134 to the Technical Specifications.

8. NRC Letter, D.E. LaBarge to J.C. Brons dated January 24, 1990 (JAF-90-028). Transmits Technical Specification Replacement Pages (Re-issues page 123 as of Amendment 134).

9. NYPA Letter, J.C. Brons to D.E. LaBarge dated November 9, 1988 (JPN-88-060). Submittal for Amendment 125 to the Technical Specifications.

10. NRC Letter, D.E. LaBarge to J.C. Brons dated February 17, 1989 (JAF-89-066). Transmits Amendment 125 to the Technical Specifications.

11. NYPA Letter, J.C. Brons to D.E. LaBarge dated September 28, 1989 (JPN-89-062). Submittal for Amendment 140 to the Technical Specifications.

12. NRC Letter, D.E. LaBarge to J.C. Brons dated October 4, 1989 (JAF-89-347). Transmits Amendment 140 to the Technical Specifications.

13. NYPA Letter, J.C. Brons to D.E. LaBarge dated May 31, 1989 (JPN-89-034). Submittal for Amendment 148 to the Technical Specifications.

14. NRC Letter, D.E. LaBarge to J.C. Brons dated December 26, 1989 (JAF-90-002). Transmits Amendment 148 to the Technical Specifications.

15. NYPA Letter, R.E. Beedle to B.C. McCabe dated December 19, 1991 (JPN-91-069). Submittal for Amendment 177 to the Technical Specifications.

16. NRC Letter, B.C. McCabe to R.E. Beedle dated February 10, 1992 (JAF-92-036). Transmits Amendment 177 to the Technical Specifications.

17. NYPA Letter, R.E. Beedle to B.C. McCabe dated December 19, 1991 (JPN-91-068). Submittal for Amendment 176 to the Technical Specifications.

18. NRC Letter, B.C. McCabe to R.E. Beedle dated January 16, 1992 (JAF-92-015). Transmits Amendment 176 to the Technical Specifications.

19. Federal Register dated March 25,1987 (52 FR 9460).

References reviewed but not specifically referenced:

1. James A. FitzPatrick Nuclear Power Plant Updated Final Safety Analysis Report Section 5.2, through Revision 5, dated January 1992.

2. James A. FitzPatrick Nuclear Power Plant Safety Evaluation Report (SER), dated November 20, 1972, and Supplements.

Attachment III to JPN-93-012

PROPOSED TECHNICAL SPECIFICATION CHANGES
MISCELLANEOUS ADMINISTRATIVE CHANGES
MARKUP OF TECHNICAL SPECIFICATION PAGES

(JPTS-92-001)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

# TECHNICAL SPECIFICATIONS

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## LIST OF TABLES

1.1 (cont'd)

B. Core Thermal Power Limit (Reactor Pressure $\leq$785 psig)

When the reactor pressure is $\leq$785 psig or core flow is less than or equal to 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

C. Power Transient

To ensure that the Safety Limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

2.1 (cont'd)

A.

1.

b. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

APRM - The APRM flux scram setting shall be $\leq$15 percent of rated neutron flux with the Reactor Mode Switch in Startup/Hot Standby or Refuel.

c. APRM Flux Scram Trip Settings (Run Mode)

(1) Flow Referenced Neutron Flux Scram Trip Setting

When the Mode Switch is in the RUN position, the APRM flow referenced flux scram trip setting shall be less than or equal to the limit specified in Table 3.1-1. This setting shall be adjusted during single loop operation when required by Specification 3.5.J.

For no combination of recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 117% of rated thermal power.

B. Not Used

2.1 BASES (Cont'd)

C. References

1. (Deleted)

2. "General Electric Standard Application for Reactor Fuel", NEDE 24011-P-A (Approved revision number applicable at time that reload fuel analyses are performed).

3. (Deleted)

4. FitzPatrick Nuclear Power Plant Single-Loop Operation, NEDO-24281, August, 1980.

14,

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1.2 and 2.2 BASES

The reactor coolant pressure boundary integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this boundary be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1,325 psig as measured by the vessel steam space pressure indicator is equivalent to 1,375 psig at the lowest elevation of the Reactor Coolant System. The 1,375 psig value is derived from the design pressures of the reactor pressure vessel and reactor coolant system piping. The respective design pressures are 1250 psig at 575°F for the reactor vessel, 1148 psig at 568°F for the recirculation suction piping and 1274 psig at 575° for the discharge piping. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: 1965 ASME Boiler and Pressure Vessel Code, Section III for pressure vessel and 1969 ANSI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10 percent over design pressure (110% x 1,250 - 1,375 psig) and the

=

ANSI Code permits pressure transients up to 20 percent over the design pressure (120% x 1,150 - 1,380 psig). The safety limit pressure of 1,375 psig is referenced to the lowest elevation of the Reactor Coolant System.

=

The current reload analysis shows that the main steam isolation valve closure transient, with flux scram, is the most severe event resulting directly in a reactor coolant system pressure increase. The reactor vessel pressure code limit of 1,375 psig, given in FSAR Section 4.2, is above the peak pressure produced by the event above. Thus, the pressure safety limit (1,375 psig) is well above the peak pressure that can result from reasonably expected overpressure transients. (See current reload analysis for the curve produced by this analysis.) Reactor pressure is continuously indicated in the control room during operation.

A safety limit is applied to the Residual Heat Removal System (RHRS) when it is operating in the shutdown cooling mode. When operating in the shutdown cooling mode, the RHRS is included in the reactor coolant system.

The numerical distribution of safety/relief valve setpoints shown in 2.2.1.B (2 @ 1090 psi, 2 @ 1105 psi, 7 @ 1140 psi) is justified by analyses described in the General Electric report NEDO-24129-1, Supplement 1, and assures that the structural acceptance criteria set forth in the Mark I Containment Short Term Program are satisfied.

3.1 BASES

A. The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.
2. Preserve the integrity of the Reactor Coolant System.
3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The Reactor Protection System is of the dual channel type (Reference subsection 7.2 FSAR). The System is made up of two independent trip systems, each having two subchannels of tripping devices. Each subchannel has an input from at least one instrument channel which monitors a critical parameter.

The outputs of the subchannels are combined in a 1 out of 2 logic; i.e., an input signal on either one or both of the subchannels will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the intent of IEEE-279 (1971) for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the average power range monitor (APRM) channel the intermediate range monitor (IRM) channels, the scram discharge volume, the main steam isolation valve closure and the turbine stop valve closure, each subchannel has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the affected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved. 1 a

Three APRM instrument channels are provided for each protection trip system. APRM's A and E operate contacts in one subchannel and APRM's C and E operate contacts in the other

3.2 (cont'd)

E. Drywell Leak Detection

The limiting conditions of operation for the instrumentation that monitors drywell leak detection are given in Table 3.2-5.

F. (Deleted)

G. Recirculation Pump Trip

The limiting conditions for operation for the instrumentation that trip(s) the recirculation pumps as a means of limiting the consequences of a failure to scram during an anticipated transient are given in Table 3.2-7.

H. Accident Monitoring Instrumentation

The limiting conditions for operation of the instrumentation that provides accident monitoring are given in Table 3.2-8.

I. 4kv Emergency Bus Undervoltage Trip

The limiting conditions for operation for the instrumentation that prevents damage to electrical equipment or circuits as a result of either a degraded or loss-of-voltage condition on the emergency electrical buses are given in Table 3.2-2.

4.2 (cont'd)

E. Drywell Leak Detection

Instrumentation shall be calibrated and checked as indicated in Table 4.2-5.

F. (Deleted)

G. Recirculation Pump Trip

Instrumentation shall be functionally tested and calibrated as indicated in Table 4.2-7.

System logic shall be functionally tested as indicated in Table 4.2-7.

H. Accident Monitoring Instrumentation

Instrumentation shall be demonstrated operable by performance of a channel check and channel calibration as indicated in Table 4.2-8.

TABLE 4.2-1

**MINIMUM TEST AND CALIBRATION FREQUENCY FOR PCIS**

| Instrument Channel (8) | Instrument Functional Test | Calibration Frequency | Instrument Check (4) |
|---|---|---|---|
| 1) Reactor High Pressure (Shutdown Cooling Permissive) | (1) | Once/3 months | None |
| 2) Reactor Low-Low-Low Water Level | (1)(5) | (15) | Once/day |
| 3) Main Steam High Temp. | (1)(5) | (15) | Once/day |
| 4) Main Steam High Flow | (1)(5) | (15) | Once/day |
| 5) Main Steam Low Pressure | (1)(5) | (15) | Once/day |
| 6) Reactor Water Cleanup High Temp. | (1) | Once/3 months | None |
| 7) Condenser Low Vacuum | (1)(5) | (15) | Once/day |

| Logic System Functional Test (7) (9) | Frequency |
|---|---|
| 1) Main Steam Line Isolation valves Valves<br>Main Steam Line Drain Valves<br>Reactor Water Sample Valves | Once/6 months |
| 2) RHR - Isolation Valve Control<br>Shutdown Cooling Valves | Once/6 months |
| 3) Reactor Water Cleanup Isolation | Once/6 months |
| 4) Drywell Isolation Valves<br>TIP Withdrawal<br>Atmospheric Control Valves | Once/6 months |
| 5) Standby Gas Treatment System<br>Reactor Building Isolation | Once/6 months |

NOTE: See notes following Table 4.2-5.

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TABLE 4.2-2

## MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS

| | Instrument Channel (8) | Instrument Functional Test | Calibration Frequency | Instrument Check (4) |
|---|---|---|---|---|
| 1) | Reactor Water Level | (1)(5) | (15) | Once/day |
| 2a) | Drywell Pressure (non-ATTS) | (1) | Once/3 months | None |
| 2b) | Drywell Pressure (ATTS) | (1)(5) | (15) | Once/day |
| 3a) | Reactor Pressure (non-ATTS) | (1) | Once/3 months | None |
| 3b) | Reactor Pressure (ATTS) | (1)(5) | (15) | Once/day |
| 4) | Auto Sequencing Timers | None | Once/operating cycle | None |
| 5) | ADS - LPCI or CS Pump Disch. | (1) | Once/3 months | None |
| 6) | Trip System Bus Power Monitors | (1) | None | None |
| 8) | Core Spray Sparger d/p | (1) | Once/3 months | Once/day |
| 9) | Steam Line High Flow (HPCI & RCIC) | (1)(5) | (15) | Once/day |
| 10) | Steam Line/Area High Temp. (HPCI & RCIC) | (1)(5) | (15) | Once/day |
| 12) | HPCI & RCIC Steam Line Low Pressure | (1)(5) | (15) | Once/day |
| 13) | HPCI & RCIC Suction Source Levels | (1) | Once/3 months | None |
| 14) | 4kV Emergency Bus Under-Voltage (Loss-of-Voltage, Degraded Voltage LOCA and non-LOCA) Ralays and Timers. | Once/operating cycle | Once/operating cycle | None |
| 15) | HPCI & RCIC Exhaust Diaphragm Pressure High | (1) | Once/3 months | None |
| 17) | LPCI/Cross Connect Valve Position | Once/operating cycle | None | None |

NOTE: See notes following Table 4.2-5.

3.5 (cont'd)

condition, that pump shall be considered inoperable for purposes of satisfying Specifications 3.5.A, 3.5.C, and 3.5.E.

H. Average Planar Linear Heat Generation Rate (APLHGR)

at

During power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure shall be within limits based on applicable APLHGR limit values which have been approved for the respective fuel and lattice types. These values are specified in the Core Operating Limits Report. If anytime during reactor power operation greater than 25% of rated power it is determined that the limiting value for APLHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, an orderly reactor power reduction shall be commenced immediately. The reactor power shall be reduced to less than 25% of rated power within the next four hours, or until the APLHGR is returned to within the prescribed limits.

4.5 (cont'd)

2. Following any period where the LPCI subsystems or core spray subsystems have not been maintained in a filled condition; the discharge piping of the affected subsystem shall be vented from the high point of the system and water flow observed.

3. Whenever the HPCI or RCIC System is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI or RCIC shall be vented from the high point of the system, and water flow observed on a monthly basis.

4. The level switches located on the Core Spray and RHR System discharge piping high points which monitor these lines to insure they are full shall be functionally tested each month.

H. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$25% rated thermal power.

3.6 LIMITING CONDITIONS FOR OPERATION

3.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the operating status of the Reactor Coolant System.

Objective:

To assure the integrity and safe operation of the Reactor Coolant System.

Specification:

A. Pressurization and Thermal Limits

1. Reactor Vessel Head Stud Tensioning

The reactor vessel head bolting studs shall not be under tension unless the temperatures of the reactor vessel flange and the reactor head flange are at least 90°F.

2. In-Service Hydrostatic and Leak Tests

During in-service hydrostatic or leak testing the Reactor Coolant System pressure and temperature shall be on or to the right of curve A shown in Figure 3.6-1 Part 1, 2, or 3 and the maximum temperature change during any one hour period shall be:

4.6 SURVEILLANCE REQUIREMENTS

4.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the periodic examination and testing requirements for the Reactor Coolant System.

Objective:

To determine the condition of the Reactor Coolant System and the operation of the safety devices related to it.

Specification:

A. Pressurization and Thermal Limits

1. Reactor Vessel Head Stud Tensioning

When in the cold condition, the reactor vessel head flange and the reactor vessel flange temperatures shall be recorded:

a. Every 12 hours when the reactor vessel head flange is $\leq 120°F$ and the studs are tensioned.

b. Every 30 minutes when the reactor vessel head flange is $\leq 100°F$ and the studs are tensioned.

c. Within 30 minutes prior to and every 30 minutes during tensioning of reactor vessel head bolting studs.

2. In-Service Hydrostatic and Leak Tests

During hydrostatic and leak testing the Reactor Coolant System pressure and temperature shall be recorded every 30 minutes until two consecutive temperature readings are within 5°F of each other.

Amendment No. ~~14~~, ~~113~~, ~~158~~

3.6 (cont'd)

B. Deleted

hours.

C. Coolant Chemistry

1. The reactor coolant system radioactivity concentration in water shall not exceed the equilibrium value of 3.1 μCi/gm of dose equivalent I-131. This limit may be exceeded, following a power transient, for a maximum of 48 hr. During this iodine activity transient the iodine concentrations shall not exceed the equilibrium limits by more than a factor of 10 whenever the main steamline isolation valves are open. The reactor shall not be operated more than 5 percent of its annual power operation under this exception to the equilibrium limits. If the iodine concentration exceeds the equilibrium limit by more than a factor of 10, the reactor shall be placed in a cold condition within 24 hr.

hours.

4.6 (cont'd)

7. Reactor Vessel Flux Monitoring

The reactor vessel Flux Monitoring Surveillance Program complies with the intent of the May, 1983 revision to 10 CFR 50, Appendices G and H. The next flux monitoring surveillance capsule shall be removed after 15 effective full power years (EFPYs) and the test procedures and reporting requirements shall meet the requirements of ASTM E 185-82.

B. Deleted

hours

C. Coolant Chemistry

1. a. A sample of reactor coolant shall be taken at least every 96 hr and analyzed for gross gamma activity.

b. Isotopic analysis of a sample of reactor coolant shall be made at least once/month.

c. A sample of reactor coolant shall be taken prior to startup and at 4 hr intervals during startup and analyzed for gross gamma activity.

hour

d. During plant steady state operation and following an offgas activity increase (at the Steam Jet Air Ejectors) of 10,000 μCi/sec within a 48 hr period or a power level change of ≥20 percent of full rated power/hr reactor coolant samples shall be taken and analyzed for gross gamma activity. At least three samples will be taken at 4 hr intervals. These sampling requirements may be omitted whenever the equilibrium I-131 concentration in the reactor coolant is less than 0.007 μCi/ml.

hour

2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 lb/hr except as specified in 3.6.C.3:

| | |
|---|---|
| Conductivity | 2 $\mu$ mho/cm |
| Chloride ion | 0.1 ppm |

3. For reactor startups, the maximum value for conductivity shall not exceed 10 $\mu$ mho/cm and the maximum value for chloride ion concentration shall not exceed 0.1 ppm, for the first 24 hr after placing the reactor in the power operating condition. During reactor shutdowns, specification 3.6.C.4 will apply.

hours

4.6 (cont'd)

e. If the gross activity counts made in accordance with a, c, and d above indicate a total iodine concentration in excess of 0.007 $\mu$Ci/ml, a quantative determination shall be made for I-131 and I-133.

2. During startups and at steaming rates below 100,000 lb/hr, and when the conductivity of the reactor coolant exceeds 2 $\mu$ mhos/cm, a sample of reactor coolant shall be taken every 4 hr and analyzed for conductivity and chloride content.

3. a. With steaming rates greater than or equal to 100,000 lb /hr, a reactor coolant sample shall be taken at least every 96 hr and whenever the continuous conductivity monitors indicate abnormal conductivity (other than short-term spikes), and analyzed for conductivity and chloride ion content.

hours

b. When the continuous conductivity monitor is inoperable, a reactor coolant sample shall be taken at least daily and analyzed for conductivity and chloride ion content.

3.6 (cont'd)

F. Structural Integrity

The structural integrity of the Reactor Coolant System shall be maintained at the level required by the original acceptance standards throughout the life of the Plant.

G. Jet Pumps

Whenever the reactor is in the startup/hot standby or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, the reactor shall be placed in a cold condition within 24 hours.

4.6 (cont'd)

F. Structural Integrity

1. Nondestructive inspections shall be performed on the ASME Boiler and Pressure Vessel Code Class 1, 2 and 3 components and supports in accordance with the requirements of the weld and support inservice inspection program. This inservice inspection program is based on an NRC approved edition of, and addenda to, Section XI of the ASME Boiler and Pressure Vessel Code which is in effect 12 months or less prior to the beginning of the inspection interval.

2. An augmented inservice inspection program is required for those high stressed circumferential piping joints in the main steam and feedwater lines larger than 4 inches in diameter, where no restraint against pipe whip is provided. The augmented in-service inspection program shall consist of 100 percent inspection of these welds per inspection interval.

3. An inservice inspection Program for piping identified in the NRC Generic Letter 88-01 shall be implemented in accordance with NRC staff positions on schedules, methods, personnel, and sample expansion included in this Generic Letter, or in acordance with alternate measures approved by the NRC staff.

G. Jet Pumps

Whenever there is recirculation flow with the reactor in the startup/hot standby or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

3.6 and 4.6 BASES (cont'd)

B. Deleted

C. Coolant Chemistry

A radioactivity concentration limit of 20 $\mu$Ci/ml total iodine can be reached if the gaseous effluents are near the limit as set forth in Radiological Effluent Technical Specification Section 3.2.a if there is a failure or a prolonged shutdown of the cleanup demineralizer.

In the event of a steam line rupture outside the drywell, with this coolant activity level, the resultant radiological dose at the site boundary would be 33 rem to the thyroid, under adverse meteorological conditions assuming no more than 3.1 $\mu$Ci/gm of dose equivalent I-131. The reactor water sample will be used to assure that the limit of Specification 3.6.C is not exceeded. The total radioactive iodine activity would not be expected to change rapidly over a period of 96 hr. In addition, the trend of the stack offgas release rate, which is continuously monitored, is a good indicator of the trend of the iodine activity in the reactor coolant. Also during reactor startups and large power changes which could affect iodine levels, samples of reactor coolant shall be analyzed to insure iodine concentrations are below allowable levels. Analysis is required whenever the I-131 concentration is within a factor of 100 of its allowable equilibrium value. The necessity for continued sampling following power and offgas transients will be reviewed within 2 years of initial plant startup.

hours.

The surveillance requirements 4.6.C.1 may be satisfied by a continuous monitoring system capable of determining the total iodine concentration in the coolant on a real time basis, and annunciating at appropriate concentration levels such that sampling for isotopic analysis can be initiated. The design details of such a system must be submitted for evaluation and accepted by the Commission prior to its implementation and incorporation in these Technical Specifications.

Since the concentration of radioactivity in the reactor coolant is not continuously measured, coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the offgas system and on the main steam lines.

Materials in the Reactor Coolant System are primarily 304 stainless steel and Zircaloy fuel cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on chloride concentration and conductivity. The most important limit is that placed on chloride concentration to prevent stress corrosion cracking of the stainless steel. The attached graph, Fig. 4.6-1, illustrates the results of tests on stressed 304 stainless steel specimens. Failures occurred at concentrations above the curve; no failures occurred at concentrations below the curve. According to the data, allowable chloride concentrations could be set several orders of magnitude above the established limit, at the oxygen concentration (0.2-0.3 ppm) experienced during power operation. Zircaloy does not exhibit similar stress corrosion failures.

However, there are various conditions under which the dissolved oxygen content of the reactor coolant water could be higher than 0.2-0.3 ppm, such as refueling, reactor startup, and hot standby. During these periods with steaming rates less

than 100,000 lb/hr, a more restrictive limit of 0.1 ppm has been established to assure the chloride-oxygen combinations of Fig. 4.6-1 are not exceeded. At steaming rates of at least 100,000 lb/hr, boiling occurs causing deaeration of the reactor water, thus maintaining oxygen concentration at low levels.

When conductivity is in its proper normal range, pH and chloride and other impurities affecting conductivity must also be within their normal ranges. When and if conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This is not necessarily the case. Conductivity could be high due to the presence of a neutral salt; e.g., $Na_2 SO_4$ , which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are, in fact, high due to purposeful addition of additives. In the case of BWR's, however, where no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the condition include operation of the Reactor Cleanup System, reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the Reactor Water Cleanup System to reestablish the purity of the reactor coolant. During startup periods, which are in the category of less than 100,000 lb/hr, conductivity may exceed 2 µmho/cm because of the initial evolution of gases and the initial evolution of gases and the initial addition of dissolved metals. During this period of time, when the conductivity exceeds 2 µmho/cm (other than short-term spikes), samples will be taken to assure the chloride concentration is less than 0.1 ppm.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hr will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses of the reactor coolant required by Specification 4.6.C.1 may be performed by a gamma scan.

hours

D. Coolant Leakage

Allowable leakage rates of coolant from the Reactor Coolant System have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to make up Reactor Coolant System leakage in the event of loss of off-site a-c power. The normally expected background leakage due to equipment design and the detection capability for determining system

leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC-sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.D on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm as specified in 3.6.D, the experimental and analytical data suggest a reasonable margin of safety such that leakage of this magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time, the Plant should be shut down to allow further investigation and corrective action.

The capacity of the drywell sump pumps is 100 gpm, and the capacity of the drywell equipment drain tank pumps is also 100 gpm. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

The performance of the Reactor Coolant Leakage Detection System will be evaluated during the first 5 yr years of plant operation, and the conclusions of this evaluation will be reported to the NRC.

It is estimated that the main steam line tunnel leakage detectors are capable of detecting a leak on the order of 3,500 lb/hr. The system performance will be evaluated during the first 5 yr years of plant operation, and the conclusions of the evaluation will be reported to the NRC.

The reactor coolant leakage detection systems consist of the drywell sump monitoring system and the drywell continuous atmosphere monitoring system. The drywell continuous atmosphere monitoring system utilizes a three-channel monitor to provide information on particulate, iodine and noble gas activities in the drywell atmosphere. Two independent and redundant systems are provided to perform this function. This system supplements the drywell sump monitoring system in detecting abnormal leakage that could occur from the reactor coolant system. In the event that the drywell continuous atmosphere monitoring system is inoperable, grab sample will be taken on a periodic basis to monitor drywell activity.

## 3.7 LIMITING CONDITIONS FOR OPERATION

## 3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. The volume and temperature of the water in the torus shall be maintained within the following limits whenever the reactor is critical or whenever the reactor coolant temperature is greater than 212°F and irradiated fuel is in the reactor vessel:

   a. Maximum vent submergence level of 53 inches.

   b. Minimum vent submergence level of 51.5 inches.

   The torus water level may be outside the above limits for a maximum of four (4) hours during required operability testing of HPCI, RCIC, RHR, CS, and the Drywell-Torus Vacuum System.

   c. Maximum water temperature

      (1) During normal power operation maximum water temperature shall be 95°F.

## 4.7 SURVEILLANCE REQUIREMENTS

## 4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. The torus water level and temperature shall be monitored as specified in Table 4.2-8. The accessible interior surfaces of the drywell and above the water line of the torus shall be inspected at each refueling outage for evidence of deterioration. Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the torus shall be conducted before resuming power operation.

4.7 (cont'd)

Type A test shall be performed at each plant shutdown for refueling or approximately every 18 months, whichever occurs first, until two consecutive Type A tests meet the acceptance criteria.*

b. Type B tests (Local leak rate testing of containment penetrations)

(1.) All preoperational and periodic Type B tests shall be performed by local pneumatic pressurization of the containment penetrations, either individually or in groups, at a pressure not less than Pa, and the gas flow to maintain Pa shall be measured.

(2.) Acceptance criteria

The combined leakage rate of all penetrations and valves subject to Type B and C tests shall be less than 0.60 La, with the exception of the valves sealed with fluid from a seal system.

* In accordance with an exemption from 10 CFR 50 Appendix J, a Type A test need not be performed during the 1988 refueling outage.

4.7 (cont'd)

(5) Type C test.

Type C tests shall be performed during each reactor shutdown for refueling but in no case at intervals greater than two ~~year.~~ years.

(6) Other leak rate tests specified in Section 4.7d shall be performed during each reactor shutdown for refueling but in no case at intervals greater than two years.

f. Containment modification*

Any major modification, replacement of a component which is part of the primary reactor containment boundary, or resealing a seal-welded door, performed after the preoperational leakage rate test shall be followed by either a Type A, Type B, or Type C test, as applicable, for the area affected by the modification. The measured leakage from this test shall be included in the test report. The acceptance criteria as appropriate, shall be met. Minor modifications, replacements, or resealing of seal-welded doors, performed directly prior to the conduct of a scheduled Type A test do not require a separate test.

* In accordance with an exemption from 10 CFR 50 Appendix J, a Type A, B, or C test is not required for: 1. The replacement of the HPCI turbine exhaust line block valve (23-HPI-11) during the 1988 outage; or 2. The repair of the Core Spray test return line weld 10-14-884A during the 1989 maintenance outage.

## 3.7 BASES

### A. Primary Containment

The integrity of the primary containment and operation of the Emergency Core Cooling Systems in combination limit the offsite doses to values less than those specified in 10 CFR 100 in the event of a break in the Reactor Coolant System piping. Thus, containment integrity is required whenever the potential for violation of the Reactor Coolant System integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception to the requirement to maintain primary containment integrity is allowed during core loading and during low power physics testing when ready access to the reactor vessel is required. There will be no pressure on the system at this time, which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period, however, restrictive operating procedures and operation of the RWM in accordance with Specification 3.3.B.3 minimize the probability of an accident occurring. Procedures in conjunction with the Rod Worth Minimizer Technical Specifications limit individual control worth such that the drop of any in-sequence control rod would not result in a peak fuel enthalpy greater than 280 calories/gm. In the unlikely event that an excursion did occur, the reactor building and Standby Gas Treatment System, which shall be operational during this time, offers a sufficient barrier to keep offsite doses well within 10 CFR 100.

The pressure suppression pool water provides the heat sink for the Reactor Coolant System energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1,020 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 56 psig, the suppression chamber design pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber (Section 5.2).

updated FSAR

3.7 BASES (cont'd)

complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

The Standby Gas Treatment System is designed to filter and exhaust the reactor building atmosphere to the main stack during secondary containment isolation conditions with a minimum release of radioactive materials from the reactor building to the environs. Both standby gas treatment fans are designed to automatically start upon containment isolation; however, only one fan is required to maintain the reactor building pressure at approximately a negative 1/4 in. water gage pressure; all leakage should be in-leakage. Each of the two fans has 100 percent capacity. If one Standby Gas Treatment System circuit is inoperable, the other circuit must be tested daily. This substantiates the availability of the operable circuit and results in no added risk; thus, reactor operation or refueling operation can continue. If neither circuit is operable, the Plant is brought to a condition where the system is not required.

While only a small amount of particulates is released form the Pressure Suppression Chamber System as a result of the loss-of-coolant accident, high-efficiency particulate filters are specified to minimize potential particulate release to the environment and to prevent clogging of the iodine filter. The high-efficiency filters have an efficiency greater than 99 percent for particulate matter larger than 0.3 micron. The minimum iodine removal efficiency is 99 percent. Filter banks will be replaced whenever significant changes in filter efficiency occur. Tests (11) of impregnated charcoal identical to that used in the filters indicate that shelf life up to 5 yr leads to only minor decreases in methyl iodine removal efficiency.

be verified operable daily.

The 99 percent efficiency of the charcoal and particulate filters is sufficient to prevent exceeding 10CFR100 guidelines for the accidents analyzed. The analysis of the loss-of-coolant accident assumed a charcoal filter efficiency of 90 percent, a particulate filter efficiency of 90 percent, and TID 14844 fission product source term. Hence, requiring 99 percent efficiency for both the charcoal and particulate filters provides adequate margin. A heater maintains relative humidity below 70 percent in order to assure the efficient removal of methyl iodine on the impregnated charcoal filters.

The operability of the Standby Gas Treatment System (SGTS) must be assured if a design basis loss of coolant accident (LOCA) occurs while the containment is being purged or vented through the SGTS. Flow from containment to the SGTS is via 6 inch Valve Number 27MOV-121. Since the maximum flow through the 6 inch line following a design basis LOCA is within the design capabilities of the SGTS, use of the 6 inch line assures the operability of the SGTS.

D. Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space

3.9 (cont'd)

3. From and after the time that one of the Emergency Diesel Generator Systems is made or found to be inoperable, continued reactor operation is permissible for a period not to exceed 7 days provided that the two incoming power sources are available and that the remaining Diesel Generator System is operable. At the end of the 7-day period, the reactor shall be placed in a cold condition within 24 hours, unless the affected diesel generator system is made operable sooner.

7 day

4. When both Emergency Diesel Generator Systems are made or found to be inoperable, a reactor shutdown shall be initiated within two hours and the reactor placed in a cold condition within 24 hours after initiation of shutdown.

4.9 (cont'd)

3. The emergency diesel generator system instrumentation shall be checked during the monthly generator test.

4. Once each operating cycle, the conditions under which the Emergency Diesel Generator System is required will be simulated to demonstrate that the pair of diesel generators will start, accelerate, force parallel, and accept the emergency loads in the prescribed sequence.

5. Once within one hour and at least once per twenty-four hours thereafter while the reactor is being operated in accordance with Specifications 3.9.B.1, 3.9.B.2, or 3.9.B.3 the availability of the operable Emergency Diesel Generators shall be demonstrated by manual starting and force paralleling where applicable.

F. LPCI MOV Independent Power Supplies

1. Reactor shall not be made critical unless both independent power supplies, including the batteries, inverters and chargers and their associated buses (MCC-155 and MCC-165) are in service, except as specified below.

2. During power operation, if one independent power supply becomes unavailable, repairs shall be made immediately and continued reactor operation is permissible for a period not to exceed 7 days unless the unavailable train is made operable sooner. From and after the date one of the independent power supplies is made or found to be inoperable for any reason, the following would apply:

   a. The other independent power supply including its charger, inverter, battery and associated bus is operable.

   b. Pilot cell voltage, specific gravity and temperature and overall battery voltage are measured immediately and weekly thereafter for the operable independent power supply battery.

   c. The inoperable independent power supply shall be isolated from its associated LPCI MOV bus, and this bus will be manually switched to its ~~maintenance~~ alternate power source.

3.9 BASES (cont'd)

C. Diesel Fuel

Minimum on-site fuel oil requirements are based on operation of the emergency diesel generator systems at rated load for 7 days.

Additional diesel fuel can be delivered to the site within 48 hours.

If one of the Emergency Diesel Generator Systems is not operable, the plant shall be permitted to run for 7 days provided both sources of reserve power are operational. This is based on the following:

1. The operable Emergency Diesel Generator System is capable of carrying sufficient engineered safeguards and emergency core cooling system equipment to cover all loss-of-coolant accidents.
2. The reserve (offsite) power is highly reliable.

D. Not Used

~~D.~~ E. Battery System

125 v DC power is supplied from two plant batteries each sized to supply the required equipment at design power following a loss-of-coolant accident with a concurrent loss of normal and reserve power. Each battery is provided with a charger sized to maintain the battery in a fully charged state while supplying normal operating loads.

~~E.~~ F. LPCI MOV Independent Power Supplies

There are two LPCI MOV Independent Power Supplies each consisting of a charger, rectifier, inverter and battery. Each independent power supply charger-rectifier is normally fed from the emergency A-C power supply system to maintain the battery in a fully charged state. In the event of a LOCA each independent power supply is automatically isolated from the Emergency A-C power system. The battery and inverter have sufficient capacity to power the MOV's essential to the operation of the LPCI System. ~~A maintenance~~ An alternate power source is provided for each LPCI MOV bus whereby in the event its independent power supply is out of service, the LPCI MOV bus may be energized directly from the Emergency A-C Power System.

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3.9 BASES (cont'd)

F. G. Reactor Protection System Power Supplies

Each of two RPS divisions may be supplied power from it's respective RPS MG set or from an alternate source which derives power from the same electrical division. The MG sets and alternate sources for both divisions are provided with redundant, seismic qualified, class 1E electrical protection assemblies between the power source and the RPS bus. Any abnormal output type failure in either of the MG sets or alternate sources (if in service) would result in a trip of one or both of the electrical protection assemblies producing a half scram on that RPS division and retaining full scram capability in the other RPS division.

Limiting operating conditions in Section 3.9.G provide a high degree of assurance that RPS buses are protected as described above.

D. Not Used



4.9 BASES (cont'd)

E.

D. Battery System

Measurements and electrical tests are conducted at specified intervals to provide indication of cell condition and to determine the discharge capability of the batteries. Performance and service tests are conducted in accordance with the recommendations of IEEE 450-1987.

F.

E. LPCI MOV Independent Power Supply

Measurement and electrical tests are conducted at specified intervals to provide indication of cell condition, to determine the discharge capability of the battery. Performance and service tests are conducted in accordance with the recommendations of IEEE 450-1987.

G.

F. Reactor Protection Power Supplies

Functional tests of the electrical protection assemblies are conducted once each six (6) months utilizing a built-in test device and once per operating cycle by performing an instrument calibration which verifies operation within the limits of Section 4.9.G.

A. High Pressure Waster Fire Protection System (Cont'd)

3. If 1. above cannot be fulfilled, place the reactor in Hot Standby within six (6) hours and in Cold Shutdown within the following thirty (30) hours.

Water

A. High Pressure Water Fire Protection System (Cont'd)

| | Item | Frequency |
|---|---|---|
| h. | Fire pump diesel engine by verifying the fuel storage tank contains at least 172 gallons of fuel. | Once/Month |
| i. | Diesel fuel from each tank obtained in accordance with ASTM-D270-65 is within the acceptable limits for quality as per the following: | Once/Quarter |
| | Flash Point - °F | 125°F min. |
| | Pour Point - °F | 10°F max. |
| | Water & Sediment | 0.05% max. |
| | Ash | 0.01% max. |
| | Distillation 90% Point | 540 min. |
| | Viscosity (SSU) @ 100°F | 40 max. |
| | Sulfur | 1% max. |
| | Copper Strip Corrosion | No. 3 max. |
| | Cetane # | 35 min. |
| j. | Fire pump diesel engine by inspection during shut down in accordance with procedures prepared in conjunction with manufacturers recommendations and verifying the diesel, starts from ambient conditions on the auto start signal and operates for $\geq$20 minutes while loaded with the fire pump. | Once/18 months |

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2. If the fire protection systems smoke and/or heat detectors in Tables 3.12.1 and 3.12.2 cannot be restored to an operable status within 14 days, a written report to the Commission outlining the action taken, the cause of inoperability and plans and schedule for restoring the detectors to an operable status shall be prepared and submitted within 30 days.

F. Fire Barrier Penetration Seals

1. All fire barrier penetrations, including cable penetration barriers, fire doors and fire dampers, in fire zone boundaries protecting safety related areas shall be functional.

2. With one or more of the required fire barrier penetrations non-functional, within one hour establish a continuous fire watch on at least one side of the affected penetration or verify the operability of fire detectors on at least one side of the non-functional fire barrier and establish an hourly fire watch patrol. Restore the non-functional fire barrier penetration(s) to functional status within 7 days or, in lieu of any other report required by Specification 6.9.A, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.B within 30 days outlining the action taken, the cause of the non-functional penetration and plans and schedule for restoring the fire barrier penetration(s) to functional status.

F. Fire Barrier Penetration Seals

1. All fire barrier penetration seals for each protected area shall be visually inspected once/1.5 years to verify functional integrity. For those fire barrier-penetrations that are not in the as-designed condition, an evaluation shall be performed to show that the modification has not degraded the fire rating of the fire barrier penetration.*

2. Any repair of fire barrier penetration seals shall be followed by a visual inspection.

* The current surveillance interval for visually inspecting fire barrier penetration seals is extended until May 15, 1992. This is a onetime extension, effective only for this inspection interval. The next surveillance interval began September 27, 1991.

## 3.12 and 4.12 BASES

The Fire Protection System specifications provide pre-established minimum levels of operability to assure adequate fire protection during any operating condition including a design basis accident or safe shutdown earthquake.

A. The high pressure water fire protection system is supplied by redundant vertical turbine pumps, one diesel driven and one electric motor driven, each design rated 2500 gpm at 125 psig discharge pressure. Both pumps take suction from the plant insure. Both pumps take suction from the plant intake cooling water structures from Lake Ontario. The high pressure water fire protection header is normally maintained at greater than 115 psig by a pressure maintenance subsystem. If pressure decreases, the fire pumps are automatically started by their initiation logic to maintain the fire protection system header pressure. Each pump, together with its manual and automatic initiation logic combined makes up a redundant high pressure water fire pump.

A third fire pump, diesel-driven, has been installed and is set to automatically actuate upon decreasing pressure after the actuation of the first two fire pumps. No credit is taken for this pump in any analyses and the requirements of Technical Specifications 3.12 and 4.12 do not apply.

Pressure Maintenance subsystem checks, valve position checks, system flushes and comprehensive pump and system flow and/or performance tests including logic and starting subsystem tests provide for the early detection and correction of component failures thus ensuring high levels of operability.

B. Safety related equipment areas protected by water spray or sprinklers are listed in Table 3.12.1. Whenever any of the protected areas, spray or sprinklers are inoperable continuous fire detection and backup fire protection equipment is available in the area where the water spray and/or sprinkler protection was lost.

Performance of the tests and inspections listed in Table 4.12.1 will prevent and detect nozzle blockage or breakage and verify header integrity to ensure operability.

C. The carbon dioxide systems provide total flood protection for eight different safety related areas of the plant from either a 3 ton or 10 ton storage unit as indicated in Table 3.12.2. Both $CO_2$ storage units are equipped with mechanical refrigeration units to maintain the storage tank content at 0°F with a resultant pressure of 300 psig. Automatic smoke and heat detectors are provided in the $CO_2$ protected areas and initiation is automatic and/or manual as indicated in Table 3.12.2. For any area in which the $CO_2$ protection is made or found to be inoperable, continuous fire detection is available and one or more large wheeled $CO_2$ fire extinguisher is also available for each area in which protection was lost.

Weekly checks of storage tank pressure and level verify proper operation of the tank refrigeration units and availability of sufficient volume of $CO_2$ to extinguish a fire in any of the protected areas.

analyses

5.5.B Bases

The spent fuel pool and high density fuel storage racks are Class I structures designed to store up to 2,797 fuel bundles. The storage racks are designed to maintain a subcritical configuration having a multiplication factor ($k_{eff}$) less than 0.95 for all possible operational and abnormal conditions. The nuclear criticality analysis for the Spent Fuel Racks (References 1 and 3) concludes that fresh fuel bundles with 3.3 w/o U-235 meet the 0.95 $k_{eff}$ limit. This design basis bundle was reanalyzed to determine its infinite lattice multiplication factor, $k_{\infty}$, when in a reactor core geometry (Reference 2). This $k_{\infty}$ was obtained under conservative calculational assumptions and reduced by 2.33 times the standard deviation in the calculation resulting in the Technical Specification limit of 1.36.

References:

1) Increased Spent Fuel Storage Modification, Stone & Webster Engineering Corporation, Boston, Mass. March 15, 1978.

2) General Electric letter, P. Van Dieman to G. Rorke, FitzPatrick Fuel Storage K-infinity Conversion, Revision 1, dated July 10, 1986.

3) Increased Storage Capacity for FitzPatrick Spent Fuel Pool, Holtec International, Mount Laurel, New Jersey, February, 1989.

conclude

an

2. An SRO or SRO with a license limited to fuel [illegible] shall directly supervise all Core Alternations. This person shall directly supervise all Core Alterations. This person shall have no other duties during this time;

Alterations

3. A fire brigade of five (5) or more members shall be maintained on site at all times. This excludes two (2) members of the minimum shift crew necessary for safe shutdown and any personnel required for other essential functions during a fire emergency;

4. In the event of illness or unexpected absence, up to two (2) hours is allowed to restore the shift crew or fire brigade to the minimum complement.

5. The Operations Manager, Assistant Operations Manager, Shift Supervisor and Assistant Shift Supervisor shall hold a SRO license and the Senior Nuclear Operator and the Nuclear Control Operator shall hold a RO license or an SRO license. 1

6. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, health physicists, auxiliary operators, and maintenance personnel who are working on safety-related systems.

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating.

However, in the event that unforeseen problems require substantial amounts of overtime to be used or during extended periods of shutdown for refueling, major maintenance or major modifications, on a temporary basis, the following guidelines shall be followed:

a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

c. A break of at least eight hours should be allowed between work periods, including shift turnover time.

d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Resident Manager or the General Manager - Operations, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Resident Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized. 1

Radiological

6.3 PLANT STAFF QUALIFICATIONS

6.3.1 The minimum qualifications with regard to educational background and experience for plant staff positions shown in FSAR Figure 13.2-7 shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions; except for the Radioligical and Environmental Services Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.3.2 The Shift Technical Advisor (STA) shall meet or exceed the minimum requirements of either Option 1 (Combined SRO/STA Position) or Option 2 (Continued use of STA Position), as defined in the Commission Policy Statement on Engineering Expertise on Shift, published in the October 28, 1985 Federal Register (50 FR 43621). When invoking Option 1, the STA role may be filled by the Shift Supervisor or Assistant Shift Supervisor. (1)

6.3.3 Any deviations will be justified to the NRC prior to an individual's filling of one of these positions.

NOTE:

(1) The 13 individuals who hold SRO licenses, and have completed the FitzPatrick Advanced Technical Training Program prior to the issuance of License Amendment 111, shall be considered qualified as dual-role SRO/STAs.

6.4 RETRAINING AND REPLACEMENT TRAINING

A training program shall be maintained under the direction of the Training Manager to assure overall proficiency of the plant staff organization. It shall consist of both retraining and replacement training and shall meet or exceed the minimum requirements of Section 5.5 of ANSI N18.1-1971.

The retraining program shall not exceed periods two years in length with a curriculum designed to meet or exceed the requalification requirements of 10 CFR 55, Appendix A. In addition, fire brigade training shall meet or exceed the requirements of NFPA 27-1975, except for Fire Brigade training sessions which shall be held at least quarterly. The effective date for implementation of fire brigade training is March 17, 1978.

6.5 REVIEW AND AUDIT

Two separate groups for plant operations have been constituted. One of these, the Plant Operating Review Committee (PORC), is an onsite review group. The other is an independent review and audit group, the offsite Safety Review Committee (SRC).

10 CFR 55.59

## 7.0 REFERENCES


(1) E. Janssen, "Multi-Rod Burnout at Low Pressure," ASME Paper 62-HT-26, August 1962.

(2) K.M. Backer, "Burnout Conditions for Flow of Boiling Water in Vertical Rod Clusters," AE-74 (Stockholm, Sweden), May 1962.

(3) FSAR Section 11.2.2.

(4) FSAR Section 4.4.3.

(5) I.M. Jacobs, "Reliability of Engineered Safety Features as a Function of Testing Frequency," Nuclear Safety, Vol. 9, No. 4, July-August 1968, pp 310-312.

(6) Benjamin Epstein, Albert Shiff, UCRL-50451, Improving Availability and Readiness of Field Equipment Through Periodic Inspection, July 16, 1968, p. 10, Equation (24), Lawrence Radiation Laboratory.

(7) I.M. Jacobs and P.W. Mariott, APED Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards - April 1969.

(8) Bodega Bay Preliminary Hazards Report, Appendix 1, Docket 50-205, December 28, 1962.

10 CFR 50.54

(9) C.H. Robbins, "Tests of a Full Scale 1/48 Segment of the Humbolt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.

(10) "Nuclear Safety Program Annual Progress Report for Period Ending December 31, 1966, Progress Report for Period Ending December 31, 1966, ORNL-4071."

(11) Section 5.2 of the FSAR.

(12) TID 20583, "Leakage Characteristics of Steel Containment Vessel and the Analysis of Leakage Rate Determinations."

(13) Technical Safety Guide, "Reactor Containment Leakage Testing and Surveillance Requirements," USAEC, Division of Safety Standards, Revised Draft, December 15, 1966.

(14) Section 14.6 of the FSAR.

(15) ASME Boiler and Pressure Vessel Code, Nuclear Vessels, Section III. Maximum allowable internal pressure is 62 psig.

(16) 10CFR50.54, Appendix J, "Reactor Containment Testing Requirements."

(17) 10CFR50, Appendix J, February 13, 1973.

10 CFR 50